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Vol. 2

Long-Range Research Plan

FY 1986-FY 1990

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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NUREG-0784, "Long-Range Research Plan, FY 1984-1988," Office of Nuclear Regulatory Research, published August 1982.

NUREG-0961, "Long-Range Research Plan, FY 1984-1988," Office of Nuclear Regulatory Research, published April 1983.

NUREG-1080, Vol. 1, "Long-Range Research Plan, FY 1985-1989," Office of Nuclear Regulatory Research, published September 1984.

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PREFACE

This Long-Range Research Plan (LRRP) is intended to provide the Commission with a framework for planning research relevant to current regulatory objectives or to future needs. It was developed in accordance with the Commission policy and planning guidance presented below.

Policy*

1. The purpose of the research program is to provide the technical basis for rulemaking and regulatory decisions; to support licensing and inspection activities; to assess the feasibility and effectiveness of safety improvements; and to increase our understanding of phenomena for which analytical methods are needed in regulatory activities.

2. There should be continued emphasis on using research results in the regulatory process and on obtaining results that are useful therein.

3. The severe accident research program must provide timely information for the Commission's decisionmaking process on severe accidents.

Planning Guidance

1. The research resources identified in NRC's budget should be allocated to support a balanced program between research to reinforce or revise the current regulatory base and conceptual research for improved reactor safety, waste management, and other licensed activities. The major resource commitment for NRC research efforts will be light water reactor safety. The staff should be alert to research which shows that we ought to change our regulations. NRC regulations should be changed when research shows them to be either too stringent or not stringent enough.

2. NRC will continue to maintain a long-range research plan which is consistent with the agency's mandate and directed toward areas of importance to the licensing and inspection processes. The research plan will be revised and updated annually and subjected to agency-wide review. The long-range research plan should continue to identify regulations likely to be substantively modified or substantiated by the research programs. Research undertaken by the staff will be consistent with the long-range plan.

3. The staff will conduct annual assessments, with input from appropriate user offices, of the progress and usefulness of specific research topics and make greater use of Research Review Groups.

*This is the section on research in NUREG-0885, Issue 4, "Policy and Planning Guidance, 1985."

4. Joint or coordinated research programs with industry groups, other government agencies and foreign groups should be pursued when possible, both to expand the technical breadth provided to projects and to maximize the benefit to be derived from limited resources. Due consideration should be given to questions of conflict of interest when contemplating joint or coordinated research with indus ry.

The senior management of user offices reviews and endorses the research program at two points in the planning process: (1) LRRP and (2) budget preparation.

User offices are asked to endorse the following in the LRRP:

1. That the regulatory needs are comprehensive and are accurately stated.

2. That the priorities assigned each need are appropriate and the expected completion dates of supporting research are consistent with NRC needs.

3. That the research product can reasonably be expected to provide the information needed to help resolve the associated regulatory need.

4. That the level of expenditures for each program appears cost effective relative to the research deliverables, resolution of the associated regulatory needs, and the inherent level of difficulty (experimental or analytical technique).

The primary objective of NRC research is to support the regulatory process and contribute to improved reactor safety. The goal of the research planning process is to develop a program with a reasonable balance between near-term (those supporting current regulatory and licensing activities) and longer-term regulatory needs.

INTRODUCTION

The Nuclear Regulatory Commission's mission--regulation to ensure that civilian activities involving the use of nuclear materials and facilities are conducted in a manner consistent with protecting the public health and safety, the quality of the environment, and the national security--calls for the exercise of the regulatory functions of rulemaking, licensing review, and inspection and enforcement.

In the process of carrying out its mission, the Commission makes policy decisions involving complex technical issues and varied and conflicting public attitudes. The Commission must base these decisions on an accurate understanding of the technical factors involved, and the NRC staff is responsible for ensuring that the Commission is fully informed.

In its responsibility for supporting the Commission's decisionmaking, the NRC staff must maintain continuing awareness and understanding of public concerns and evolving understanding of issues that might signify a need for changes in the regulatory process. In addition to maintaining a state-of-the-art analytical capability to ensure the technical accuracy of its assessments, the staff must use those new insights gained from operating experience as a basis for reassessing technical criteria with the goal of improving the regulatory process. The staff is assisted in these areas by the research program of the Office of Nuclear Regulatory Research (RES).

The research program provides the technical basis for rulemaking and regulatory decisions to support licensing and inspection activities, to assess the feasibility and effectiveness of safety improvements, and to increase our understanding of phenomena for which analytical methods are needed in regulatory activities.

The major objective of the NRC research program is to provide an understanding of phenomenology and verified analytical methods to permit identification of important accident sequences and well-founded realistic (or best-estimate) analysis of their consequences. To this end, much of the research program consists of a mixture of experimental work and code development work aimed at understanding complex system transients. Because the data points from large, complex, integral facilities tend to be few in number and of limited applicability, current and future research is based on experiments with a smaller scale than some earlier experiments to ensure cost effectiveness. The data obtained will be used to validate codes for use in safety analyses. Other objectives are to provide the methodology to make more effective use of probabilistic risk assessment in the regulatory process and to improve confidence in the data base for risk assessment. This combination of experiments, code work, and risk analyses will produce thoroughly validated codes for use by licensing reviewers and will identify the areas in the regulatory process where improvements are needed. Also using the research will be response personnel for the purpose of gaining more understanding of accidents and thus improving the quality of Operations Center drills, of plant condition assessments, and of protective action decisionmaking recommendations.

Development of this Long-Range Research Plan (LRRP) is the first step in the process of ensuring that the Commission's research program is directed toward areas of importance to the regulatory program. The LRRP is intended to assist the Commission in establishing priorities to ensure effective utilization of limited resources. It identifies broad regulatory issues and describes programmatic approaches for research to support the resolution of these issues over a 5-year period.

The Long-Range Research Plan is distributed broadly within the NRC for review and comment. In addition, it is distributed to DOE and to such industry groups as the Electric Power Research Institute (EPRI). The Commission believes that DOE and the nuclear industry have a major responsibility to perform safety research to ensure that nuclear power plants and other nuclear facilities are designed and operated safely and reliably. The distribution to DOE and the industry groups is intended to foster cooperation and coordination among NRC, DOE, and the nuclear industry to ensure that the appropriate level of effort is directed toward resolving safety issues and to prevent unnecessary duplication. The LRRP is updated annually to reflect completed tasks, to identify new regulatory and research needs, and to incorporate comments on the plan of the previous year.

The final selection of research programs is based on the Commission guidance; the needs submitted to RES by other NRC offices; the comments and technical insights from the Advisory Committee on Reactor Safeguards (ACRS), industry, the public, the national laboratories, and international organizations; and the availability of resources to ensure timely delivery of research results. This LRRP is divided into the following chapters, each of which covers a major research program:

<u>Chapter 1. Operating Reactor Inspection, Maintenance, and Repair</u>. This research program will study such time-related issues as the mechanisms of aging and degradation, methods of examination and testing to determine the condition of components, and interpretation of results of these tests for appropriate action. This work will provide the bases by which the staff can assess with confidence industry test and examination methods and results. These assessments in turn provide bases for licensing decisions on whether operating plants continue to meet health and safety requirements in effect at the time of licensing and subsequently imposed health and safety requirements.

<u>Chapter 2. Equipment Qualification</u>. This research program will study the methods used for qualifying equipment used in nuclear power plants taking into account such factors as effects of synergism, order or sequence of tests, accelerated aging techniques, and methods for simulating accident environments. Methods will be validated and new methods developed as appropriate to ensure that qualification test results reported by applicants and licensees provide a basis for licensing decisions that ensure protection of the public health and safety.

<u>Chapter 3. Seismic Research</u>. This research program will study earthquakes, which are among the most severe of the natural hazards faced by nuclear power plants. The seismic hazard of nuclear power plants on the eastern seaboard of the United States has become a major regulatory issue. The question is whether

an earthquake of the magnitude of the Charleston earthquake could occur elsewhere on the eastern seaboard, placing a nuclear power plant at hazard. Another issue is the public risk associated with seismic events in excess of the current design basis, the safe shutdown earthquake (SSE). The third seismic issue is the margin inherent in the seismic design of older plants. New information available on seismicity, soil-structure interaction, and plant seismic response can result in changes in the predicted loads on structures, systems, and equipment. These changes must be compared to the inherent margin of the original design before a decision can be made.

Chapter 4. Reactor Operations and Risk. This research program supports the development of probabilistic risk assessment (PRA) methods and their use within the regulatory structure to identify those elements of reactor operations that are the most significant contributors to risk and the causal factors associated with them and to permit comparative evaluations of risk levels associated with various regulatory actions. Past efforts in this area have identified the man-machine interactions as an area of significant uncertainty and therefore a potentially large contributor to risk. This work includes the development and trial use of models, methods, procedures, and other analyses required to support Commission decisions on a broad range of critical issues relating to power reactor safety and the acquisition of data to support the application of PRA methods to the regulatory process.

Chapter 5. Thermal-Hydraulic Transients. This research program provides the experimental data and analytical methods needed to predict and understand the operation of primary and secondary coolant systems during all types of plant transients, including the full range of pipe rupture sizes. The resulting analytical methods are used to quantify margins of Appendix K to 10 CFR Part 50, to assist the regulatory assessment of operator guidelines for accident management, and to analyze the operational control of complex plant system transients as well as their recovery to normal conditions.

Chapter 6. Severe Accidents. This research program provides the data base and validated methodology for the reassessment of the regulatory treatment of severe accidents. It includes the coordinated phenomenological research programs needed to develop a sound technical basis for NRC decisions concerning the ability of reactors to cope with these accidents. The following elements are included in this chapter: accident likelihood evaluation, severe accident sequence analysis, behavior of damaged fuel, hydrogen generation and control, fuel-structure interaction, containment analysis, fission product release and depletion or transport, containment failure mode, fission product control, risk code development, accident consequence and risk reevaluation, and risk reduction and cost analysis.

Chapter 7. Radiation Protection and Health Effects. This research program provides a technical and scientific data base that will permit identification and measurement of sources of radiation exposure and a definition of the relationship between exposure and potential health effects, thus allowing radiation exposure limits to be set with a sound scientific basis. Uncertainties in these areas can result in policies that are either overly protective, which would not be cost effective, or too lax, which would expose workers and the public to unacceptable risks. Significant uncertainties remain in some areas of radionuclide metabolism and internal dosimetry, in the characteristics of dose-effect relationships, and in radiological dose measurements in the work place.

Chapter 8. Waste Management. This research program will provide the technical capability to assess compliance of a waste management system with the regulatory requirements for operational safety, occupational radiological protection, and long-term waste isolation.

The proposed funding levels for the major research program areas described in the LRRP for Fiscal Years 1986-1990 are shown in Table 1.

A correlation between needs and research products is provided in this LRRP, i.e., Research Product 1 refers to Need 1. Dates* are included after each statement of regulatory need and each research product. The date provided after the research product indicates when the research should be available; the date provided after the statement of regulatory need indicates when the regulatory product should be available. The symbol † following a statement of regulatory need points out that a modification of the regulations may result from the research.

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*FY (for fiscal year) has been omitted from the targeted dates provided in this LRRP. It may be assumed, however, that the period of time indicated is the fiscal year.

Table 1

FUNDING LEVELS*

FY 1986-FY 1990

	FY 1986	FY 1987	FY 1988	FY 1989	FY 1990
Operating Reactor Inspection, Maintenance, and Repair	$ 24.2	$ 28.1	$ 26.1	$ 26.4	$ 25.3
Equipment Qualification	5.8	5.4	3.4	2.5	1.0
Seismic Research	10.0	12.1	14.0	9.4	8.9
Reactor Operations & Risk	13.0	12.3	10.2	10.0	10.5
Thermal-Hydraulic Transients	20.9	23.4	21.5	20.0	20.0
Severe Accidents	39.8	35.1	28.5	21.0	18.5
Radiation Protection and Health Effects	2.5	2.2	2.5	2.8	3.0
Waste Management	4.8	9.9	10.2	10.8	11.6
TOTAL	$121.0	$128.6	$116.4	$102.9	$ 98.8

*Dollars in millions.

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1. OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR

Research is needed to study and understand time-related issues such as the mechanisms of aging and degradation, methods of examination and testing to determine the condition of components, and interpretation of results of these tests for appropriate action. This work will provide the bases by which the staff can assess with confidence industry test and examination methods and results. These assessments in turn provide bases for licensing decisions on whether operating plants continue to meet health and safety requirements in effect at the time of licensing and subsequently imposed health and safety requirements.

1.1 Reactor Vessels

This research applies to the structural integrity of pressure vessels especially as affected by irradiation embrittlement and growth of postulated cracks in service.

1.1.1 Major Regulatory Needs and Their Justifications

1. Validated methodology for fracture analysis of reactor vessels under accident conditions, to provide a basis for the development of licensing criteria and a regulation and to assist the staff in assessing submittals implementing the actions taken to resolve unresolved safety issue (USI) A-49 (1986).†
   Justification: The validated fracture analysis methodology is needed to enable the staff to independently evaluate vendor and utility submittals concerning the ability of a reactor vessel to safely withstand conditions imposed during accidents. Note that Section III of the ASME Code is fundamentally a design code, and certain aspects of aging are beyond its scope; thus, safety criteria for operating vessels (which are subject to aging degradation) must be developed by NRC.

2. Data base on fracture toughness and crack arrest toughness of irradiated vessel steel and weld metal, to provide a basis for the development of licensing criteria, an amendment to the regulations, a revision to Regulatory Guide 1.99, and recommendations for a possible update of Section XI of the ASME Code and to assist the staff in assessing submittals implementing the actions taken to resolve USI A-49 (1986).†
   Justification: Generic and specific data are required for irradiation effects on the fracture initiation toughness and crack arrest toughness of vessel steels and weld metal so that safety decisions on vessel integrity can be made. Otherwise, the staff would not know how much a vessel steel had degraded or what was an acceptable level to ensure continued safety.

†A modification of the regulations may result from the research.

3. Experimentally validated surveillance methodology to allow accurate predictions of neutron fluence and radiation effects so as to provide bases for recommendations for updating American Society for Testing and Materials (ASTM) standards that are or will be endorsed by regulatory guides (1987). Justification: Neutron flux can be measured with reasonable accuracy in experimental facilities. In these test reactor facilities, irradiated Charpy V notch specimens are tested to establish the empirical relationships between fluence and the reduction of the Charpy specimen's fracture toughness. In an operating reactor, the core flux leakage calculations form the basis for our predictions of the fluence impinging on the vessel wall during the lifetime of the plant. These predictions are validated indirectly by periodically testing the Charpy specimens from the vessel's surveillance capsules and then using the developed relationships to establish the effective fluence. This procedure has shown that significant errors exist in the calculational methods used to predict fluence and radiation effects. The research will reduce the error band in these calculation methodologies and hence significantly improve the ability to accurately assess the structural integrity of these structures at any period in their expected life during both normal and postulated accident conditions.

4. Data base on environmentally assisted fatigue crack growth rate in vessel steels and welds, to be the basis for licensing criteria and for recommendations for updating Section XI of the ASME Code (1988). Justification: Knowledge of the rate of environmentally assisted fatigue crack growth in nozzles, piping, and vessels is necessary to decide if cracks or flaws discovered during inspections can grow to critical size in subsequent operation (and thus must be removed) or if they can be allowed to remain as benign imperfections with no potential impact on the safety of the primary system during normal operations or accidents.

5. Validated methodology for annealing to recover material fracture toughness properties of irradiated reactor vessels, to develop licensing criteria and recommendations for updating ASTM standards and Section XI of the ASME Code (1989). Justification: The validation of annealing methodology for recovery of properties degraded due to vessel irradiation is needed so that safety decisions on vessel integrity can be made based on actual engineering test data and with a minimum of risk of impacting continued safe operation.

1.1.2 Research Program Description

The ability of the NRC licensing staff to make decisions concerning the present and continuing safety of reactor pressure vessels under both normal and abnormal operating conditions is dependent upon the existence of verified analysis methods and a solid background of applicable experimental data. This program is to provide both the analytical methods and the experimental data needed. Specifically, this program develops fracture mechanics analysis methods and design criteria for predicting the stress levels and flaw sizes required for crack initiation, propagation, and arrest in light-water-reactor (LWR) pressure vessels under all known and postulated operating conditions. To do this, not only must the methods be developed but they must be experimentally validated. Further, the materials data necessary for input to these analytical methods

must also be developed. Thus, in addition to methods development and large-scale experimental verification, this program also develops data to show that slow-load fracture toughness, rapid-load fracture toughness, and crack-arrest toughness obtained from small laboratory specimens are truly representative of the toughness characteristics of the material behavior in pressure vessels in both the unirradiated and irradiated conditions.

Another significant role of this program is to determine the ways and the extent to which the LWR environment (particularly radiation) changes and degrades the pressure vessel materials during their operational life. Thus, elements of this program deal with the determination of the sensitivity of the pressure vessel's steel to fracture toughness degradation as a function of irradiation exposure and with methods such as thermal annealing required to regain this toughness parameter. Also part of this program are studies to improve and standardize dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effects of neutron exposure to these steels.

The strategy for the research in this element is (1) to develop experimentally verified fracture mechanics analysis techniques that can be rapidly brought to bear in resolving licensing issues dealing with the assurance of reactor pressure vessel integrity during normal operation and postulated upset or accident conditions and (2) to establish statistically acceptable material data bases to be used in conjunction with the presently accepted and newly developed analytic techniques.

The research effort is divided into two phases: (1) relatively short-term, high-priority programs to develop improved methodologies and sufficient supporting data to be used in establishing generally acceptable and defensible regulatory positions on critical issues such as pressurized thermal shock, structural performance of low upper-shelf energy materials, and irradiation embrittlement rates of presently operating reactor pressure vessels; and (2) a longer-term effort to establish procedures for degraded material fracture toughness recovery and the revision or replacement of existing standards, codes, and criteria that deal with the fabrication and expected aging effects of reactor pressure vessels. Incorporated into this research program will be work done by outside groups (Electric Power Research Institute (EPRI), several European governments and the Japanese government, and the efforts of European technical commmunity groups such as the Program for Inspection of Steel Components (PISC)), which is anticipated to approximately equal the NRC research effort in costs.

The short-term, high-priority effort will be completed by 1986 with the longer, confirmatory research reaching completion by 1989.

The major research products will be:

1. a. Large-scale verification of unified fracture mechanics methodology by pressurized thermal shock and wide plate tests, PTSE-2 and WPT series (1986).

   b. Completion of summary report, detailing in one document the accomplishments of AEC/NRC pressure vessel research program, its

impact on regulation, and remaining areas of needed research (1986).

c. Completion of development of a "visco-plastic" mechanism model to define rate effects on propagation and arrest of rapidly running crack (1988).

d. In cooperation with industry, complete development of statistical data base defining defect distribution in as-fabricated reactor pressure vessels (1989).

2. a. Establishment of basis for NRC position on ASTM standard on crack arrest testing specimen (1986).

b. Development of basis for fracture toughness requirements under conditions of thermal shock to reactor vessels (1986).

c. Irradiated specimen test completed to validate ASME Section XI fracture toughness curves (1987).

d. Technical bases for revising 10 CFR Part 50 governing reactor fracture toughness requirements under both normal and accident operating conditions (1988).

e. Development of crack arrest data base for irradiated steels and weldments for first generation and present practice reactor pressure vessels (1989).

f. Review and evaluation of industry-developed fracture toughness data base for new, high-strength materials for next generation reactor pressure vessels (1990).

3. a. Documentation of metallurgy and dosimetry for Pool-Side Facility simulated vessel wall and void box benchmarks, basis for pressurized thermal shock (PTS) embrittlement extrapolation and shield tank support column embrittlement predictions (1986). (Also applies to Needs 1 and 2.)

b. Development of basis for drop in upper-shelf energy toughness criterion (1987).

c. Complete development of irradiation damage mechanism model to allow accurate transition temperature shift predictions beyond empirically developed trend curves (1988). (Also applies to Need 2.)

d. Extension of interpretation of data bases and irradiation damage mechanism model to allow prediction of effects of irradiation damage to post-heat-treat annealed reactor pressure vessels (1990). (Also applies to Need 5.)

4. a. Revision of fatigue curves for ferritic materials in ASME Section III (1987).

b. Final revisions recommended for the environmentally assisted fatigue curves in ASME Section XI (1988).

5. a. Development of methodology (engineering design and instrumentation requirements) for guidance on cooperating with industry in a full-scale vessel annealing experiment (1987).

   b. In cooperation with industry, completion of construction of full-scale vessel annealing equipment (1988).

   c. In cooperation with industry, completion and experimental validation of methodology for recovering fracture toughness properties by in situ annealing (1989).

1.2 Steam Generators

The research discussed below deals with corrosion, cracking, and degradation of steam generator tubing during service; integrity of tubing as degraded by the water and stress environment during normal operation and upset conditions; and integrity of tubing over the long term as affected by decontamination and tube bundle cleaning and by other causes.

1.2.1 Major Regulatory Needs and Their Justifications

1. Validated data on integrity of tubing having cracks, dents, wastage, and other forms of degradation, to be the basis for licensing criteria and revisions to regulatory guides (1986).
   Justification: Steam generator tubes have been and are degrading in the form of cracks and general wastage. The staff must know the potential remaining integrity in tubes having various degrees and types of degradation, cracks, etc., for requirements on tube plugging and additional inspection of tubes. If inspection "indications" translate to potential cracking and leakage, plugging or augmented inspections would be required; if indications are benign, the plant can be returned to service.

2. Correlation of nondestructive examination (NDE) signals with tube integrity, to be the basis for licensing criteria and revisions to regulatory guides (1986).
   Justification: The only way to predict tube integrity is from knowledge of the signal taken from inspection. Thus, signal evaluation errors must be reduced, and signals must be carefully correlated to exact defect type and size, as well as to measurements of tube burst strength.

3. Long-term effect of decontamination, cleaning operations, and repairs on tube integrity and on vibration aspects of tubes, to provide an improved basis for approval of applications (1987).
   Justification: Processes are proposed and in use for decontaminating the primary side (to reduce man-rems), for cleaning the secondary side (to help reduce corrosion problems), and for repairs. The chemical or mechanical means employed can possibly damage the tubes, induce large residual stresses, or leave corrosive residues that can continue to attack the tubing, negate cleaning, and hasten future cracking. In addition, the increased clearance between supports and tubes as a result of chemical

cleaning may induce additional vibration and cause unacceptable fretting and wear of the tubes during operation. The limits on acceptable increased clearances are not known. Because the long-term effects are not known, they must be independently studied so that informed decisions can be made on applications from utilities.

1.2.2 Research Program Description

Research on steam generators at NRC is focused on the Steam Generator Group Project at Richland, Washington, where four cosponsors from the United States and abroad have joined with NRC to conduct the program described below. The program aims to develop validated models, based on experimental data, for prediction of margins-to-failure under burst and collapse pressures of steam generator tubing found to be service degraded by eddy current inservice inspection. This is to be accomplished by using an out-of-service degraded steam generator as a test bed for a confirmatory research program that includes NDE development/validation; optimization of inservice inspection procedures, sampling plan, and inspection period; validation of tube integrity predictive models; optimization of tube plugging criteria; and evaluation of proposed chemical cleaning and decontamination processes and procedures with respect to near-term integrity and long-term effects on corrosion, degradation, and safety.

EPRI has under way some steam generator reliability work, a large amount of which may be validated in the NRC steam generator project through EPRI's participation in that project. The project has become a focus for international efforts in steam generator research as France, Italy, and Japan, as well as EPRI, have joined the work through financial contributions.

Specifically, service-degraded tubes from a retired-from-service steam generator will be used to compare and validate different and advanced NDE methods that will show the best methods for detecting and characterizing flaws, to remove these tubes and precisely characterize the type and extent of cracking or other degradation, and to subject the tubes to pressure in burst or collapse mode to establish the residual strength in the tubes. In this manner, an exact correlation can be developed between the flaw signal, as detected nondestructively, and the tube integrity. Thus, the licensing criteria for tube inspection plans and tube plugging can be validated or modified as needed to reflect the research findings. Of particular value is the ability to use service-degraded tubes with flaw and degradation characteristics that have been carefully documented to validate advanced eddy current NDE methods as well as models for predicting stress corrosion cracking in tubing. (NDE research for steam generators is also discussed in Section 1.5.) The longer-term safety and integrity implications of decontamination and of cleaning and crud removal are examined to ensure that application of such procedures will not create future problems.

The major research products will be:

1. a. Burst and leak rate testing of tubes removed from generator (1986). (Also applies to Need 2.)

   b. Validation of current and advanced NDE results through examination of removed tubes (1986). (Also applies to Need 2.)

2. Correlation of remaining tube integrity from burst and leak tests with NDE to validate regulatory guide inservice inspection (ISI) plans and tube plugging criteria (1986).

3. a. Evaluation and recommendations on possible tube vibration and damage during operation after chemical cleaning (1986).

   b. Demonstration of long-term generator integrity after cleaning and decontamination procedures (1987).

1.2.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

As a result of budget reductions in FY 1986, the program to demonstrate long-term generator integrity after cleaning and decontamination procedures is unfunded.

1.3 Piping

This research applies to the structural integrity of piping degraded during service by the water, stress, and temperature environment. This degradation is in the form of stress corrosion cracking, fatigue and cyclic crack growth, and toughness loss because of long-time aging at temperature. Evaluation of the factors causing cracking and of proposed fixes is included. Pipe rupture investigations are also a part of this research program.

1.3.1 Major Regulatory Needs and Their Justifications

1. Experimentally validated analysis methodology for the loading capacity of flawed and degraded piping during normal operation, accidents, and earthquakes; validation of the leak-before-break concept; and data on the true failure modes of cracked piping, to provide the basis for new or modified regulatory guides, regulations, and standard review plans (1987).†
   Justification: Decisions are regularly needed on the safety of pipes and welds containing flaws or cracks discovered during inservice inspections. Even if a cracked pipe could withstand normal operating loads, it might not be able to withstand the loads from all postulated accidents and earthquakes. These concerns have been heightened by the occurrence of stress corrosion cracking in large-diameter BWR piping. Furthermore, a basis is required for deciding if pipes will leak prior to break and, thus, if massive pipe whip restraints are needed or not. In the event that the postulated full-flow area break (guillotine or longitudinal slot) is eliminated, replacement criteria will have to be developed.

2. Independent basis for evaluating factors causing stress corrosion cracking and proposed fixes in stainless steel piping and welds, to be used in dealing with operating reactor problems (1986-1987).
   Justification: Decisions are regularly required on the "fixes" proposed to eliminate stress corrosion cracking in stainless steel piping, including repairs and remedies. Background information is needed for independent evaluation of these fixes and repairs. In particular, procedures have

†A modification of the regulations may result from the research.

been developed in the United States and elsewhere (including Japan and Sweden), but the presently available data bases are generally not sufficient for the needs of NRC for use in regulatory decisions regarding long-term integrity. Of special concern is BWR pipe cracking.

3. Data base on crack growth rate in piping steel and welds, to be used in developing licensing criteria and recommendations for updating Section XI of the ASME Code (1988).
   Justification: Knowledge of rate of growth of cracks under operating environment and loading conditions is necessary to decide if cracks or flaws discovered during inspections can grow to critical size in subsequent operation (and thus must be removed) or if they can be allowed to remain as benign imperfections with no potential impact on the safety of piping during normal operations or accidents.

4. Data base for evaluating toughness loss in existing cast duplex (austenite and ferrite) stainless steel components from long-term aging at reactor operating temperature and for making recommendations to limit long-term aging in new components, to be the basis for developing a regulatory guide and revising the standard review plan (1988).
   Justification: A certain level of toughness is required to ensure safety in piping and other primary system components, especially to resist failure if flaws should develop in service and under accident loading. Long-term time-at-temperature can cause precipitation of another phase in the ferrite of the duplex stainless steel resulting in a reduced toughness of the original material. The time-temperature-material conditions under which this occurs and the degree to which it occurs in service must be known so that licensing decisions are made in full knowledge of the future strength and toughness condition of piping and other primary system components such as pump casing.

5. Evaluation of aging and environmental degradation in LWR materials, including the effects of temperature, irradiation, and environment, to be the basis for changes in regulations dealing with operations and maintenance of nuclear power plants (1989).†
   Justification: Recent reports have highlighted the possible detrimental effects of hydrogen on stress corrosion cracking of Inconel 600, of irradiation on intergranular stress corrosion cracking (IGSCC) of annealed 304 SS, and of thermal aging on toughness loss of ferritic steels. These phenomena need to be evaluated for their effect on components and materials of interest in the primary system. The degree of degradation as a function of time caused by these phenomena needs to be established for proper safety analyses.

6. A technical basis sufficient for making licensing decisions concerning requirements for postulating pipe ruptures in reactor coolant loop and other safety-class piping (1986).
   Justification: The understanding of the reliability of reactor coolant loop piping of each United States vendor must be improved. This understanding should be expressed in a probabilistic sense and must include an uncertainty analysis.

†A modification of the regulations may result from the research.

1.3.2 Research Program Description

The principal long-term objectives of the piping research program are to determine the validity of the leak-before-break concept in LWR piping systems and to provide the capability to evaluate potential fabrication and operating improvements directed at eliminating pipe cracking. The program for evaluating leak before break in LWR piping systems is a multifaceted effort that will integrate research in the areas of piping degradation modes, piping fracture mechanics, nondestructive examination, leak rates, and leak detection. A major program to validate elastic-plastic fracture mechanics analyses and to develop a material properties data base for piping was initiated in 1983. Initially these programs will address materials that exist in operating plants. The fracture properties of new and replacement materials will be evaluated during later phases of the program. Ongoing programs will provide information on NDE techniques, leak rates, and leak detection systems. The integration of results from these programs will be a continuing effort, culminating in a position on the acceptability of leak before break as a function of the piping system, material of fabrication, and other pertinent factors. Appropriate development of regulatory guides, modifications to the standard review plan, and rulemaking will then be pursued.

Research programs directed at environmentally assisted crack growth and aging effects in piping will provide the necessary basis for evaluating the acceptability of fixes proposed by the industry to eliminate or reduce the frequency of pipe cracking and to eliminate the degree of age-related degradation in piping materials. The data to be generated will be applicable to evaluating improved fabrication and repair procedures, proposed new materials, and changes in operating environment.

Elements of the research program are directed toward providing timely information to assist in the development of licensing criteria regarding stress corrosion cracking in BWR piping. These elements include determination of the reliability of detection and sizing of stress corrosion cracks, identification of improved NDE techniques, development of stress corrosion crack growth rate data, validation of fracture mechanics analyses for evaluating stress corrosion cracks, and evaluation of short-term and long-term fixes. The short-term and long-term fixes of interest include weld overlays, induction heating stress improvement, last-pass heat sink welding, oxygen control, and piping replacement.

The object of pipe rupture investigations is to support the implementation of the leak-before-break hypothesis and the revision of Criterion 4 of Appendix A to 10 CFR Part 50.

The major research products will be:

1. a. Evaluation of pipe cracking predictive models, proposed fixes, and weld repair criteria (1986). (Also applies to Needs 2 and 3.)

   b. Initial findings on toughness of cast stainless steels, including austenitic and other nickel alloy weld material deposited using a flux process such as SMAW (Shielded Metal Arc Welds) and SAW (Submerged Arc Welds), for use in leak-before-break study (1986). (Also applies to Need 4.)

c. Data and conclusion regarding the effectiveness of short-term fixes for pipe cracking (1986). (Also applies to Needs 2 and 3.)

d. Experimental validation for elastic-plastic fracture mechanics analyses (1986).

e. Computerized data base on piping materials fracture toughness and crack growth rates transmitted to NRR for use in licensing evaluations (1987).

f. Evaluation of effectiveness of long-term fixes for pipe cracking (1987). (Also applies to Need 2.)

g. Technical basis for licensing decision on acceptance of leak before break in LWR piping systems (1988).

h. Validation of ductile fracture mechanics analyses for complex piping geometries and components (1989).

2. a. Initial sensitization and IGSCC predictive models developed for evaluation of welding and repair-welding stainless steels (1986).

   b. Licensing criteria proposed for establishing limits on environmental variables to control pipe cracking in LWR piping systems (1987).

3. Acceptance criteria developed for welded and repair-welded stainless steel (1988).

4. Licensing criteria proposed for prevention of toughness degradation due to aging in LWR cast stainless steel piping materials (1988). (Also applies to Need 1.)

5. a. Evaluation of the effect of different system characteristics coupled with hydrogen on stress corrosion cracking (SCC) of BWR piping materials (1986-1987), of thermal aging on ferritic materials, and of irradiation on SCC of internal stainless steel materials (1989).

   b. Recommendations on degree of degradation induced by hydrogen, aging, and radiation exposure of LWR materials and on its significance to structural integrity and safety (1989).

6. a. Probabilistic information on BWR recirculation loop and main steam and main feedwater pipe cracking, leaking, and rupture, for assisting in licensing decisions on replacement materials for recirculation loop piping (1986).

   b. Evaluation of small-break LOCA probabilities (1988).

1.4 Electrical and Mechanical Components (Nuclear Plant Aging Research - NPAR)

This generic nuclear plant aging research applies principally to the time-related degradation of electrical and mechanical components during service and

the potential impacts of degradation of plant systems involving these components upon public safety. The major goals of the program are to (1) identify aging and service wear effects associated with electrical and mechanical components and systems that, if unchecked, could impair plant safety, (2) identify methods of inspection, surveillance, and condition monitoring of electrical and mechanical components and systems that will be effective in detecting significant aging and service wear effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented, and (3) identify and recommend acceptable maintenance practices that can be undertaken to mitigate the effects of aging and to diminish the rate and extent of degradation caused by aging and service wear. The technical knowledge gained from these tasks will be translated into practical application guidelines to monitor and mitigate the aging of components and structures in nuclear power plants. The technical knowledge will also be translated into recommendations for standards and guides concerned with monitoring equipment degradation and the prediction, prevention, and mitigation of equipment failures that can adversely affect public health and safety.

The general nuclear plant aging research program will be coordinated with other aging research under way in NRC, in other United States Government agencies, in industry, and in other countries. This aging research will complement and be closely coordinated with the research on equipment qualification described in Chapter 2, "Equipment Qualification."

1.4.1 Major Regulatory Needs and Their Justifications

1. Assurance that previously unaccounted for aging and service wear effects that could have a significant impact on safety over the life of a nuclear power plant are identified so that modifications to regulatory requirements to mitigate such effects may be developed on a timely basis (1986-1988). Justification: Although aging effects have previously been recognized to be potentially important to nuclear plant safety, only a limited number of Institute of Electrical and Electronics Engineers (IEEE) and ASME standards provide guidance on how to account for aging of electrical and mechanical components. In the case of the IEEE standards, emphasis has been placed on pre-aging (primarily accelerated aging based on the Arrhenius model) prior to qualification testing. However, it is generally recognized that all aging and service wear effects cannot be modeled within the context of the Arrhenius theory. Both IEEE standards and the ASME operations and maintenance standards have recommended needs for equipment surveillance, degradation monitoring, and maintenance. However, the guidance on surveillance and maintenance in these standards is very general in nature, and specific indicators of incipient aging-related defects prior to catastrophic failure modes are not provided. Also, standards have been developed to date for only a limited number of component types. Taking into consideration the multiplicity of equipment types and the variety of instances of aging-related equipment failures or precursors of such failures reported over the years in licensee event reports (LERs), maintenance records, and inspection reports, it is not clear that the national consensus standards have adequately provided guidance on how to account for the aging effects that could degrade plant safety over the expected 40-year life of a typical commercial nuclear plant.

A systematic evaluation is needed to identify potentially significant aging effects, i.e., those that could cause an increase in frequency or severity of plant transients or an unacceptable degradation in the capability of safety equipment to withstand or mitigate design basis events. Such an evaluation should include consideration of the severity of aging processes to equipment and components, the impact of equipment degradation on safety system performance, and the overall potential impact on risk to the public. This evaluation would provide a basis for judgment that the relative importance to safety of the identified aging effect has been appropriately characterized and would guide the development of regulatory requirements.

2. Criteria to be used as the technical basis for evaluation of industry surveillance testing and monitoring, maintenance, and replacement programs to determine whether those programs adequately mitigate aging and service wear effects that could have a significant impact on plant safety (1988-1990).
<u>Justification</u>: National consensus standards include limited guidance regarding surveillance, maintenance, and inservice inspection to account for aging; nuclear plant technical specifications include requirements for periodic surveillance/testing; and utilities have instituted maintenance and replacement programs. However, no regulatory criteria have been developed for evaluating these programs to determine whether significant aging effects can be adequately mitigated. Also, there is general recognition that artificial pre-aging techniques based on the Arrhenius theory do not realistically simulate actual aging and service wear processes and degradation for all types of components and systems. Although artificial pre-aging is currently applicable primarily to electrical equipment to be qualified for harsh environments, consideration is being given to whether mechanical equipment (or at least the nonmetallic materials contained in such equipment) should be pre-aged prior to testing for harsh environments and to whether equipment located in mild environments should be aged prior to seismic and dynamic qualification.

Criteria are needed to (1) evaluate the surveillance, maintenance, and replacement programs instituted by nuclear utilities to determine whether these programs will adequately prevent significant impairment of safety function; (2) evaluate which, if any, surveillance/testing intervals in plant technical specifications should be modified as a function of plant age to account for or to reduce potential aging and service wear effects; and (3) evaluate surveillance monitoring programs developed to supplement and perhaps partially replace artificial pre-aging of equipment prior to qualification testing. (As noted above, current requirements differ depending on whether the equipment is mechanical or electrical and on whether the environment is harsh or mild.)

In all cases, the evaluation criteria should be limited to equipment determined likely to be vulnerable to aging effects that could cause significant impact on plant safety. In addition, for surveillance programs the criteria should be based on indicators of aging or service wear degradation that can be monitored at reasonable cost and with the minimum possible accumulation of occupational exposure to radiation. Proper condition monitoring techniques should effectively indicate the approach of a level of degradation that would render the equipment incapable of performing its safety function during design basis accidents.

For equipment for which surveillance is impractical as a means of determining with confidence the approach to an unacceptable level of degradation, replacement and maintenance schedules may have to be based on the concept of "predicted service life." Although theoretical bases for predicting effective service lifetimes have been and are being developed, there exist considerable uncertainties in the resultant predicted lifetimes because of the statistical inadequacies in the data base. Criteria are needed for evaluating such predictions, and such criteria should be based on an adequate technical data base, including previous experience from operating plants and analysis of aged equipment.

3. Improved predictions of long-term deterioration of sealer materials and of limits on hostile environmental exposures (1987).
   Justification: This information will help to resolve Generic Issue B-26.

4. Support of licensing dry spent fuel storage prior to ultimate disposal, with major attention to spent fuel oxidation during dry conditions and dry cask heat transfer data for BWR spent fuel rods (1989).
   Justification: The National Waste Policy Act of 1982 requires that interim spent fuel storage and monitored retrievable storage of spent fuel and high-level radioactive waste shall be licensed by the NRC.

5. Basis for determining adequacy of decommissioning plans submitted by utilities through collection of actual decommissioning data (e.g., Humboldt Bay, Shippingport, TMI) (1990).
   Justification: Decommissioning rules and regulatory guides needed to implement the rules are being developed. The NRC needs current and actual information to evaluate licensee decommissioning activities and plans.

6. Basis for establishing financial requirements for decommissioning and the establishment of guidance on steps that should be taken to facilitate decommissioning during design, operation, and actual decommissioning (1990).
   Justification: Final decommissioning rules and regulatory guides necessary for rule implementation are being developed. Information from licensees on financial assurance and facilitation is required by these rules, and the NRC needs current information to adequately evaluate licensee decommissioning activities and plans in these areas.

1.4.2 Research Program Description

The strategy for this research is to develop a technical data base to predict in a timely manner the onset of significant component aging and service wear phenomena that can adversely affect public health and safety and to develop guidelines and criteria for surveillance testing and degradation monitoring, maintenance, and replacement programs to mitigate aging and service wear effects in electrical and mechanical components important to ensure plant safety.

The initial phase of this research effort will provide input to the data base for the aging assessment of components and systems currently under way. This study will identify significant component/environment aging mechanisms that can

lead to the inoperability of vital electrical and mechanical components. The study will also yield recommendations, including priorities and schedules, for further specific research. Specifically, the approach and key activities to address aging assessment of critical electrical and mechanical components include the following:

- Selection of equipment to be studied.
  - System/aging/risk-oriented evaluations and setting of priorities for components and structures to be investigated.
  - Acquisition of pertinent knowledge of experts through workshops, questionnaires, and licensing reviews.
  - Review and analysis of existing aging data from operating experience, including LERs, reported occurrences, and evaluation of maintenance, refurbishment, and replacement programs that contend with the degradation of components and structures.
  - Review of applicable codes, standards, and guides.
- Development of technical basis for comprehensive assessment of aging of nuclear power plant components and structures.
  - Review and analysis of equipment specifications, designs, and operating parameters.
  - Postservice examination and laboratory testing of aged equipment from decommissioned (e.g., Shippingport) and operating facilities to determine aging mechanisms and aging-related failure modes.
  - Review and application of past and ongoing research on aging of materials and components.
  - Cooperative collection of data by monitoring equipment on site at one or more operating reactor facilities.
- Development of application guidelines and criteria for detection and mitigation of functional degradation of electrical and mechanical components with high-risk factors before major safety problems develop.
  - Evaluation of existing degradation monitoring techniques that would be effective in identifying functional degradation and the remaining functional capability of components.
  - Determination of practical, cost-effective indicators of functional capability.
  - Analysis of applicable codes, standards, guides, and industry practice.
  - Recommendations for advanced methods of condition monitoring.

- Risk/cost/benefit analysis for practical and cost-effective degradation monitoring techniques.

- Cooperative degradation monitoring and surveillance programs to monitor critical equipment on site at one or more operating reactor facilities to confirm feasibility of techniques.

o Addition of information data base on the effects of storing spent fuel in a dry environment.

- Verification of the phase changes that occur during the oxidation of UO_2 in air and oxygen.

- Determination of the effects of various manufacturing processes on the oxidation process.

- Evaluation of existing heat generation codes and their applicability to BWR fuel assemblies.

o Acquisition of data from decommissioning Humboldt Bay, Shippingport, and German reactors on dose rates, labor requirements, techniques, waste disposal, waste shipping, and costs, and determination of the amounts and distribution of induced radioisotopes in samples of Shippingport components.

o Acquisition of information on current techniques and costs of decommissioning nuclear facilities and on occupational doses associated therewith and information on methods that facilitate decommissioning by reducing doses and waste volumes.

The major research products will be:

1. a. Setting of priorities and selection for comprehensive aging assessment of electrical and mechanical components important to safety and susceptible to functional degradation due to aging (1986).

   b. Comprehensive aging assessment of selected plant components, including postservice examinations and failure mode analyses (including assessment of significance of aging as a factor in capability to withstand seismic and dynamic stresses) (1986-1988).

2. a. Identification of practical and cost-effective techniques for monitoring equipment or service wear effects for selected components (1987).

   b. Assessment of effectiveness of surveillance monitoring in supplementing artificial pre-aging (prior to qualification testing) for selected electrical equipment (1988) and mechanical equipment (1989).

   c. Assessment of necessity to modify surveillance/testing intervals of selected systems in plant technical specifications to account for age (1988).

   d. Assessment of methodologies for predicting service lifetime of equipment (1989).

e. Evaluation criteria for surveillance, maintenance, and replacement programs for selected components (1989).

3. Information on long-term material deterioration of sealer materials and on limits on hostile environmental exposures (1987).

4. Series of reports and upgraded data base on spent fuel oxidation during dry storage (1989); refinement of data base on heat transfer from BWR spent fuel in dry cask storage (1989).

5. Information on actual decommissionings of nuclear reactors to be used to update data base for reviewing decommissioning activities and plans (1990).

6. Current information on safety, costs, and facilitation of decommissioning, to be used in data base for reviewing funding assurance and decommissioning plans (1990).

1.4.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

As a result of budget reductions in FY 1986, the work on the effects of storing spent fuel in a dry environment and on techniques and costs of decommissioning nuclear facilities (including occupational doses associated therewith) will be unfunded.

1.5 Nondestructive Examination

This research applies to the validation of reliable, reproducible NDE techniques for detection and characterization of cracks and flaws, etc., for pressure vessels, piping, and steam generator tubing as well as the associated interpretation and analysis for decisionmaking. Assuming the current industry effort in this area will continue, the NRC research role is expected to significantly decrease in the 1986-1988 period.

1.5.1 Major Regulatory Needs and Their Justifications

1. Documentation and upgrading of the reliability and reproducibility of ultrasonic and eddy current inspection methods during preservice and inservice inspections for detection and characterization (sizing, orientation, etc.) of flaws, cracks, and other defects. The planned and completed research work formed the basis for resolving and dropping USI A-14, "Nondestructive Examination," and will provide the basis for revising regulatory guides and for recommendations for updating Section XI of the ASME Code (1987).
   Justification: Methods currently in use for flaw detection and characterization are not necessarily always consistent, reproducible, or interpretable. Nevertheless, preservice and inservice inspections are counted upon to find and characterize flaws in reactor components. For safety evaluations such as for pressurized thermal shock (PTS), it is very important to know if the very small flaws capable of crack initiation under PTS accident conditions are present or not. Thus, the methods currently in use must be quantified with respect to their reliability.

2. Criteria and validation for use of acoustic emission for leak detection and for continuous monitoring for cracking in vessels and piping, to provide the basis for licensing criteria, amendments to technical specifications, and recommendations for changes to the ASME Code (1986).
   Justification: Locations exist in plants where conventional inspection techniques are inadequate for proper examinations for flaws. Thus, alternative techniques are very useful. One such technique is acoustic emission. Here, a growing crack will produce an acoustic signal that can be monitored to produce warning, or a leak will also cause an acoustic signal that can be detected. Although such methods are desirable, no criteria exist for acceptance by NRC or for operation of the techniques in service, nor are the parameters and their appropriate useful ranges listed and justified.

3. Evaluation and validation of NDE techniques for conducting accurate and reliable "end-of-licensing-period" nondestructive baseline examinations of reactors to establish the condition of primary system components with respect to (1) any flaws or cracks initiated during prior service and (2) the material toughness and other mechanical properties after initial service (1990).
   Justification: To make licensing decisions related to extension of service for nuclear power reactors beyond their first licensing period, evaluations of the integrity of the primary system reactor components and their fitness for continued use must be performed to ensure continued safety. To perform these evaluations, the condition of the components with respect to flaws initiated during prior service and the extent of any degradation in material toughness and other mechanical properties must be accurately known. Nondestructive techniques for these assessments need to be developed and validated.

1.5.2 Research Program Description

The strategy for this research is to establish the reliability of current techniques and procedures for NDE, especially those embodied in the ASME Code, and to validate improved or advanced techniques and procedures so that better accuracy of inspection can result and so that less conservatism need be applied in licensing decisions wherein flaw size and location are issues. The NRC research program is well coordinated with the major efforts under way in the United States, especially at EPRI, and also with major overseas efforts, especially the PISC efforts of the Organization for Economic Cooperation and Development (OECD) in Paris, France.

The research approach is twofold: (1) A series of test plates and pipes are prepared with known flaws for round-robin detection and characterization trials from which conclusions can be drawn about the reliability of current and advanced NDE methods and procedures so that the currently approved code and guide procedures can be either validated or updated and (2) the basic techniques for ultrasonic test, eddy current, and acoustic emission for continuous monitoring are upgraded through development studies and proved in realistic field studies using operating reactors and components where possible.

The first approach employing round-robins is especially illuminating because it is possible to quantify the reliability of techniques currently called out in the ASME Code or guides and those employed in advanced methods. The round-robins have included piping of wrought stainless steel, centrifugally cast

stainless steel, and clad carbon steel. Thermal fatigue cracks were emplaced in all three types of pipe material, while IGSC cracks were also included in some of the wrought stainless steel pipes. Piping round-robins are continuing, especially on wrought stainless steel piping with IGSC cracks in typical but hard-to-inspect locations. Other round-robins either being conducted by NRC or in which NRC is participating include those on plate and nozzles wherein defects have been emplanted in realistic situations or wherein the defects are true manufacturing defects. This approach has already yielded a series of recommendations for changes to improve the code procedures for ultrasonic testing; it has also yielded a data base that permits a valuable revision of the guides currently approved for ultrasonic inspection.

Regarding the second approach, a key validation tool for eddy current steam generator tube inspection is the retired-from-service steam generator discussed in Section I.2. Here, accurate knowledge of the flaw sizes and types and of the extent of degradation measured in tubes after removal from the generator provides the ultimate means to evaluate and validate the inspection method by comparing the actual flaw to the in situ eddy current inspection results. Continuous monitoring to detect the onset of cracking or leakage through use of acoustic emission is validated by large-scale pressure vessel cyclic crack growth tests and by studies on components of actual operating reactors. A critical part of the program is validation of state-of-the-art ultrasonic test methods for both detection and evaluation of flaws in vessels, piping, and nozzles. Automated systems for real-time inspection and evaluation of flaws in these environments are validated following development in mockups such as at the PISC plates and the EPRI NDE Center and also in the field in operating reactors when appropriate opportunities arise. Because of the validation and accuracy achieved through these means, licensing criteria and code or guide procedures can be drawn up and used with assurance of improved reliability.

A most important use of the research results is as the basis for criteria for qualification of personnel, equipment, and procedures, especially for ultrasonic inservice inspection of piping and other primary system components. It is because of the insights gained into the effects on detection and evaluation reliability of different inspection procedures and use of equipment that such qualification criteria can be set out.

The major research products will be:

1. a. Recommendations for improvement of ASME Code, Section XI, rules for inservice inspection of nuclear power plant components, to improve the reliability of required inspections (1986).

   b. Completion of evaluations and establishment of reliability of currently practiced and advanced eddy current and other NDE methods, using retired steam generator (1986).

   c. Validation of improved SAFT-UT (synthetic aperture focusing technique for ultrasonic testing) flaw detection and evaluation method in field tests to obtain accurate flaw data for licensing decisions on piping, thick sections, welds, and multimetal joints (1986).

d. Code acceptance for the improved SAFT-UT method for flaw evaluation and detection and for continuous acoustic emission (AE) leak monitoring (1986). (Also applies to Need 2.)

e. Improved inspection plan for implementation in licensing actions for inservice inspection of steam generator tubing (1986).

f. Recommendations for ASME Code acceptance of new and improved methods for ultrasonic and eddy current inservice inspections and for continuous AE monitoring (1986-1987). (Also applies to Need 2.)

g. Code acceptance of unified set of inspection requirements for piping and vessels based on NDE flaw detection reliability, component material properties, and service conditions to ensure a suitably low failure probability (1986-1987).

h. Code acceptance of recommendations for improved inservice and continuous monitoring inspections (1986-1988). (Also applies to Need 2.)

i. Capability in ultrasonic testing, AE, eddy current, and related NDE methods for prompt reaction to inspection problems (1986-1989). (Also applies to Need 2.)

2. Code acceptance of continuous AE monitoring for cracks and validation of leak monitoring by AE for licensing use where improved monitoring is necessary or conventional methods cannot be used (1986).

3. Development and validation of NDE techniques for detection and characterization of service-produced flaws in reactors under consideration for extension-of-service licenses after initial licensing period has expired (1987-1990).

2. EQUIPMENT QUALIFICATION

This program will study the methods used for qualifying equipment used in nuclear power plants taking into account such factors as effects of synergism, order or sequence of tests, accelerated aging techniques, and methods for simulating accident environments. Methods will be validated and new methods developed as appropriate to ensure that qualification test results reported by applicants and licensees provide a basis for licensing decisions that ensure protection of the public health and safety. The elements discussed in this chapter include environmental and dynamic qualification of electrical and mechanical equipment. The research programs described below are part of an agencywide NRC effort on equipment qualification.

2.1 Qualification of Electrical Equipment for Harsh Environments

The purpose of the electrical equipment qualification research program is to study the methods for qualifying safety-related electrical equipment to demonstrate the equipment's ability to function both during and following design basis accidents that produce harsh environments, including high radiation, temperature, pressure, and humidity, and to identify ways of reducing the likelihood of undesired failure modes. Qualification methods for a loss-of-coolant accident (LOCA), main steam line break, hydrogen burn, and other design basis accident conditions will be emphasized. The research will also include a limited study of severe accident conditions beyond the design basis.

2.1.1 Major Regulatory Needs and Their Justifications

1. Evaluation criteria for environmental qualification testing of safety-related electrical equipment, to be used in developing amendments to the regulations, new regulatory guides, and revisions to existing regulatory guides (1988).†

   Justification: Electrical equipment plays a significant role in the safety-related nuclear systems that mitigate design basis accidents. Thus, the satisfactory functioning of such equipment during and following an accident is essential to the protection of the public. The electrical equipment located in containment will be exposed to harsh environments should a LOCA or main steam line break occur. Hence, the rules and regulatory guides on equipment qualification are expected to make a significant contribution to safety by ensuring that electrical equipment is adequately tested to demonstrate capability to perform during and after design basis accidents.

   A rule (§ 50.49 of 10 CFR Part 50) was issued to provide the nuclear industry with specific requirements pertaining to the environmental qualification of electrical equipment located in areas of potentially harsh environments. A method acceptable to the NRC staff for demonstrating compliance with the requirements of that rule has been described in a

†A modification of the regulations may result from the research.

revision to Regulatory Guide 1.89, "Environmental Qualification to Electric Equipment for Nuclear Power Plants." Relevant national standards such as IEEE 323 and various daughter standards are evaluated by the NRC and if determined to be suitable are endorsed through regulatory guides, generally with modifications considered acceptable to the NRC. In some cases, the IEEE standards define only in general terms the steps needed for adequate qualification and the regulatory guides must provide greater definition. This research ensures that the provisions of the regulatory guides have firm technical bases and that acceptable test procedures are defined in sufficient detail.

2. Criteria and methods for accelerated radiation and thermal aging of electrical equipment to realistically duplicate the time-related degradation at the end of its qualified life, to be the basis for revisions to regulatory guides and the standard review plan (1988).
Justification: For most electrical equipment, the margin to the failure threshold under accident conditions decreases as the equipment approaches the end of its qualified life. Thus, the qualification testing of equipment for survival and functionability during and following an accident normally requires testing the equipment in its worst time-related degraded condition. The qualified life of equipment is dependent on circumstances but some items are expected to have a 40-year life. Accelerated radiation aging at high dose rates and thermal aging at higher than normal temperatures are typically used by industry to age-test specimens in order to place them in an end-of-life condition for qualification testing. NRC research studies have found instances where realistic aging degradation is not adequately simulated by the accelerated aging procedure allowed in national standards and guides. Specifically, aging degradation to cable and gasket polymers using low normal plant dose rates has been observed to be generally more severe than using high dose rates for the same total dose. Also, more severe degradation has occurred when radiation aging precedes thermal aging than for other sequence options. These synergistic effects and the role played by oxygen diffusion in causing such synergisms will be assessed in this research.

3. Information on the behavior of polymers (elastomers), electronics, and other materials used in safety-related electrical equipment to determine their expected life and failure mode under accident conditions and thus to provide the basis for licensing review (1987).
Justification: Some materials such as polymers (elastomers) used in gaskets, seals, lubricants, cabling, etc., and integrated circuits are particularly vulnerable to degradation from nuclear radiation and the steam/water atmosphere accompanying a nuclear accident. Limited information exists on the detailed behavior of many of these materials. Research on their behavior during normal plant exposure and accident conditions is being studied in order to provide the licensing staff with data so that nuclear plant equipment and designs can be adequately evaluated to ensure the public safety. NRC research in this area is being performed in collaboration with the French Commissariat à l'Energie Atomique (CEA) and the Ontario Hydro Company in Canada. Further joint arrangements are under consideration with the Japanese Atomic Energy Research Institute (JAERI). The cost of this research is thus being minimized where possible by international cooperative agreements.

There is a need to determine whether practices currently used by industry for qualifying high-range radiation monitors that are located in containment to survive and function during and after an accident adequately account for radiation heating in the detector and degradation effects from the accident steam environment.

4. Development of the technical basis for the classification of instrumentation, control, and electrical systems and equipment (ICE S/E) important to safety, to be the basis for regulatory guidance on classification of such systems and equipment (1986).
   Justification: ICE S/E important to safety that are not Class 1E or safety related do have safety significance of varying degrees. This fact is not currently reflected by classifying these systems and equipment as either Class 1E or non-Class 1E. These systems and equipment perform operational functions important to safety (e.g., feedwater controls, reactor power control) that could possibly initiate or compound events that should also be considered design basis events. ICE S/E important to safety should be classified on the basis of importance to safety by a corresponding set of design and qualification requirements. The expected result from this effort will be a new classification designated as Class 2E for ICE S/E important to safety.

5. Experimental data base to quantify the characteristics of credible fires, the consequent environments, and the damage thresholds of components (1989).
   Justification: Past probabilistic risk assessments of nuclear power plants have shown that the contribution of fire to the overall risk is significant. There are large uncertainties in the assessment of fire risk, however, because of the inadequacy of the deterministic data base. This program addresses that need.

2.1.2 Research Program Description

Research studies are being conducted to provide criteria for evaluating the methods employed by industry for qualifying safety-related electrical equipment. The procedures for duplicating the end-of-life condition of the equipment by accelerated aging and the effects of exposures to radiation and other environmental conditions (e.g., humidity) are considered in the research programs. The question of how one simulates the radiation, hydrogen burn, and LOCA steam exposure to the equipment during and following the accident is also addressed in this research.

Evaluation criteria are to be developed on how to simulate the mechanism of equipment damage from beta and gamma fission products by the use of a cobalt gamma simulator. A new program in cooperation with the French CEA is to be initiated to develop a gamma-damage-equivalent model for beta radiation. Almost all qualification testing being done today uses a cobalt gamma source for radiation exposures.

The source term fission product from an accident release used in current regulatory guides on qualification has been based on release models developed for determining the site exclusion boundary. The results of the ongoing NRC

and IDCOR (Industry Degraded Core) program source term research efforts and results from the Department of Energy (DOE) evaluation of TMI-2 radiation and fission product distribution measurements are to be used to develop improved models for determining accident radiation levels and doses to equipment in containment. The results of the above research will provide criteria for validating or revising, as necessary, requirements in § 50.49 of 10 CFR Part 50 and Regulatory Guide 1.89.

The Institute of Electrical and Electronics Engineers (IEEE) prepares industry standards that list general procedures for qualifying Class 1E electrical equipment (IEEE 323) and methods for qualifying specific items of Class IE equipment such as electrical penetration assemblies (IEEE 317), electrical cables and splices (IEEE 383), electric valve operators (IEEE 382), and lead storage batteries (IEEE 535). The procedures given in these standards for accelerated aging and accident simulation are being examined in qualification research tests of electrical components such as batteries and cables. Those Class 1E components potentially subject to common mode failure in an accident or for which specific licensing concerns exist with the qualification procedures will be selectively used to test the validity of the methods in the IEEE standards.

The results of qualification research tests will be compared and correlated to actual plant experience by examining and testing aged equipment removed from nuclear power plants. For example, the DOE-sponsored examination of equipment removed from TMI-2 has already identified a number of failure modes in radiation monitoring equipment required to follow the course of an accident. Safety-related electrical equipment located in the containment of nuclear power plants will be examined when it becomes available.

Significant questions have arisen regarding acceptable methods for the artificial accelerated aging of equipment called for in the standards as part of the qualification testing sequence. Dose rate effects in aging are being investigated by exposing a large number of polymers used in electrical equipment to different levels of radiation. Synergistic effects between thermal and radiation aging are also being studied in these tests. In some polymers it has been possible to correlate the dose rate effects with oxygen diffusion rates. However, interpreting the degradation of the many types of polymers (e.g., PE, PVC, CLPO, EPDM, TEFZEL) used in electrical equipment is complicated by variances in the competition among breaking bonds, oxidation, and crosslinking during radiation. Accelerated thermal aging using the Arrhenius rule is frequently employed. This is valid only where a single rate mechanism dominates. For example, failure modes that do not duplicate or match failures to be expected in a harsh environment have been observed with electronic components such as transmitters during LOCA simulation tests when Arrhenius aging is used. Research on radiation damage simulation and thermal aging is continuing in order to develop a mechanistic understanding of dose rate effects, reaction rates (Arrhenius), etc.

The behavior of materials used in electrical equipment is being studied in the research tests described above on qualification methods for simulating aging and accidents. Recent qualification simulation tests of polymers in which the LOCA simulation used pure steam versus steam plus a partial pressure of oxygen

simulating the containment air (oxygen) present prior to the accident have shown that the presence of air (oxygen) significantly increases the degradation of some polymers. Further research on this question is to be pursued under a cooperative agreement with JAERI. The behavior of integrated electrical circuits to radiation and possible synergistic effects from combined aging, radiation, and humidity (which contributed to the failure of radiation monitoring equipment at TMI-2) are to be evaluated in order to provide a basis for the development of requirements for the qualification of postaccident monitoring and control equipment located in containment that is required to function during an accident. The degradation of gasket and seal materials (which play a vital role in ensuring the containment integrity of personnel penetrations, freight doors, vent valves, etc.) under radiation and accident conditions is another area of investigation.

Valves, switches, transmitters, etc., are required to function over long periods of time following an accident. Research tests are planned to quantify the degradation of electrical equipment during this period and to develop an acceptable method for accelerating the postaccident aging period in equipment qualification tests.

Laboratory research to resolve the significant safety questions currently identified with equipment qualification is to be completed in 1987. This research will provide the criteria and basis for validating, developing, or revising, as necessary, the regulatory requirements and guides for electrical equipment qualification.

I&C and electrical components will be evaluated for their reliability and performance capability in performing their function(s) under the conditions expected to be encountered when they are needed. Specifically, their ability to perform their required function in terms of accuracy and response time over the required range will be assessed for all service conditions, including normal, postulated off-normal, design basis accident, and severe accident conditions.

Results from these programs will be factored into the standards activities to ensure that the standards and regulatory guides focus on the important component characteristics. Where appropriate, results of this research can be factored directly into the licensing and inspection and enforcement programs in the form of equipment qualification acceptance criteria, IE bulletins, or guidance to licensees.

The major research products will be:

1. a. Reassessment of the basis for determining equipment radiation levels in accidents and the adequacy of simulations (1987). (Also applies to Need 3.)

   b. Acceptable methods for aging and accident simulation qualification tests of electric penetrations and electric motors (1986).

   c. Validation of aging and accident simulation methods by correlating artificial aging research results from examining and testing equipment from operating plants and accident exposure at TMI-2 (1987).

2. a. Valid accelerated aging procedure for lead storage batteries (1986).

   b. Criteria and basis for regulatory requirements for aging in equipment qualification (1987).

3. a. Evaluation of importance of beta radiation and the gamma damage equivalence in accident simulation (1987). (Also applies to Need 1.)

   b. Assessment of degradation of gasket and seal materials and lubricants in the radiation and environment accompanying a LOCA or severe accident (1986).

   c. Evaluation of vulnerability of electronic equipment with integrated circuits to accident conditions and development of qualification requirements (1986).

   d. Assessment of qualification methods used with high-range radiation monitors (1986).

   e. Acceptable method for accelerating the postaccident aging period during equipment qualification tests (1987).

4. Technical basis for regulatory guidance for instrumentation, control, and electrical power systems important to safety but not safety related (1986).

5. a. Compilation of data on fire damage thresholds of safety equipment (1986).

   b. Development of a fire environment model and computer code (1986).

   c. Results of a set of benchmark tests yielding fire environments resulting from credible fires (1986).

   d. Evaluation of margins of safety in plant enclosures, including the control room, with specific equipment configurations (1986).

   e. Evaluation of margins of safety in typical containments (1987).

   f. Data base for probabilistic fire risk assessment (1987-1989).

2.2 Qualification of Mechanical Equipment (Environmental)

This research will provide the technical basis for developing requirements for environmental qualification of mechanical components. Environmental parameters include temperature, pressure, humidity, radiation, chemicals, and submergence. They do not include consideration of dynamic loads whether these originate from outside the equipment (e.g., seismic or other transmitted vibration) or from inside the equipment (e.g., dynamic effects from process flow). These loads are addressed in Section 2.3.

2.2.1 Major Regulatory Needs and Their Justifications

1. Determination of the environmental parameters affecting the ability of the equipment that is required to perform a safety function during and following design basis events, to be the basis for licensing decisions and for assessing qualification programs submitted by applicants and licensees (1987).
   Justification: Since mechanical equipment will be subjected to many different environmental parameters, it is necessary to determine which environments may affect the safety function of the equipment.

2. Evaluation of proposed methods of qualifying equipment for design basis events at new and operating plants and those under construction, to be the basis for licensing decisions and the development of regulations or regulatory guides that endorse national consensus standards (1987).†
   Justification: Evaluation of equipment qualification methods is needed so the staff can assess vendor and utility submittals. Currently, standardized qualification methods do not exist for mechanical equipment. Thus, independent, unbiased evaluation of numerous methods must be performed by the NRC.

2.2.2 Research Program Description

Those environmental parameters that are significant in affecting the equipment's functional capability will be studied to determine if the assumptions currently used concerning the environmental loads are correct and to determine if there are any synergistic effects when those loads are combined.

This effort will then be combined with the results of studies, e.g., probabilistic risk assessment, to determine which components, based on potential reduction in risk to the public, should be subjected to codified environmental qualification requirements. It is anticipated that techniques such as probabilistic risk assessment will mature to the stage that they will be accurate to the major component level and will characterize not only the normal operational environmental loads but also the accident environmental loads.

The environmental effects are of concern only for limited subcomponents of mechanical equipment such as seals, gaskets, and packing. The technical bases for evaluating the environmental effects on mechanical equipment will come from the program described in Section 1.4, "Electrical and Mechanical Components."

The major research products will be:

1. Identification of failure modes in pumps and valves for subcomponents such as gaskets, seals, and packings using existing experience data (1986).

2. Data base for modifying or developing standard, to include methodology for qualifying main coolant pump shaft seals (1986).

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†A modification of the regulations may result from the research.

2.3 Dynamic Qualification of Equipment

This research will provide the technical basis for developing the qualification requirements involving dynamic loads whether they originate outside the equipment (e.g., seismic or other transmitted vibration) or from inside the equipment (e.g., dynamic effects from process flow) for electrical and mechanical equipment. It includes environmental loads to the extent that they may be combined with the dynamic loads. Also included is research on extrapolation, characterization of loads, load sequencing, load combinations, margins, uncertainties, and qualification by testing and analysis.

2.3.1 Major Regulatory Needs and Their Justifications

1. Determination and characterization of those loads affecting the ability of the equipment that is required to perform a safety function during and following design basis events, to be the basis for licensing decisions and for assessing qualification programs submitted by applicants and licensees (1987).
   Justification: Since mechanical and electrical equipment will be subjected to many different loads during the life of a plant, it is necessary to determine what characteristics of the loads may affect the safety function of the equipment and the uncertainty of the magnitude or level of the loads to be simulated during qualification. The research will also identify areas in which the uncertainties in defining the dynamic parameters may be beneficially reduced.

2. Establishment of data on equipment responses, failure modes, and margins, to be the basis for ensuring the seismic capability of mechanical and electrical equipment in new and operating plants (1987).
   Justification: Establishment of such data base is needed for assisting the staff in judging structural integrity and functional operability of equipment claimed to have been seismically qualified by applicants and licensees. Such information will also be useful in probabilistic risk assessment for nuclear power plant design as well as the implementation of USI A-46, "Seismic Qualification of Equipment in Operating Plants."

3. Determination of safety margins that are available in the existing mechanical and electrical equipment design against the Safe Shutdown Earthquake (SSE) for new and operating plants (1987).
   Justification: Determination of the safety margins is needed for assessing equipment capability under the effects of earthquakes with magnitudes greater than the design basis earthquake.

4. Evaluation of proposed methods of qualifying equipment for design basis events in new and operating plants and those under construction, to be the basis for licensing decisions and the development of regulations or regulatory guides that endorse national standards (1987).†
   Justification: Evaluation of methods of mechanical and electrical equipment qualification for dynamic loads are needed for the staff to assess vendor and utility submittals. Currently, standardized qualification methods do not exist for mechanical equipment. Thus, independent, unbiased evaluation of various methods must be performed by the NRC.

†A modification of the regulations may result from the research.

5. Establish criteria for determining what qualification methods are acceptable for new and operating plants and those under construction, to be the basis for a regulatory guide (1987).
   Justification: Prudent acceptance criteria for mechanical and electrical equipment qualification methods for dynamic loads must be developed. These criteria must account for uncertainties in definition of the dynamic loads and the qualification methods, yet must provide a measure of reduction in risk over mechanical and electrical equipment qualified to other less appropriate criteria.

2.3.2 Research Program Description

This research program will provide the technical basis to evaluate the qualification procedures for mechanical and electrical equipment subjected to dynamic loads. This program (which is to end in FY 1987) consists of experimental and analytical efforts that will provide data and information for determining and establishing margins for components such as valves, pumps, motors, and electrical controls when subjected to postulated accidents. The loading characteristics, modes, and levels of failure (if any) for these components will be identified and will aid in providing criteria for evaluating equipment in operating plants, plants currently being licensed, and new plants.

Included with the above effort is that related to determining the characteristics and the effects of long-term system loads on the operability of pumps and valves. Acceptable methods for accounting for these loads in the qualification process will be developed.

The important considerations regarding the dynamic qualification of equipment such as valves and electrical controls by test and/or by analysis will be identified. Specifically, equipment categories will be established and qualification criteria for each of these categories will be developed. The question as to when a component can be qualified by analysis rather than by test will be answered. In addition, modeling techniques and methods for simulating the effects of component nonlinearities in analytical models will be developed. Methods for identifying and determining the flow-induced force characteristics will also be developed.

Areas of importance for dynamically qualifying categories of equipment by test rather than by analysis include accounting for the influence of system interactions and the simulation of component foundations, including methods for ensuring input due to cross-coupling.

Experimental studies sponsored by other NRC research programs as well as foreign and domestic agencies will be used to the maximum extent possible. Cooperative agreements with EPRI in the area of equipment qualification are being explored. The use of the HDR facility in the Federal Republic of Germany may be another possibility. High-level vibratory tests at the HDR facility may include evaluation of active equipment under simulated seismic conditions. Negotiations with the HDR project are under way.

During the experimental parts of the program, where possible, information pertaining to containment leak integrity will be provided to the severe accident research program.

Interaction with various committees that develop component qualification standards will be undertaken when appropriate milestones in this program are achieved.

The major research products will be:

1. Criteria and guidelines for qualification by test and/or analysis for valves (e.g., containment boundary valves), charging pumps, motors, and electrical controls (1987).

2. Establishment of a data base of equipment responses and failure modes to determine safety margins available in equipment such as valves, pumps, motors, and electrical controls (1987).

3. Evaluation of safety margins available in mechanical and electrical equipment such as valves, pumps, motors, and electrical controls subjected to the SSE (1987).

4. Development and evaluation of methods that are acceptable for dynamic qualification of mechanical and electrical equipment such as valves, pumps, motors, and electrical controls (1987).

5. Development and evaluation of criteria for determining acceptable methods for qualifying mechanical and electrical equipment such as valves, pumps, motors, and electrical controls (1987).

3. SEISMIC RESEARCH

Earthquakes are among the most severe of the natural hazards faced by nuclear power plants. The bases for NRC's seismic research needs are three issues dealing with seismic hazard, seismic risk, and seismic margin.

Since the U.S. Geological Survey clarified its position on the Charleston earthquake, the seismic hazard of nuclear power plants on the eastern seaboard of the United States has become a major regulatory issue. The question is whether an earthquake of the magnitude of the Charleston earthquake could occur elsewhere on the eastern seaboard, placing a nuclear power plant at hazard.

Another issue is the public risk associated with seismic events in excess of the current design basis, the safe shutdown earthquake (SSE). Should this risk turn out to be large, the technical bases for potential changes in the regulations would be needed.

The third seismic issue is the margin inherent in the seismic design of older plants. Frequently, the NRC is faced with decisions relating to the seismic design of operating plants. New information available on seismicity, soil-structure interaction (SSI), and plant seismic response can result in changes in the predicted loads on structures, systems, and equipment. These changes must be compared to the inherent margin of the original design before a decision can be made.

3.1 Seismic Hazard

3.1.1 Major Regulatory Needs and Their Justifications

1. Data concerning seismic source zones in the Eastern United States, including Charleston, New Madrid, New England, and others, and data concerning seismic energy propagation in the East needed for the analysis of seismotectonic provinces as required by Appendix A to 10 CFR Part 100, to be the basis for amendments to Appendix A to 10 CFR Part 100 and for developing regulatory guidance (1988).†
   Justification: Except for the New Madrid seismicity, the distribution of seismicity in the East, including New England and the vicinity of Charleston, S.C., is not well defined. No working hypothesis for the cause of the seismicity is generally accepted by the geoscience community. The stress fields that drive the major faults are poorly known.

2. An information base for the development of site-specific response spectra, to be the basis for amendments to Appendix A to 10 CFR Part 100 and for developing regulatory guidance (1988).†
   Justification: The recent earthquakes in New Brunswick, New Hampshire, and Arkansas have generated important strong-motion records that for the first time provide a significant opportunity to compare real data with theoretical ground motion and attenuation models for the Eastern United

†A modification of the regulations may result from the research.

States. Analysis of these records will address important regulatory questions concerning the interpretation of this type of record and its use in licensing decisions.

3. Methods for handling the uncertainties in assessing the potential risk from seismic hazards, including such topics as the applicability of using the historic method of seismic hazard analysis, the impact of the 1982 New Brunswick earthquake sequence, and the verification of Holocene movement on the Meers fault, to be used to revise current siting regulations (1989).†
   Justification: The current seismic siting regulations do not provide guidance on how to handle the uncertainty associated with the licensing decisions and judgments being made at the forefront of a rapidly developing science, seismology. Decisions must be made based on the best available data that are steadily being updated. This program is designed to develop statistical or probabilistic tools to aid the decisionmaking process.

4. Improved data base and analysis techniques for predicting soil failure, including soil liquefaction, particularly for small or moderate changes in the acceleration value at which the design spectrum is anchored, to be used for regulatory guidance (1987).
   Justification: The recent high acceleration records from New Brunswick, New Hampshire, and Arkansas and the potential for large "anchor-point" accelerations because of the Charleston earthquake issue may reduce the safety margins associated with current soil-failure-prediction techniques.

5. Verification of methods for predicting seismic soil settlements (1987-1988).
   Justification: The consequences of soil settlement, liquefaction, or the soil failures at high earthquake levels (above SSE) could be a significant contributor to overall risk. Current probabilistic risk assessments (PRAs) do not adequately address this problem, and at the present time there are no verified methods for estimating seismic soil settlement.

3.1.2 Research Program Description

Uncertainty in seismic hazard analysis is a fundamental issue. The strategy to reduce this uncertainty involves three programs: (1) development of a better seismic zonation through studies of the causes of earthquakes in the Eastern United States, (2) determination of more accurate seismic wave attenuation relationships, and (3) development of better data and models of site-specific spectral response.

The program to establish a better seismic zonation in the Eastern United States is directed at determining the cause of the seismicity in the East. This program consists of monitoring the seismicity in the East through a series of seismographic networks, crustal structure determinations in critical areas (such as Charleston, S.C., Moodus, Conn., and the Ramapo fault area, N.Y. and N.J.), crustal stress measurements, studies of recent crustal movement, and detailed evaluation of features such as the Meers fault in Oklahoma. Involved are significant interaction and cooperation with other Federal and State agencies such as the U.S. Geological Survey and State geological surveys and the use of utility and EPRI data sets where available. Other programs involve

†A modification of the regulations may result from the research.

the more precise placement of seismic instrumentation and the analysis of data to establish seismic wave attenuation relationships and to limit uncertainties in ground motion.

A network of strong-motion seismographs has been established to gain information on site-specific response and attenuation relationships. A cooperative agreement is being pursued with EPRI for access to the strong-motion data being collected on Formosa at the EPRI test facility. A representative piping system has been constructed on an instrumented test platform in the northern part of Formosa, an area frequently subjected to strong seismic ground motion. Data analyses and theoretical studies of strong motion are being conducted for the Eastern United States in cooperation with the U.S. Geological Survey engineering seismology group.

The Army Corps of Engineers has studied soil-failure codes and concluded that the DESRA code is the best candidate for validation. Validation experiments will be conducted at Cambridge University.

A cooperative workshop on soil liquefaction, supported by the NRC, the National Academy of Sciences/National Research Council, and the National Science Foundation, is being planned.

The major research products will be:

1. Seismographic network data used on a day-to-day basis by licensing staff, by staff involved in PRAs, by the seismic hazard characterization project, and for rulemaking decisions and engineering research projects (1986-1988); geophysical data for determining crustal structure in areas of suspicious geologic structures (1986-1987); and data from the in situ stress measurement program in the Northeastern United States (1986).

2. Techniques for calculating site-specific response spectra (1988).

3. a. Probabilistic sensitivity study of probable ground-motion dependence on the various proposed causes of seismicity in the Eastern United States (1986).

   b. Preliminary results of field investigations of the Meers fault in Oklahoma (1986).

4. Validation of the DESRA code and recommendations regarding the use and criteria for the use and importance of SSI analytical methods for layered soil, high water tables, and liftoff effects (1987).

5. Determination of seismic risk contribution from dam and embankment failure (1988).

3.1.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

As a result of budgetary reductions for FY 1986, a program to determine the contributions of dam and embankment failures to seismic risk has been deferred. Probabilistic techniques to evaluate the effects of earthquakes equal to or larger than the SSE are needed for reviews of the geotechnical

features of Seismic Category 1 dam and embankment structures to ensure that they can perform their functions during the life of the nuclear plant.

3.2 Seismic Risk

3.2.1 Major Regulatory Needs and Their Justifications

1. Complete development of seismic PRA methods (1986).
   Justification: The seismic risk methodology developed in the Seismic Safety Margins Research Program (SSMRP) was demonstrated by application to the Zion Unit 1, a PWR plant. Thus, all the systems analysis models (initiating events, event trees, and fault trees) and all the structural and piping models were developed for a PWR plant even though the methodology developed in the SSMRP should be equally applied to both PWRs and BWRs. A seismic risk analysis of a BWR is being performed to verify the applicability of the SSMRP methodology to both PWRs and BWRs.

2. Validation of current seismic PRA methods (1986-1989).
   Justification: The SSMRP and industry methodologies for estimating seismic risk are fairly new and have not been subjected to experimental validation. The seismic hazard and fragility data rely heavily on expert opinion, and critical system modeling assumptions are made in the risk analyses. Validation will increase our confidence in seismic PRA methods and their effectiveness so that they may be used in the regulatory decisionmaking process.

3. Development of a methodology for flood probability estimates based on stochastic relationships between deterministic data and flood probabilities (1988-1989).
   Justification: Deterministic methods of evaluating severe hydrological and hydrometeorological events (i.e., floods, droughts) do not support risk or safety margin assessments. Methods for making reasonable and defensible flood probability estimates and developing flood hazard curves are needed by the NRC staff to replace interim procedures using default values.

4. Investigation of the levels of uncertainty in current tornado wind speed estimates based on calibrations of structural damage to inferred wind speeds (1987).
   Justification: A determination of the error bands and quantification of uncertainty estimates in the tornado data base used by the NRR staff is needed to reduce uncertainties in risk evaluations, including those for operating plants, concerning hazards due to high winds.

5. Evaluation of real-time atmospheric dispersion and plume rise models for dose projections during emergencies, to determine model accuracy and reliability, uncertainties, and data requirements under various terrain and meteorological conditions (1987-1988).
   Justification: Information concerning the accuracy and reliability of real-time dispersion models used for dose projections within the emergency planning zone, including uncertainties and data requirements, under a variety of meteorological and terrain conditions is needed by the IE and

NRR staffs for reviews of licensee/applicant emergency response organization and accident assessment capabilities and for evaluation of models selected to be installed at the NRC Operations Center.

6. Confirmation of washout coefficients for released, water-soluble particulate and gaseous fission products that determine the fraction of the effluent plume that is scavenged and brought down to the ground as a result of natural rainfall occurring along the plume pathway during the course of an accident (1987-1989).
Justification: A confirmation of the values for the washout coefficients for released, water-soluble gaseous and particulate fission products due to rainfall, which are currently controversial, is needed by the NRC staff to reduce uncertainty ranges in probabilistic risk analyses and for emergency response applications.

3.2.2 Research Program Description

The development of a simplified BWR seismic risk methodology by adding BWR-specific features to the simplified SSMRP methodology, which was used for the Zion Unit 1 PWR, will be completed in early 1986. This simplified PWR/BWR methodology will be used in the Risk Methodology Integration and Evaluation Program (RMIEP), a program incorporating and applying state-of-the-art techniques to assess risk from events initiated both internally and externally.

In the past, a general lack of instrumentation has prevented the correlation of earthquake damage to specific input spectra. At present, a program instituted by the U.S. Bureau of Reclamation associated with the California Department of Water Resources has caused a significant increase in the placement of seismic instrumentation. This should prove useful for this validation program. As an example of instrumentation that was strategically located to provide useful information, the Pleasant Valley station did so during a May 2, 1983 earthquake at Coalinga, California.

The techniques and data used for predicting the seismic hazard at sites will be validated through the seismotectonic program (Section 3.1.2). Products from this program will also help reduce the large uncertainties now associated with predicting the occurrence of earthquake levels of interest in seismic PRAs.

The SSMRP has been the most comprehensive effort to date in the seismic risk field. The products of this program will serve to benchmark other PRAs that consider seismically induced accidents. While a number of comparisons with other analytical methods have been performed under the program, an increased effort to independently validate the SSMRP methodology using experimental and observed earthquake data is being pursued.

In 1984, an expert panel on seismic risk validation formulated recommendations for research needed to verify or improve methods of structural response and failure prediction that are in use in current seismic risk studies. The expert panel on quantification of seismic margins has also identified seismic PRA validation needs. In 1985, a program plan for seismic risk validation will be completed. This plan will coordinate ongoing research and will develop new structural research to meet the needs recognized by these panels. Two important elements of this plan were started in 1984 with the establishment of NRC cooperation in the high-level HDR shake tests in West Germany and in EPRI's containment model experiments in Taiwan.

The RMIEP will validate the system analysis aspects of the SSMRP methodology. The SEISIM code will be evaluated as well as the SSMRP's procedures for fault tree culling, uncertainty analysis, and importance ranking.

The uncertainties identified in the existing SSI analytical methods are currently being resolved by the Standard Problems for Structural Computer Codes at Brookhaven National Laboratory. For this purpose, available experimental and recorded data have been collected and evaluated and will be analyzed to establish benchmarks for standard SSI problems. It appears that there is not a sufficiently broad data base to answer all the uncertainties. Thus, analytical studies will be made to test the significance of some of the uncertainties. It is planned to conduct such studies relating to the effect of layering, water table location, and nonlinear effects (liftoff and slip) on the interaction problem.

A feasibility study on the adaption of a Canadian Atomic Energy Control Board study and other existing methodologies of comparing presently used deterministic techniques to stochastic techniques for estimation of design basis floods for nuclear power plant siting will be performed.

Also presently used NRC staff methods relating probable maximum events (i.e., probable maximum flood (PMF), probable maximum hurricane (PMH), etc.) to probabilistic-based events and estimating the degree of error in the developed relationship will be evaluated.

Based on the results of the feasibility study, a second study will be performed to review and assess the problem of determining severe event probabilities up to the probable maximum event (i.e., PMF, PMH, etc.) and the various methodologies proposed in the feasibility study and in the technical literature.

The merits and deficiencies of the proposed methodologies will be evaluated.

Also, the current NRC staff methods for determining the probabilistic relationship to deterministic-based severe flood events against the proposed methods will be assessed and alternative methods or modification of existing methods will be recommended.

The levels of uncertainty in current tornado wind speed estimates based on calibrations of structural damage to inferred wind speeds will be investigated. The feasibility of quantifying uncertainties in tornado hazard curves through analysis of known sources of error will be determined.

Dispersion models for emergency response applications will be evaluated by comparing their projections of plume location, shape, arrival time, and concentrations with data from field tests to determine the validity and accuracy of each model under various terrain and meteorological situations. The most accurate and reliable models will be identified along with the meteorological data input that best ensures the accuracy of the model projections. Existing methodologies for characterizing plume rise will be evaluated to determine which are most applicable to emergency situations at nuclear facilities, including fires. Investigations to confirm washout coefficients for fission product plumes under natural rainfall occurrences during the course of an accident will build on information obtained from previous wind tunnel experiments and will use data obtained from small-scale field tests.

The major research products will be:

1. a. A summary report describing and evaluating the seismic risk methodology as it relates to the Zion Unit 1 PWR (1986).

   b. A summary report describing and evaluating the seismic risk methodology as it relates to the LaSalle County Station Unit 2 BWR (1986).

2. a. Nonlinear response data from HDR shake tests (1986-1987).

   b. RMIEP's evaluation of the SSMRP's system analysis methods (1986).

   c. Earthquake response data from EPRI's scale-model containment in Taiwan (1986-1990).

3. A report describing the technical base to be used to develop guidance for an acceptable methodology for making flood probability estimates and for developing flood hazard curves for use in risk assessments and safety margin evaluations (1988).

4. A report defining uncertainties and resulting error bands associated with tornado wind speed estimates inferred from structural damage surveys and quantifying uncertainties in tornado hazard curves (1988).

5. Based on comparisons of model predictions with results from field tests, information concerning the suitability and reliability of various atmospheric dispersion models for real-time emergency response applications related to plume characterization and dose projection within the emergency planning zone (1989).

6. Final report providing information concerning appropriate values for fission product plume washout coefficients under rainfall or other natural precipitation events (1990).

3.2.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

As a result of reductions in the FY 1986 budget, the following programs have been deferred: development of a methodology for flood probability estimates, investigation of the levels of uncertainty in current tornado wind speed estimates, validation of real-time dispersion and plume rise models for emergency dose projections during accidents, and confirmation of washout coefficients for airborne, dry, radionuclide plumes that are impacted by rainfall.

3.3 Seismic Margins

3.3.1 Major Regulatory Needs and Their Justifications

1. Evaluate current seismic design criteria and the effects of uncertainties to assist the licensing staff in their assessment of near-term operating licenses and systematic evaluation program (SEP) plants (1986-1990). Justification: Current regulatory requirements are based on conservative assumption and expert opinion and lack appropriate test or earthquake experience data to quantify the degree of conservatism. Knowing and

understanding the level of conservatism in the current seismic design criteria will enable the staff to judge the necessity of modifying and requalifying structures and components in older plants or improving design criteria for new plants.

2. Data to predict the nonlinear response and failure modes of nuclear shear wall structures (1986-1990).
Justification: Recent seismic PRAs indicate that structural failures leading to failure of the equipment housed or supported are dominant risk contributors. Our lack of appropriate test or earthquake experience data has resulted in large uncertainties associated with structural fragilities. It is also necessary to more clearly understand the nonlinear response of shear wall structures so that the high seismic load input to equipment can be better defined.

3. An improved seismic fragility data base for mechanical and electrical equipment (1987).
Justification: The fragility data used in current PRAs and in future seismic margin evaluations rely heavily on expert opinion and thus are subject to large modeling uncertainties. A better understanding of equipment failure modes and levels and the parameters that control them is needed as a basis for equipment qualification decisionmaking and for standards leading to more balanced plant design.

4. Assessment of the margins inherent in the seismic design of structures, piping, and equipment in nuclear power plants and quantification of seismic design margins using simplified approaches in terms of risks associated with seismic events beyond the design basis (1986).
Justification: Potential changes in plant seismic resistance from new information on seismicity, soil-structure interaction, plant response, and fragilities must be assessed and compared with the inherent margin of the original design. The seismic design margins in older plants are of particular interest since these generally had less conservative design criteria for their original design.

3.3.2 Research Program Description

The Seismic Category I structures program will provide analytical and experimental data assessing how the parameters used in the design of safety-related equipment and noncontainment structures are affected by earthquake loads above the initial design level. These parameters include, but are not limited to, floor response spectra, structural frequencies, accelerations, and displacements. Sensitivities of these design parameters to changes in internal and external wall configurations, design practices, and magnitude and duration of seismic input motion will be determined as the model configurations are subjected to quasi-static and seismic tests of increasing magnitude to cause elastic and inelastic responses. Failure modes and levels will be determined for all configurations tested.

The structural loads combination program will provide the rationale for improving the design requirements for all loadings, including seismic loads. Probabilistic analyses will provide recommendations on how normal and accident loads are to be best combined on an equal level probability without introducing unnecessary conservatisms.

In 1984, the NRC Piping Review Committee made several recommendations for research needed to support future piping design criteria changes. The NRC piping research program will address these needs through new research tasks conducted both individually and by cooperation with EPRI's Piping and Pipe Fitting Reliability Program. This latter program will provide valuable non-linear response and failure data needed to determine realistic piping system margins and failure modes. The objective of this research is to remove unnecessary conservatisms in piping design (leading to too many supports) and to increase overall piping reliability.

Information on seismic fragilities of mechanical and electrical components will be obtained by testing and by assembling, analyzing, and interpreting existing component fragility data. This information will be used to improve seismic PRAs and to obtain better estimates of seismic margins by using more realistic test-based component fragilities. The impact of scaling, aging, anchorages, and other parameters such as vintage, manufacturer, and construction practices will be evaluated.

The Seismic Design Margins Program (SDMP) provides the technical basis for assessing the significance of design margins in terms of overall plant safety and will identify potential weaknesses that might have to be addressed. This, in conjunction with past studies and ongoing validation and fragility efforts, should be effective in resolving the quantification of seismic design margin issues.

A general definition of seismic design margin is expressed in terms of how much larger than the design basis earthquake an earthquake must be to compromise plant safety. In this context, margin needs to be defined at the plant, system/function, structure, and component level.

The objective of SDMP is to develop the technical basis to resolve regulatory needs relating to seismic design margin. The several steps for attainment of this objective are accomplished through a multiphase plan. The first phase, which involves assessing existing information, estimating existing margins, identifying generic attributes, assessing margin adequacy, and developing screening guidelines and methods for their application, will be completed in 1985.

The major research products will be:

1. Experimental validation of assumptions and analytical methods used to predict the fundamental natural frequency, accelerations, and displacements of structures (1986-1988).

2. a. Determination of the changes in floor response spectra, structural frequency, acceleration, displacement, and damping for input motion resulting in elastic and inelastic behavior (1986-1988).

   b. Failure modes and failure level data for various concrete configurations (1986-1988).

   c. Recommendations for structural load combination criteria (1987).

3. a. Fragility data from component testing (1986-1987).

   b. Recommendations for damping values to be used in piping analysis (1986).

   c. Risk comparisons among piping systems designed to various proposed new criteria (1986-1987).

4. a. Using screening guidelines and recommendations developed in 1985, review plants individually or on some sort of selective group basis (1986).

   b. Refine screening guidelines and recommendations based on initial plant reviews, and conduct additional reviews as necessary (1986-1987).

4. REACTOR OPERATIONS AND RISK

This chapter describes the research being carried out to support the development of probabilistic risk assessment (PRA) methods and their use within the regulatory structure to identify those elements of reactor operations that are the most significant contributors to risk, the causal factors associated with them, and to permit comparative evaluations of risk levels associated with various regulatory actions. Past efforts in this area have identified the man-machine interactions as an area of significant uncertainty and therefore a potentially large contributor to risk. This work includes the development and trial use of models, methods, procedures, and other analyses required to support Commission decisions on a broad range of critical issues relating to power reactor safety and the acquisition of data to support the application of PRA methods to the regulatory process.

This program is divided into three elements corresponding to the principal topical areas within this program, i.e., reliability and risk methodology, data base development and evaluation, and regulatory and inspection applications. These elements are closely tied to the severe accident risk efforts discussed in Sections 6.1, 6.2, 6.10, and 6.11.

4.1 Reliability and Risk Methodology

Research is directed toward developing, testing, documenting, and, to the extent possible, validating methods for estimating the probabilities and consequences of reactor accidents and toward identifying the major contributors to the uncertainties and risk in such estimates. This information serves as one basis for evaluating the need for regulatory action and provides methods for analyzing the viability of proposed resolutions.

4.1.1 Major Regulatory Needs and Their Justification

1. Methods for the systematic identification and evaluation of principal reactor accident sequences and their precursors, to support decisions regarding severe accidents (see Chapter 6, "Severe Accidents") and to provide a consistent basis for reliability assurance and emergency response, both in-plant and ex-plant (1986).

2. Techniques for incorporating the contribution of common-cause failures, including fire and systems interactions, into PRA methods to support the proposed Integrated Safety Assessment Program (ISAP) and other PRA-related regulatory issues and to complement and extend PRA procedure guidance provided in NUREG/CR-2815 (1986-1990).

3. Methods for quantifying the effects of severe natural phenomena (e.g., seismic activity (1986-1987) and external floods (1986-1988)), including screening procedures, effects of secondary failures, impact of mitigating systems, consideration of recovery actions, and evaluation of potential multiple initiators triggered by the same external event.

4. Methods for acquiring reliable human error data for use in making quantitative and qualitative assessments of nuclear power plant operator and maintenance personnel unreliability as part of the PRA process (1986-1989).

5. Procedures and models for adequately assessing the impact of human error on overall plant risk and for providing insights into the causes of man-man and man-machine errors and into their elimination (1986-1990).

Justification of Above Needs: Quantitative assessments of the probabilities and consequences of severe reactor accidents are becoming an increasingly important element in the regulatory decisionmaking process. The Commission's Policy Statement on Safety Goals includes quantitative risk-based design objectives. Similarly, policies presently being considered by the Commission (SECY-84-370, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations") will require that future Commission decisions regarding the imposition of additional safety requirements on operating plants be justified on the basis of the best available evidence (including probabilistic analyses) of the safety of reactors and the potential for risk reduction as well as other considerations. In other areas, PRA techniques and models being developed in this program will become increasingly important tools for assessing the safety significance (and priorities) of many important issues facing the Commission (e.g., the unresolved safety issues), for evaluating new potential safety issues, and for evaluating and reviewing industry-sponsored PRA applications.

Although methods to assess the likelihood and consequences of reactor accidents are reasonably well developed, a number of sources of uncertainty remain that limit their usefulness. For example, a substantial amount of research needs to be performed so that common-cause failures and human errors can be more comprehensively included in risk analyses. Research is also planned to strengthen our capability to assess (i.e., to quantify with known error bands) the risk associated with extreme natural phenomena such as external floods and seismic activity.

Modifications are also required in the models used to estimate the consequences of severe reactor accidents considered by PRAs. Present risk codes for predicting fission product behavior do not reflect the results of ongoing and planned experimental research programs. Improved quantification of the uncertainties in likelihood and consequence prediction is needed to improve the use of PRA in decisionmaking.

4.1.2 Research Program Description

This research is being conducted with the ultimate objective of integrating the use of PRA into the regulatory process. The program will provide detailed, documented methods to ensure consistency in PRA execution. These methods will be exercised and tested in an integrated and realistic manner in the Risk Methodology Integration and Evaluation Program (RMIEP) by applying the newly developed methods in a test bed probabilistic risk analysis. Finally, improvements will be made in methods to facilitate the use of these techniques (i.e., by making them faster and simpler).

The program is a blend of short-term information and analyses needed to support high-priority Commission decisions and longer-range activities required to

refine the methods for increasing the applicability of PRA to a greater range of generic problems.

The major research products will be:

1. Identification and review of accident sequences and their likelihood, including accident precursors, using improved methodology and development of simplified models combined with data bases that will permit evaluation of a wide variety of issues and review of operating history from a risk perspective (1986-1987).

2. A procedure to incorporate dependency analyses, including common-cause failures, into a PRA (1986-1987).

3. Procedures for incorporating results of simplified Seismic Safety Margin Research Program methodology into PRA methodology (1987). (To be developed and applied for trial use in the RMIEP).

4. a. Computer-based modeling for developing human error probability for selected nuclear power plant operations and support personnel functions (1989).

   b. Procedures for fully integrating human risk analyses into the PRA process to achieve more adequate assessments of the effects of human performance on overall plant risk (1987-1990).

5. Methods for systematically using human risk analysis and PRA results to identify man-man and man-machine interfaces of safety importance at nuclear power plants (1987).

4.2 Data Base Development and Evaluation

Data collection and analysis efforts serve as a foundation for the application of risk methods to design, regulation, and operational evaluations.

4.2.1 Major Regulatory Needs and Their Justification

1. Improved data bases for estimating component and system failure rates, extending to root causes of failure and appropriate classification systems, to support regulatory evaluations of significant safety issues (1986 and beyond as additional data become available).

2. Methods to assess and interpret the quantitative uncertainties in PRA modeling and data in support of regulatory decisionmaking, including importance measures and sensitivity analyses (1986-1990).

Justification of Above Needs: As discussed in Section 4.1, quantitative assessments of probabilities and consequences of severe accidents are becoming an increasingly important element in regulatory decisionmaking. The decisionmaker should be provided with quantified results based on the best data available and with a complete statement of the uncertainties associated with these estimated results.

4.2.2 Research Program Description

Together with the methods developed in Section 4.1, the data evaluation projects are being conducted to better integrate PRA analyses into the regulatory process. This requires that a common data base be established for use in quantitative safety analysis. This research will continue compiling existing raw data, augment existing techniques for analyzing data uncertainties, and generate a data base that can be used for generic reliability purposes.

The data base development and evaluation program is comprised of two concurrent phases. The first phase contains a series of tasks dedicated to continued systematic planning and investigation and data acquisition tasks of particular import and urgency. The second phase consists of those long-term data acquisition tasks identified in the detailed data acquisition prioritization effort in Phase 1.

The major research products will be:

1. Component and system reliability data base developed from evaluations of plant operating data, licensee event reports (LERs), and vendor information (1986 with annual revision to reflect most recent data and operating experience).

2. Testing of integrated methods using improved data involving internal, external, and common-cause risk assessment techniques in the RMIEP, for better identifying and displaying quantitative uncertainties (1986).

4.3 Regulatory and Inspection Applications

As the evaluation of data and the development of risk methods proceed, they will be available for application to aid in resolving current regulatory issues and for incorporation into NRC inspection and licensing programs. The research programs described here apply to (1) the development and demonstration of methods to ensure that the accepted level of risk associated with a specific plant is maintained at that level over the lifetime of the plant by ensuring that nuclear plant systems needed for safety are sufficiently reliable and to provide the technical basis for future Commission actions relative to operating plants, (2) the development of methods for identifying cost-effective procedural changes to reduce the risk with regard to severe accidents and in the areas of emergency response and safeguards, (3) the improvement of time-dependent reliability modeling needed for optimization of technical specification limiting conditions for operation from a risk perspective, and (4) the development of methods to assist in focusing the activities of NRC inspectors on areas important to risk.

4.3.1 Major Regulatory Needs and Their Justification

1. Demonstration of the feasibility, usefulness, and cost effectiveness of the selected operational safety reliability methodology by applying it to two nuclear power plants, a PWR and a BWR (1986-1987).

2. Assistance in providing risk-based information to Inspection and Enforcement (IE) to assist in the development and implementation of modules for CP, preoperation, and operating license (OL) inspections, which take into

account current analyses of accident sequences, plant operating data, and accident likelihood (1986).

3. Development of strategies for identifying simple procedural changes or minor equipment modifications to reduce the risk associated with severe accidents in a cost-effective manner, to aid in the implementation of severe accident policy (1986, 1987).

4. Determination of protective action effectiveness under various sets of representative site conditions, accident sequences, and plant containment types and determination of the impact on protective action effectiveness based on the new source terms and ability to predict accident progress (e.g., containment failure time) (1986).

<u>Justification of Above Needs</u>: The NRC has a continuing responsibility to ensure that the risk to the public presented by nuclear power plant operations is maintained at acceptable levels. The NRC must therefore ascertain that licensees have in place (and maintain) adequate procedures for the installation, operation, maintenance, and testing of systems and equipment important to safety based on their desired levels of reliability, commensurate with their influence on an overall plant risk.

The NRC must also ensure that it focuses attention on the principal contributors to risk and provides appropriate procedures for ensuring that plant operators are knowledgeable with regard to principal accident sequences and are trained to respond appropriately to abnormal events.

The NRC is developing and about to implement a severe accident policy statement. Low cost-effective procedural changes and modest equipment modifications, if identified, can offer attractive alternatives to major plant modifications or to regulatory requirements on emergency preparedness. These alternatives may be easier to regulate and more efficient for both the regulator and the licensee. There is a need to develop methods for identifying such modifications and for preparing value-impact evaluations of the proposed changes.

Appendix E to 10 CFR Part 50 requires that the licensee recommend the appropriate protective action to offsite officials. This research will provide a basis to develop improved guidance on identifying the most effective protective actions under various sets of representative site conditions, accident sequences, and plant containment types for use by the NRC Operations Center and regional response teams. The protective action strategies and conditions to be considered will cover prompt evacuation of the area near the site before a release, evacuation of the area within the path of the plume, and sheltering. This research will also provide a basis for revising or developing NRC inspection procedures used to evaluate the adequacy of licensee emergency response procedures.

4.3.2 <u>Research Program Description</u>

This program will develop the necessary management structure, procedures, methods, and requirements, using information gained from nonnuclear applications of reliability assurance techniques to the extent practical and feasible, to provide assurance that nuclear power plant safety systems meet and maintain

desired reliability levels. Due consideration will be given to information gained from research conducted at Kennedy Space Center, Rome Air Development Center, and other sources. The program will be developed so as not to duplicate either the requirements of Appendices A and B to 10 CFR Part 50 or other regulatory requirements related to maintaining reliability. The recommended reliability program will be capable of being directly integrated into NRC and industry practices. Elements that appeared effective for operating reactors were developed based on a screening evaluation of practices in other industries and guided by a working group of experts from other agencies (NASA, FAA, Navy, etc.), industry (EPRI, etc.), and NRC (IE, the regions, etc.). In 1986, a realistic trial application of the selected elements will be completed (on both a PWR and a BWR) to demonstrate the prospects for cost effectiveness, institutional compatibility, and technical coverage of potential reliability problems by reliability assurance program (RAP) elements. The research will result in guidance, suitable for use by NRR, IE, and industry, on the adaptation of RAP elements to the institutional and technical context of nuclear plant safety assurance. Coverage will include design, construction, startup, and operations and maintenance phases of the plant life cycle. The IE inspection program will be reviewed and, where possible, PRA-based information will be developed to (1) assist inspectors in setting priorities for new and current activities and (2) aid IE in developing inspection modules that can be directly related to reducing plant risk.

The study of management of accident risk will focus on means for preventing or delaying containment failure in severe accident situations as well as on means for preventing severe accidents. Because of the large effort expended by the industry and NRC on emergency operating procedures, this research will be addressed primarily to post-core-damage considerations. Sequence event trees, response trees, operation action event trees, and detailed containment event trees, together with conventional fault trees and event trees, will be employed.

Emergency preparedness research provides an improved technical basis for regulatory needs on positions, regulations, and guidance to ensure adequate emergency preparedness at nuclear facilities. Close coordination and cooperation with the Federal Emergency Management Agency (FEMA), EPA, and with State and local government authorities will continue.

The major research products will be:

1. Summary in a final report of the results of demonstration and recommendations for reliability assurance program (1986).

2. Based on risk assessment insights and updated analysis of dominant accident sequences, information for PRA analyses to be developed to aid in developing and setting of priorities for IE inspection activities (continuing program, 1986-1990).

3. Plant-specific analysis of accident management strategies (1986) and published methods application guide for use on other plants (1987).

4. Evaluation of emergency action level identification (1986).

Deferred/Unfunded Research Needs (as of May 1, 1985)

In anticipation of possibly significant reductions in the FY 1986 budget, RES decided, as a matter of policy, to eliminate all further research in the area of Human Factors beginning in FY 1986. This program area was funded in FY 1985 at a level of $1.2 million and comparable funding had been programmed into earlier FY 1986 budget estimates. As a result of this decision, we believe it will be possible, at the currently proposed funding level, to maintain a program that will achieve its major objectives, as described in this chapter. It should be recognized that RES will nevertheless maintain a strong program in the area of human reliability as such reliability affects the risk presented by nuclear power plants.

5

5. THERMAL-HYDRAULIC TRANSIENTS

This program provides the experimental data and analytical methods needed to predict and understand the operation of primary and secondary coolant systems during all types of plant transients, including the full size range of pipe ruptures. The resulting analytical methods are used to quantify margins of Appendix K to 10 CFR Part 50, to assist the regulatory assessment of operator guidelines for accident management, and to analyze the operational control of complex plant system transients as well as their recovery to normal conditions. The research related to Appendix K is nearly completed and will culminate in revisions to Appendix K this year. A summary of this completed LOCA research will be prepared in 1985. The data obtained through the separate effects research programs enable code developers to produce models that represent physical phenomena. The assembled models are then compared with the data obtained from the scaled system test facilities, both PWR and BWR, to determine whether or not the models developed from individual testing adequately represent those phenomena occurring in an integral system. The ability of the computer codes to adequately predict systems test results is providing increased confidence in the predictions made by the codes of full-sized plant behavior under similar conditions.

The emphasis has now shifted from LOCA research to the application of codes to the analysis of plant transients. Problems encountered in these applications frequently require the development of specific models such as fluid mixing in the downcomer. This sort of model development is often supported by the testing of systems response in facilities such as Semiscale, FIST, PKL, LOBI, and ROSA. The past emphasis has been on LOCA-related research for both computer code model development and code assessment. The shifting of research emphasis will provide a similar capability for abnormal transients (those generally occurring as a result of equipment failure or operator error) and conditions such as pressurized thermal shock (PTS). The elements contained in this chapter include separate effects experiments and model development, integral systems experiments, code assessment and application, and plant analyzer and data banks.

5.1 Separate Effects Experiments and Model Development

This research consists of experiments designed to provide data specific to various phenomena such as two-phase (steam/liquid) heat transfer, downcomer thermal mixing, and flow characteristics in the range of conditions that occur in reactors during transients and accidents. Such experiments are performed under well-controlled conditions to provide data to develop and assess accurate correlations of the parameters used for the prediction of these phenomena.

5.1.1 Major Regulatory Needs and Their Justifications

1. Evaluation of integral transient behavior from facilities of varying size, to resolve scaling issues (1986).

Justification: Results of scaled reactor system tests must be confidently extended to full-sized reactors. Small test facilities are needed to support major facilities to investigate the scaling of the phenomena. Scaling also must be understood to determine if small facilities can be used to replace larger and more expensive facilities as the identified work in the larger facilities is completed.

2. Revision and assessment of heat transfer package in the RELAP and TRAC codes, to be used in evaluating operator guidelines (1987).
   Justification: The heat transferred from fuel rods to water is calculated in current RELAP and TRAC code versions according to correlations such as the "re-wet criterion" and critical heat flux (CHF) that are known to be conservative when used in emergency core cooling system (ECCS) licensing evaluations. The conservatisms have been found to be unrepresentative of actual phenomena. More representative correlations should therefore be developed and introduced for use in the analysis of system response to anticipated transients. Such system response analysis is used to evaluate licensee submittals of operator guidelines.

3. Validation of steam generator heat transfer models, including the effects of tube rupture and iodine transport through the secondary system, for the balance of the plant (BOP) in advanced codes for use in evaluating anticipated transients (1986).
   Justification: NRR routinely relies on the use of TRAC and RELAP computer codes to evaluate various accident scenarios involving steam generators during the licensing process. Because current steam generator models in TRAC and RELAP are not based on directly relevant experimental data, unknown uncertainties may be introduced in the above analysis. The data base generated by these experiments will quantify these uncertainties and thereby point to the need for code modification, if any.

4. Determination of critical flow through pipe cracks (1986).
   Justification: Stress corrosion cracking has been observed in BWR piping. It has been postulated that leakage through a crack would be of a sufficient quantity to be detected prior to the crack's growing large enough to rupture the pipe. The existing critical flow models are not based on data from cracks, however, and applicable data are required to evaluate or develop new models.

5. Fluid temperature fluctuations at the intersection of the high-pressure injection (HPI) lines and the cold legs of BWR and PWR systems at a variety of HPI and cold-leg fluid mixing conditions (1987).
   Justification: Failure of the weld that joins the HPI line and cold leg would simultaneously initiate a small-break LOCA and reduce the ability of the ECCS to mitigate such an event. Only a limited amount of experimental data is available on thermal fluid mixing at the HPI and cold-leg interface. This information was requested by NRR to determine the possibility of cyclic fatigue failure at these welds.

6. Boron mixing in the lower plenum of BWRs during a postulated anticipated transient without scram (ATWS) (1987).
   Justification: Data on mixing of cold boron solution are required so that computer codes can accurately analyze ATWS in BWRs.

5.1.2 Research Program Description

PWR loop experiments at the Massachusetts Institute of Technology (MIT) are being performed to learn which phenomena are responsible for flow oscillations observed during natural circulation in PWR system components. Results from this program will provide a definition of the unexpected states that are possible in a partially voided reactor system and, in turn, will aid the reactor operator when flow reversals are observed under natural circulation conditions.

A small-scale B&W simulation reactor loop at the University of Maryland has been completed to provide (1) separate effects testing of important transients and (2) scoping tests for input to the B&W Multiloop Integral System Test (MIST) facility. A related facility is being sponsored by EPRI at the Stanford Research Institute. In addition, separate effects testing of the B&W hot leg is being performed to determine conditions responsible for flow interruption and resumption.

Development of two-phase flow models and correlations is continuing for the purpose of providing a foundation needed for validated LWR safety analyses. This program will be investigating (1) inverted annular flow and the mechanism for jet breakup, (2) similarity laws under natural circulation conditions, and (3) a hydrodynamic model for entrainment of water from a pool.

A thermal fluid mixing program is being planned to obtain thermal-hydraulic data that can be used to develop and assess models that describe the extent of thermal fluid mixing in a reactor downcomer and cold leg as a result of ECC injection and to apply to boron mixing in BWRs. This research will provide NRR with valuable support in evaluating the PTS issue and ATWS analysis.

The MB-2 program is being conducted to study heat transfer during small breaks, tube rupture events, and other system transients. Effects of tube rupture, including iodine transport through the secondary system, are also being investigated under this and other programs at Northwestern University, MIT, and the Oak Ridge National Laboratory (ORNL).

The major research products will be:

1. Evaluation of scaling of integral test data and of scaling methods at alternative reactor system test facilities (1986).

2. Revision of heat transfer packages in RELAP5 and TRAC, based on assessment results (1987).

3. Validated steam generator data from MB-2 (1986) and improved models for iodine transport through the secondary side (1987).

4. Critical flow models for pipe cracks (1986).

5. Thermal mixing model to predict fluid temperature fluctuations at the HPI and cold-leg interface (1988).

6. Boron mixing model (1987).

5.1.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

GE has proprietary data on boron mixing that they are willing to sell to the NRC. NRR has requested that RES obtain the data. In addition, a small university program was planned to extend the GE results, which RES has informally reviewed. Action on these items has been deferred until funding availability has been resolved.

In order to provide a stable base of expertise in fundamental understanding of thermal-hydraulic processes under reactor accident conditions, a strong research program is needed in universities. The development of university "centers of excellence" for both analysis and experiments was severely curtailed by FY 1985 and FY 1986 budget restrictions. Past experience has shown that some of the best work on fundamental understanding of reactor safety thermal-hydraulic processes came from universities.

5.2 Integral Systems Experiments

This element includes experimental simulations of integral thermal-hydraulic systems of PWR and BWR reactors. The United States facilities involved are the Semiscale, MIST, Once-Through Integral System (OTIS), and Loss-of-Fluid Test (LOFT) facilities that simulate PWR behavior and the Full Integral Simulation Test (FIST) facility that simulates BWR behavior. (The LOFT facility is operated under an international consortium.) Current plans call for the operation of the Semiscale facility through 1986 and the completion of the MIST/OTIS program in 1987. The FIST program has been completed, but the facility will be maintained. Foreign agreements and understandings provide for data exchanges with similar facilities in the Federal Republic of Germany (PKL and UPTF), Commission of the European Community (LOBI), and Japan (ROSA IV, CCTF, and SCTF). Transients simulated include the full-break-size spectrum of LOCAs, loss of feedwater, steam line and feedwater line break, steam generator tube rupture, ATWS, and various safety and control system failures. These scenarios are defined in close communication with the NRR staff to help them obtain the data needed to resolve related licensing and safety issues. Since the identified programs for these facilities are nearing completion, a major study of future needs in this area is being performed. This element is closely associated with TRAC and RELAP5 code improvement, maintenance, and assessment in that the TRAC and RELAP5 codes are tested against experiments conducted in these facilities in order to improve and validate the codes (see Section 5.3).

5.2.1 Major Regulatory Needs and Their Justifications

1. Experimental data from scaled simulations of reactor transients, including steam line breaks, feedwater line breaks, small-break LOCA without high-pressure injection, large-break LOCA with upper-head injection (UHI) and upper-plenum injection (UPI), and scaling tests in Westinghouse and Combustion Engineering geometry (1987).

2. Experimental data from scaled simulations of small-break LOCA reactor transients in B&W geometry as a result of TMI-accident concerns (1987).

3. Evaluation of NRC's long-term needs for integral facilities as the identified programs are completed and plan for integral facility to accommodate these needs, including unusual plant designs such as BWR-2 (1986).

4. Upgrading of the MIST facility to allow full-power testing of B&W geometry for transients such as steam line and feedwater line breaks, loss of feedwater, steam generator overfill, ATWS, station blackout, and feed/bleed using high-point vents (1988).

Justifications of Above Needs:

1. A need exists to evaluate the calculations of transient conditions used as a basis for specifying operator actions in response to various plant transients to ensure the adequacy of these procedures. The actual response of the systems, as well as the indicated information provided by various instruments during transients, also needs to be evaluated to ensure that the operator can correctly identify the transients and take appropriate actions. It is especially important that the operator be able to recognize precursor events in order to be able to head off potential accidents and mitigate potentially serious accidents. These tests will be used directly and, through use of the data, to assess computer codes, to identify precursor events, and to improve our understanding of PWR and BWR transients and (in conjunction with the work on operator interactions) the adequacy of operator guidelines.

2. Revisions to operator guidelines, safety system setpoints, and additional safety systems are periodically proposed to the NRC by reactor owners. The data obtained from these test programs, in conjunction with improved calculational capability, will allow a better evaluation of the risk from PWR and BWR operation and the influence of proposed changes on the risk.

3. Past LOCA research has identified a large margin in the LOCA evaluation model calculations, which indicates the potential for relaxing some operating restrictions through the use of improved evaluation models. This research will provide additional information for use by the NRC staff in proposing revisions to evaluation model requirements and for use during review of improved evaluation models submitted by applicants.

5.2.2 Research Program Description

The approach taken to meet the needs described above is to provide an experimental program integrated with the development of calculational capability. In an iterative manner, codes are used to plan experimental simulations of the various transients. The results of the tests then provide a basis to judge the adequacy of the codes. The data from these scaled reactor system transient simulations form the basis for assessment of the codes and thus contribute toward improved calculational capability. These codes are then used to determine the reactor plant response during transients and the influence of operating procedures and equipment malfunctions.

The Semiscale facility testing in 1985 and 1986 will concentrate on accident and transient sequences that have been identified as being needed to provide data for assessing aspects of computer codes not previously assessed. The tests will be designed to evaluate both the anticipated sequences and typical operational procedures. Computer code analyses will be assessed on the basis of the test data. This will provide further understanding of the abilities of large systems codes (such as RELAP5 and TRAC) to calculate transient conditions.

Under a bilateral agreement with Japan, data from ROSA IV will be available. Tests of small breaks, natural circulation, steam generator heat transfer, alternative ECC systems, and transient recovery techniques will be conducted in 1985 and 1986. This program will provide integral PWR data beyond the planned phaseout of Semiscale and data from a new and different facility for further assessment of RELAP5 and TRAC.

The Integral System Test (IST) program has recently been formulated to investigate the unique features of the B&W reactor system. Experimental data are needed to address licensing concerns, to verify operator guideline procedures, and to assess code capabilities to predict B&W design-related phenomena.

The IST program consists of two series of tests conducted in the OTIS and MIST facilities. The OTIS facility is a 1x1 (1 cold leg, 1 hot leg) representation of the B&W raised loop configuration without reactor coolant pumps. OTIS testing primarily investigated small-break LOCA events with completion in 1984. The MIST facility will employ two hot legs, two steam generators, four cold legs, and pumps to model the 2x4 lowered-loop B&W design. MIST testing to be conducted in 1986 will include 6 months of debug and characterization tests followed by 6 months of composite testing. The composite test matrix consists of 41 tests divided into four general groups--small-break LOCA, natural circulation, steam generator tube rupture, and feed-and-bleed cooling. A follow-on program for full-power MIST testing in 1987 is being developed.

The Department of Energy (DOE) owns and operates the LOFT facility, which is under the sponsorship of the OECD LOFT project, and funding is shared by approximately 10 countries and agencies. The NRC is a member of the project based on a one-time contribution of $25 million in FY 1983. Included in the test program will be a complete loss of feedwater, small hot-leg breaks with and without primary pump operation, double-ended large breaks in the cold and hot legs, fission product pathways during large-break LOCAs, a small lower plenum break, and the NRC's fuel clad balloon and burst test, L2-6. This program will run from 1983 to 1985, with follow-on analyses through 1987.

Identified testing has been completed in the FIST facility, but the facility will be maintained so as to be available if the needs for additional testing are identified.

Full-scale testing of ECC performance in the international 2D/3D facilities (UPTF and SCTF) will be accomplished during 1986 and 1987. Data will be provided on fluid-fluid mixing (PTS-related), ECC bypass, upper-plenum deentrainment, and fallback and condensation effects.

RES is initiating a study of NRC's long-term needs for integral facilities and how best to meet these needs. Studies will include the evaluation of smaller facilities using alternative scaling methods to permit more cost-effective maintenance of test facility capability.

The major research products will be:

1. a. Semiscale data to be used to assess steam line and feedwater line break calculations (1986).

   b. Semiscale data to be used to assess small-break LOCA without HPI (1986).

   c. Semiscale data to be used to assess analysis codes for large-break LOCA in UHI-equipped plants (1986).

   d. Semiscale tests similar to LOBI and ROSA IV tests to study scaling (1986).

   e. Full-scale fluid-fluid mixing and ECC bypass data from UPTF and SCTF (1987).

2. Experimental once-through-steam-generator tube rupture data from OTIS and MIST:

   a. Integral natural circulation data from MIST (1986).

   b. Integral small-break LOCA tests in MIST (1986).

   c. Integral feed-and-bleed tests in MIST (1986).

   d. Further B&W transient simulations (1987).

3. Evaluation and recommendations as to how best to meet NRC's long-term needs for integral facilities (1986).

4. Full-power testing in MIST (1988).

5.2.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

Three additional feed line/steam line break tests and two additional small-break LOCA tests were requested by NRR to be performed in Semiscale to accommodate their data requirements. These tests must be cancelled because of the decision to close Semiscale at the end of FY 1986.

No agreement has yet been reached on the additional funding needed for FY 1987 to make the full-power modifications to MIST necessary for performing the additional tests requested by NRR.

Because of the delay, due in part to funding deficiencies, in deciding on the design of an Advanced Test Facility, there will be no United States integral facility to perform tests on Westinghouse and Combustion Engineering or GE geometry after FY 1986. At best, with the proper fiscal planning, it will not be until FY 1988 that such testing can be accomplished.

5.3 Code Assessment and Application

This element includes the application of computer codes to the analysis of transients in full-scale LWRs to help resolve licensing and safety issues such as pressurized thermal shock and the assessment of these analytic capabilities against experimental data in order to ensure the accuracy and reliability of the computed results. Development of RELAP5 and TRAC was completed in 1984. These codes will basically remain fixed until the end of 1986. Improvements as needed will be incorporated in 1987. These improvements may be required by NRR to resolve licensing issues or to provide user-friendly features. There will be additional requirements by the participants of the International Code Assessment Program to resolve their licensing issues and to improve reactor operations.

5.3.1 Major Regulatory Needs and Their Justifications

1. Maintenance and evaluation of operational TRAC and RELAP5 codes (1986-1987).
   Justification: Assessments of the fixed versions of these codes are important to the licensing staff. Screening criteria are to be developed and applied. Changes are to be limited to those that would affect perception of transients and accidents. User-requested features such as automatic initialization will be added.

2. Assessment of computer codes for use in resolving licensing and safety issues (1990).
   Justification: Use of these codes to analyze highly complex reactor transients and accidents in full-scale geometry requires extensive assessment against test and plant data to ensure the reliability of results.

5.3.2 Research Program Description

The strategy for the research in this element has been to develop methods and codes and simultaneously test them against data. At an early date, each code is prereleased to a group independent from the DOE laboratory developers (usually other DOE laboratories) for more extensive testing. Beginning in 1984, foreign organizations are also participating in the assessment effort. Collection and analysis of nuclear data and plant design and operating histories are ongoing efforts that support this element. Technical information developed jointly with outside groups (EPRI, foreign governments) is often used to improve and assess the NRC computer codes. RELAP5 and TRAC will be used in a fixed form in the 1986-1987 period.

This research program will develop code packages for analyzing complete LWR systems for transients ranging from anticipated transients through design basis events. In addition, methods to predict system behavior for small-break transients coupled with operator actions will be developed both for a fast-running and a detailed analysis code. Physical models and numerical methods for prediction of two-phase flow behavior will be improved in the codes.

The independent assessment of these set (or fixed) codes will be performed by DOE laboratories and by various foreign organizations either acting as agents for their governments or conducting research under government contracts. NRC is in the process of negotiating bilateral agreements with various governments to implement this program. Essentially all independent assessment at DOE laboratories will be completed in 1987 after which changes based on the application of the codes during the 2-year period will be incorporated and a small but important program for independent assessment of PWR and BWR plant analyzers (see Section 5.4) will be conducted at the DOE laboratories. Support for the International Code Assessment Program is planned to increase over the next several years as resources are shifted from code development and independent assessment at DOE laboratories. Code assessment by foreign organizations should continue until 1990. Using the results of the International Code Assessment Program, the codes will be maintained and improved.

After 1984, the advanced system transient codes will be maintained and improved based on independent assessment results from the International Code Assessment Program and test data to be obtained from facilities such as UPTF, CCTF, SCTF, ROSA IV, MIST, and the new integral test facility (Section 5.2.2).

The major research products will be:

1. a Development of statistical uncertainty for large thermal-hydraulic analysis codes (1986).

   b. Summary status report on ECCS research (1986).

   c. Summary reports on assessment, accuracy, and limitations of frozen versions (1985) of TRAC and RELAP5 for PWRs and BWRs, including BWR-2 plants and PWRs with UHI and UPI (1986).

   d. Improved versions of the advanced multidimensional two-fluid transient analysis codes: TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1, including user-requested features such as automatic initialization (1987).

   e. Assessment of TRAC-BF1 by DOE laboratories (1987).

2. Status of assessment results from International Code Assessment Program and their significance in resolving licensing issues and improving reactor operations (1990).

5.3.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

Automatic initialization, as well as other user-requested features, will have to be delayed until the next frozen versions of the codes are released in FY 1987. It is believed that the present initialization features are adequate, although not optimal, and such a delay should not lead to major inconveniences.

Improvements to models for countercurrent flow limitation (CCFL) post-CHF heat transfer, turbulent behavior in the upper plenum, and condensation due to ECC injection will be delayed and handled on a priority basis as funding permits.

5.4 Plant Analyzer and Data Bank

This element includes computational technique improvements and the development of user-oriented capabilities to use advanced versions of these codes in the form of an interactive plant analyzer. It also includes the acquisition and manipulation of plant data needed to develop input specifications for plant-specific analyses. This analytic capability is being used directly by the licensing staff to perform needed analyses and to provide user input in the later stages of development.

5.4.1 Major Regulatory Needs and Their Justifications

1. User-convenient system analysis codes for use in evaluating transients and accidents with the capability for interactive operational manipulations at midpoints during long transients (1986).
   Justification: NRC and contractor personnel need the capability to perform analyses of full-scale LWRs in order to help resolve safety and licensing issues in a timely fashion. These analyses should be easily initiated, should be economical to run and fast, should allow user interaction, and should provide easily understandable output results in order to have high utility. To meet these needs, the plant analyzer is being developed. Minimum software development is planned. Initial versions are now in use by the licensing staff.

2. Geometric and operating data for selected licensed plants to allow plant-specific calculations to be performed (1987).
   Justification: The conduct of these analyses will be facilitated if the plant-specific data are contained in a data bank. There must be a complete set of geometric as-built data and thermal-hydraulic and neutronic characteristics available in computer language. These data will be automatically converted into input decks for the plant analyzer. A program for obtaining and inputting plant-specific data will be developed for timely implementation.

5.4.2 Research Program Description

As the development phase of NRC codes is nearing completion, more emphasis is being placed on making them available in a user-oriented form. This development is focused in three areas: (1) use of the latest available computer hardware and improved software to allow computation time of up to ten times faster than real time for LWR system transients; (2) display of the computed transient on terminal consoles so that the user can easily understand the calculated results and interact with the calculation, if desired; and (3) incorporation of LWR plant data into a data bank that is easily accessible for the development of input decks for computer codes and the plant analyzer. Colorgraphic replay of previous analyses will be available, as will selected experimental results. In addition to user convenience, the plant analyzer and data bank will provide long-term benefits in the traceability and auditability of reactor data, code input decks, and reproducibility of results.

While the major effort has been devoted to use of the codes as programmed for large central computers, a small effort was undertaken to investigate the potential benefits of reprogramming existing codes for small special-purpose, high-speed computers that would serve as dedicated machines. A terminal tied to the INEL computer was provided to NRR in 1984. This office can now provide user input directly for the further development of the nuclear plant analyzer.

The specifications of the plant analyzer reflect our current experience with the RELAP5 and TRAC codes. Since it commonly takes 4 to 6 months to prepare an input deck for such codes, it is planned to use information stored in a plant data bank to compile input decks in an automated fashion with minimal input from the user. The data bank itself is in existence and functioning, but the production of input decks represents significant effort, and the planning of such work is now being addressed. Similarly, there is a need for interactive features to allow the user to follow a computed transient in detail and to intervene at some step to make a change in plant condition to replicate a projected action. Current practice would require successive restarts of runs, approximately three per change for long runs to follow a transient. An interactive feature being developed will significantly reduce computational effort and speed the acquisition of results. In addition, a goal of the development effort is to minimize running costs. Since it is probable that the analyzer will be in use during prime computer rate periods, this goal has high priority.

Computer technology is developing, and it is expected that more advanced computational techniques such as vectorization and parallel processing will become available in future years. Using the advanced computational techniques will increase the speed of calculations. Special plant analyzer versions of the advanced systems codes (Section 5.3) will be maintained and improved as the computer technology develops.

The major research products will be:

1. a. Demonstration of typical PWR plant analyzer with incorporated plant data bank (1986).

   b. Speeding of TRAC by vectorization and two-step numerics in vessel components on CRAY-1 and speeding of RELAP5 by conversion to CRAY-1 (1986).

   c. First general user version of PWR system plant analyzer completed, with input decks, and demonstrated for use by NRC personnel (1986).

   d. Advanced version of PWR system plant analyzer with complete developmental assessment (1987).

   e. First general user version of BWR system plant analyzer with complete developmental assessment and demonstrated for use by NRC personnel (1988).

   f. PWR and BWR plant analyzers improved, independently assessed using experimental and plant data, and maintained for all NRC codes being used in analysis (1987-1988).

2. a. Plant secondary systems and control systems implemented in plant data bank (1986).

   b. Use of plant analyzer and plant data bank to analyze transients in full-scale LWRs to resolve licensing and safety issues (1986-1990).

5.4.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

Because of difficulties in implementing the nuclear plant data bank software and in obtaining plant data from utilities, the effective use of the data bank will have to be delayed unless additional funding is provided to fix the software and to develop the data-entry books for the utilities.

6. SEVERE ACCIDENTS

This program provides the data base and validated methodology for the reassessment of the regulatory treatment of severe accidents. It includes the coordinated phenomenological research programs needed to develop a sound technical basis for NRC decisions concerning the ability of reactors to cope with these accidents. The following elements are included in this chapter: accident likelihood evaluation, severe accident sequence analysis (SASA), behavior of damaged fuel, hydrogen generation and control, fuel-structure interaction, containment analysis, fission product release and depletion or transport, containment failure mode, fission product control, risk code development, accident consequence and risk reevaluation, and risk reduction and cost analysis.

The program outlined in this chapter anticipates that a significant level of confirmatory work will be required following the Commission's final decision on severe accident policy. It also attempts to address areas of phenomenological uncertainty that were identified as part of the American Physical Society review of NRC source term research. This program is also addressed in a forthcoming supplement to NUREG-0900, "Plan for Severe Accident Research and Regulatory Implementation." This chapter will be reoriented in 1986 to reflect intervening Commission policy decisions.

6.1 Accident Likelihood Evaluation

Work in this element is directed toward the reassessment of severe accident scenarios and their related probabilities. This reassessment will be made using, among other things, the results of recently completed probabilistic risk assessment (PRA) studies, safety studies, new data on system and component reliability, and evaluation of licensee event reports (LERs). In this regard, this element will pull together the information gained from other accident likelihood assessments, will update this information with current knowledge, and, based on this update, will reassess the predictions of severe accident sequences, their likelihoods, and, to a degree, their consequences. Increased emphasis will be placed on the identification of common-mode failures and operator response. This information will then be provided for use in other severe accident evaluations and will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.1.1 Major Regulatory Needs and Their Justifications

1. Identification and probabilistic evaluation of accident sequences, including uncertainties that would have the potential for leading to severe core damage or core melt (1986-1987).
   Justification: Identification and description of the principal accident sequences that could lead to core damage will provide an essential element for defining the scope and direction of the SASA program (see Section 6.2), for the systematic analysis of the risk-reduction effectiveness of backfitting safety systems on operating plants (see Section 6.12), and for other NRC programs and applications, including incident reviews, assessment of generic issues, and real-time diagnosis and prognosis of incidents in progress. Identification of plant classes will also be provided to the severe accident elements and to other regulatory activities.

2. Validation of the information base of accident sequence likelihood characteristics based on recent PRA studies, LER data, and updated plant information (1986-1987).
   Justification: Future decisions on the incorporation of additional safety requirements into the regulations, elimination of unnecessary requirements, and backfitting on operating plants require that the quantification of accident likelihoods be made using the most recent data available.

3. Automation of risk assessment tools (event tree and fault tree models) and the accident sequence likelihood characteristics information base (1986-1989).
   Justification: Automation or computerization of the accident likelihood evaluation results can improve the NRC plant operational assessments for determining the risk status of plants from implication of generic issues, backfits, etc.; evaluating operational experiences to determine the effectiveness and pertinence of NRC regulations; ensuring NRC effectiveness during emergency response situations; and improving the NRC inspection effectiveness.

6.1.2 Research Program Description

The basic objective of this research is to provide a reliable set of major accident sequences and their related probabilities for use in defining the direction of the SASA program (see Section 6.2), in supporting decisions on backfitting additional safety features to existing plants (see Section 6.12), and in supporting NRC's overall goals in plant operational assessment. This research will also attempt to provide a set of generic plant classes for use in the overall severe accident rulemaking.

Reviews are to be made of the accident sequence evaluations in plant-specific risk assessments such as the Reactor Safety Study (RSS), the RSS Methodology Applications Program (RSSMAP), and several industry-sponsored PRAs. These risk assessments and the accident sequence likelihood assessments from this program will be generalized for applicability to the spectrum of current plant designs and used as the foundation from which the second and subsequent risk/cost analyses of possible plant modifications will be made (see Sections 6.2 and 6.12). The objective of the accident sequence reevaluation will be to consider and incorporate new information and make them more generically applicable for use in the value/impact analyses. More specifically, modifications will be made to differentiate between sequence variations of various plant types that are important for the value/impact analyses needed to support decisions on the scope of the severe accident rulemaking. This work is also being directed toward establishing the feasibility of determining, on a generic basis (i.e., to specific classes of plants), the need for augmented safety features such as filtered vents, additional decay heat removal, etc.

To provide additional validation of accident likelihoods, events in operating LWRs are being reviewed for their potential, when combined with other events, to lead to a severe accident. After an initial screening to define the more important events, estimates of the likelihood of these events resulting in a severe accident will be made.

The major research products will be:

1. Identification, review, and delineation of accident sequences, including accident precursors (1986-1987); reports on likelihood of accident sequences and related probabilities (1986-1987); and identification of generic plant class (1986-1987). (Also applies to Need 2.)

2. Evaluations of plant operating data, LERs, and vendor information for component and system reliability (1986). (Also applies to Need 3.)

3. Automation of risk assessment models and accident sequence likelihood characteristics information base (1986-1989).

6.2 Severe Accident Sequence Analysis

Research uses the analytical assessment of plant accidents beyond the design basis to provide strategies for severe accident prevention, management, and mitigation. Plant models are simulated in best-estimate state-of-the-art computer codes (e.g., RELAP, TRAC, MARCON-2.0B, SCDAP). The results will be used to develop better insights into automated response requirements for the plant and to evaluate operator intervention at the precursor stage and during the course of the accident. The findings will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.2.1 Major Regulatory Needs and Their Justifications

1. Analysis of severe accident scenarios for specific types of plant designs, to be used in licensing reviews (1986-1990).
   Justification: A number of postulated high-risk sequences leading to possible severe accidents are being identified by risk assessments of the Interim Reliability Evaluation Program (IREP) and RSSMAP. Detailed analyses of these high-risk sequences are needed to determine appropriate operator actions and any need for special instrumentation. Specifically recommended operational techniques for managing accident recovery from accident management and human factor research and consequent algorithms to be used by the operator to prevent, diagnose, and respond properly to accidents will be analyzed as a basis for appropriate regulatory actions over the time interval.

2. Resolution of licensing and safety concerns expressed by NRR in its review of operator guidelines (1990).
   Justification: NRR has transmitted to RES for analysis by the SASA program some current safety concerns that arose during the course of licensing reviews and unresolved safety issue (USI) resolution activities. NRR has requested studies to determine the viability of proposed alternative operator actions and the capability of plant systems to restore the plant to a controllable status. A continuing need exists to support NRR by responding to concerns as they arise. The survivability of equipment and the designation of severe accident environmental conditions is an additional concern to be evaluated by the SASA program. It is noted that these are safety concerns with respect to public safety but are not licensing concerns for the design basis accident.

3. Evaluation of information that the operator needs in order to take proper action and evaluation of instrumentation functional needs to enhance the man-machine information flow when accidents occur, this information to be used in regulatory reviews of existing and proposed instrumentation improvements (1987).
   Justification: From the evaluation of plant-specific response characteristics for a range of accident scenarios, requirements for instrumentation to actuate automated preventive action or to inform the plant operator of the need for manual intervention will be more fully identified. Some operational instrumentation systems on operating plants may require re-evaluation as a result of new functional requirements emanating from these studies.

4. Fission product release and transport assessments for use in equipment qualification, probabilistic risk assessment, and the definition of siting and emergency planning requirements (1989).
   Justification: Fission product release rates are being evaluated for dominant accident scenarios. This work will assist in assessing the degree of conservatism currently imposed on licensing requirements with respect to fission product transport and source term models.

6.2.2 Research Program Description

The strategy is to analyze dominant accident sequences derived from risk assessment studies for specific plant designs to evaluate areas of uncertainty and system functional requirements, to assess prevention and mitigation of core melt during severe accidents, to evaluate equipment and system survivability in severe accident environments, and to evaluate the impact of proposed prevention and mitigation features on dominant severe accident sequences. Test programs such as the Power Burst Facility (PBF), Semiscale, and Full Integral Simulation Test (FIST) will be used to provide definition for issues of concern. These test programs will also produce data that can be used to evaluate SASA analysis results. Risk assessment programs such as the IREP will define high-risk sequences for consideration. This program will in turn characterize sequences that have been defined by risk assessment studies and will provide a data base for assessment of IREP-developed methodologies. Licensing and safety concerns generated by licensing reviews will serve to define SASA issues.

The major research products will be:

1. Technical bases for improved risk assessments (1986-1990).

2. Analytical base for licensing rulemaking decisions (Hydrogen) (1986).

3. Basis for detailed licensing decisions (e.g., recent industry proposal and NRC acceptance of manual switchover of reactor core isolation cooling suction in three specific plants) (1986).

4. Basis for ex-containment source term as distinct from source term release to containment (1986-1990).

6.3 Behavior of Damaged Fuel

This element describes research to determine the general behavior of damaged fuel in the 1100K to 3000K (1500°F to 5000°F) temperature range, the fission product release and in-vessel attenuation, the hydrogen release, and the coolability limits in various stages and configurations. The data base and models developed from this research are to provide a technical basis for decisions and actions by NRC concerning accident conditions beyond the current design basis. This element does not cover fuel behavior during operational transients, for which we believe no further research is needed or planned.

6.3.1 Major Regulatory Needs and Their Justifications

1. Determination of the actual hydrogen release from the core and confirmation of regulatory action (1987).
   Justification: The rate of hydrogen generation during a degraded-core accident is dependent on the fuel and coolant behavior in the accident sequence. Evaluation of the adequacy of mitigation and control features requires knowledge of the time and quantity of hydrogen release.

2. Determination of the general behavior of severely damaged fuel in the 1100K to 3000K (1500°F to 5000°F) temperature range, for use in confirming implementation of severe accident policy (1987).
   Justification: Consideration of imposing additional regulatory requirements is dependent on a substantial reduction of uncertainty in the estimated likelihood of core melt during an accident that results in fuel damage. To achieve this reduction, more detailed knowledge is needed on how the core behaves under degraded cooling.

3. Determination of the coolability limits and cooling requirements of damaged cores at various stages of degradation, to be used by the NRC staff in reviewing proposed accident recovery and emergency planning procedures (1987).
   Justification: The key question is: Under what conditions can we be assured the fuel will not melt through the vessel? Data and verified models on the coolability of reactor cores with different degrees of core damage are needed in order to determine the range of conditions for which emergency core cooling system (ECCS) reflood can provide accident recovery. This information is needed for accident management and emergency planning and for risk assessment. It is also needed for assessing the adequacy of ECC systems and operational plans. The TMI-2 accident gave definite evidence that severe core damage does not inevitably lead to full core melt and that recovery from severe accidents is possible.

4. Improvements in PRA consequence calculational methods (1986).
   Justification: The present state of risk assessment techniques suffers from limitations both in the methodology and in the incompleteness of the phenomenological data base. A better understanding of the phenomenology of accidents involving fuel damage will permit realistic treatment and should much improve the usefulness of risk assessment.

6.3.2 Research Program Description

The strategy is to develop, for a range of accident conditions beyond the design basis, a data base and verified analytical models for assessing the state of a severely damaged core, the hydrogen generation, the fission product release from the core and the in-vessel attenuation, and the coolability of the damaged core by reflooding. Key interfaces with other elements include risk assessment, SASA, accident management, hydrogen generation and control, and fission product release, depletion, and transport. The major contribution of this program to risk assessment is in the consequence side of risk, i.e., an improved understanding of the phenomenological aspects of fuel behavior at high temperatures and its subsequent effects on radiological consequences.

The planned severe fuel damage research program is a four-part integrated program of in-reactor and laboratory experiments and analysis. The output of the integrated program is more substantive than the sum of the individual parts.

The first part consists of multi-effect in-pile tests in the PBF at the Idaho National Engineering Laboratory (INEL) and also in the NRU reactor in Canada to provide scoping data on governing phenomena and on multirod interactive effects. The second part consists of separate effects experiments on the governing phenomena, both in the Annular Core Research Reactor (ACRR) at Sandia and in the laboratory, to furnish a data base for model development and assessment and to cover the necessary range of accident parameters on a cost-effective basis. The third part consists of validation of the mechanistic computer codes SCDAP and MELPROG and analysis using these codes. The SCDAP code treats the development of fuel damage in the original core volume, starting with intact rod geometry. The melt progression (MELPROG) code treats the relocation of liquefied and molten fuel and particulate debris as it attacks the core support plate, core barrel, and reactor vessel, the attack on the reactor vessel, and the conditions of vessel failure. These two codes furnish a mechanistic basis for evaluating appropriate parts of such codes as MARCH and MAAP and the advanced risk assessment code, MELCOR. The fourth part of the program consists of benchmark data to be obtained from independent examination of the TMI-2 core.

There will be continuous active interaction and feedback between these analyses and experimental programs. The foundation of the severe fuel damage (SFD) program is the PBF series (four tests) that were completed in FY 1985.

The major research products will be:

1. Report on NRU severe fuel damage (SFD) full-length verification tests (1987). (Also applies to Need 2.)

2. a. Report on analysis of PBF tests (1986).

   b. Final report on independent examination of selected TMI-2 core samples (1987).

   c. MELPROG-Mod 1 code released (1986).

3. a. Report on ACRR debris formation experiments through FY 1985 (1986).

   b. Assessment of MELPROG with PBF and ACRR results (1987).

   c. Report on ACRR melt progression separate effects experiments with irradiated fuel (1988).

4. a. Assessment of final version of SCDAP with PBF, NRU, and ACRR results through FY 1985 (1986).

   b. Report on limited sensitivity studies using MELCOR/SCDAP/MELPROG (1987).

6.3.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

One NRU test has been deferred until FY 1988 as a result of budget reductions in FY 1986. This deferral affects the ability to fully confirm core damage predictions for various accident conditions.

6.4 Hydrogen Generation and Control

Research conducted under this program is providing information and analytical models to quantify the loads on containment from hydrogen burning that could exceed the ultimate strength of the building. The research is providing information to assess the efficiency of proposed mitigation systems. This work includes the development of analytical models that will permit better understanding of hydrogen transport, mixing, and combustion phenomena.

6.4.1 Major Regulatory Needs and Their Justifications

1. Data from all areas of hydrogen research such as generation, ignition conditions, and mixing to support rulemaking with regard to hydrogen control for LWRs with large, dry containments (1986).†
   Justification: These is concern for large, dry PWRs that, in the absence of a hydrogen control system, detonable mixtures might form and detonate. This program will assess detonability as a function of temperature and hydrogen-steam concentrations typical of accident conditions in a large, dry PWR.

2. Technical data and information on hydrogen generation and control to help formulate the Commission policy on hydrogen regulations not covered by the existing rule (1986).
   Justification: This program will be assessing the threat posed by hydrogen from core-melt accidents more severe than those currently covered by the final hydrogen rule. This information will be used in PRA to determine if additional control and mitigation requirements are cost effective.

3. Specific data on hydrogen combustion phenomena for ice condenser and Mark III containments (1986).

†A modification of the regulations may result from the research.

Justification: In the licensing review of plants with ice condenser and suppression pool types of containments, a number of issues have been related to combustion phenomena such as flame acceleration and flame stability. This research and the research being conducted by EPRI on hydrogen safety will confirm regulatory assessments in these areas.

6.4.2 Research Program Description

As a consequence of an accident, significant quantities of hydrogen can be generated in the reactor vessel from steam-metal reactions and in the containment building from molten-core/concrete interactions. Burning of this hydrogen leads to pressure loading of the containment.

The hydrogen behavior program is developing multicompartment combustion, improved detonation, and diffusion flame models to predict thermal loads and containment pressure histories after hydrogen combustion. The models include heat transfer by radiation, convection to surfaces, and condensation and evaporation of sprays. Work on understanding the phenomena of flame acceleration and transition from deflagration to detonation in containment is being carried out, along with work on hydrogen stratification, mixing, and transport effects. The experimental portion of the program includes the determination of combustion and detonation limits in air and steam and the effect of the strength and location of the ignition source, geometry, and obstacles. Temperature and pressure profiles as a function of time will be measured. The effects of hydrogen burning on source term attenuation and composition will also be assessed.

A supporting program is studying the prevention and mitigation of hydrogen combustion by deliberate ignition of lean mixtures of hydrogen with catalytic igniters. Studies include the effects of sprays and aerosols on igniter performance under accident conditions.

The major research products will be:

1. Analysis of three to five specific plants for degraded core and severe accidents (1986). (Also applies to Need 2.)

2. a. Assessment of effects of aerosols on hydrogen control system and the effects of hydrogen burning on source term (1986).

   b. Preliminary hydrogen diffusion model and flame acceleration model (1986).

3. See RES Products 1 and 2.a and b.

6.4.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

High-temperature hydrogen steam mixtures may jet out of breaks or openings from the reactor's primary coolant system into the containment building. These jets may either ignite spontaneously or a flame may flash back from an ignition source in the containment to the point of release. The issue is: Under what conditions will autoignition of jet flames occur for release of a steam-hydrogen or hydrogen jet into an ambient mixture? The detailed diffusion flame modeling effort has been cancelled, thus eliminating the capability to provide a stand-

alone model to predict the local environment of safety-related equipment. In addition, flame acceleration modeling will be discontinued after 1986. This would have provided a useful predictive tool for assessing potential damage to equipment and structures cause by accelerated flames.

Also unfunded will be the program to assess the effects of aerosols on the hydrogen control system and the effects of hydrogen burning on the source term.

6.5 Fuel-Structure Interaction

Experimental research described in this element will obtain data on the consequences of ex-vessel interactions of high-temperature core fuel debris following escape from the vessel in severe accidents. The types of interactions of concern are thermal and chemical interactions between core fuel and (1) the reactor cavity concrete basemat, (2) water present at the time of fuel debris escape or subsequently introduced to the cavity, and (3) the containment atmosphere.

6.5.1 Major Regulatory Needs and Their Justification

Experimental research and analytical studies are needed to assess characteristic interaction responses for:

1. Heat generation and release for analysis of containment performance during severe accidents (1986).

2. Noncondensable gas and aerosol release for analysis of containment performance during severe accidents (1986).

3. Rapid steam generation with potential for containment failure (1987).

4. Rapid pressurization of the containment in high-pressure melt ejection with potential for early containment failure (1987).
   Justification of Above Needs: Current assessments of the margins of safety to containment failure under core-melt sequences have overlapping uncertainty bounds between load and response estimates. Thus, the primary need for this experimental research is to develop data upon which better quantitative assessments of the challenge to the containment structure can be made for postulated severe accidents involving release of core debris to the reactor containment. This applies to the four needs listed above. The sources for the containment challenge are not adequately known, and more research is needed to evaluate (1, 2, and 4) effects of the interaction between hot fuel debris and concrete basemat materials and between core debris and the containment atmosphere; (3) effects involving rapid steam generation; and (1-4) the quantification of parameters used to provide a basis for establishing and verifying analytical models used in severe accident assessment and accident management planning.

6.5.2 Research Program Description

The plan of research is to conduct small- and large-scale scoping and phenomenological tests of core-melt/concrete, core-melt atmosphere, and core-melt/coolant interactions in order to quantify analytical models for noncondensable gas and

aerosol sources and heating of the containment atmosphere. In addition, the effects of introducing coolant to the fuel-melt mass will be evaluated. Hot solid interaction tests will be performed to assess long-term cooling problems of solidified melts.

In-vessel steam explosions as well as nonexplosive rapid steam generation transients will be investigated experimentally in the dropping and reflood contact modes of interaction.

The major research products will be:

1. Verification of fuel-melt/concrete interaction models in CORCON to be obtained from large-scale test facilities (1986).

2. a. CORCON and CONTAIN code verification (1986).

   b. Experimental data for modeling the interaction of hot solidified melts with concrete and long-term cooling characteristics of solidified core melts (1986).

   c. Predictive mechanistic models for the explosive and nonexplosive interaction of thermite melts with water in dropping and reflood contact modes in the Fully Instrumented Test Series (FITS) facility and with uranium and zirconium mixtures in the Large Melt Facility (1986). (Also applies to Need 3.)

3. Experimental tests from high-pressure ejection tests to quantify:

   a. Dispersion of core debris out of reactor cavity (1986).

   b. Debris particle size distribution and aerosol generation (1986).

   c. Rapid pressurization of containment due to thermal and chemical interactions of dispersed core debris with containment atmosphere (1987).

4. For steam explosions, quantification of functional relationship between energy conversion ratio and mass of molten material or alternatively of the fraction of mass of molten material that participates in interaction (1987).

6.6 Containment Analysis

This element will provide analytical tools for the assessment of the challenge to the reactor containment system from postulated severe accidents. The types of challenges produced by postulated accidents may consist of overpressure from steam generation and noncondensable gases, fission products, and aerosol releases, hydrogen burns, fuel-structure interactions in the reactor vessel cavity, and direct heating of the containment atmosphere by the core debris discharged from the reactor vessel under high pressure. The results of these analyses will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.6.1 Major Regulatory Need and Its Justification

The CONTAIN code, a comprehensive, integrated systems code, is under development to meet the need for an analytical tool for assessing the abnormal loading imposed on containment systems by severe accident conditions (1989). The code is structured to be readily modified to reflect new data and models as they are developed. Experimental programs are under way to compile data for code validation and improved model development.
Justification: The containment analysis task is the key to integrating the research on severe accident phenomenology into a tool that can be used for regulatory audits of containments and that can be used to provide an improved basis for risk analysis. Decisions on future regulatory actions dealing with severe accidents will require a state-of-the-art capability for analyzing containment performance.

6.6.2 Research Program Description

This research undertakes to develop a computer code capable of simultaneously assessing the many-faceted challenges to containment during severe accidents. Incorporation of phenomenological models for the many simultaneous processes occurring within the containment system into a single code structure is being pursued. However, where more manageable, data interfacing with existing codes or codes separately developed by modeling specific phenomena of concern will be adopted to provide integrated analyses of the total containment challenge. The analytical package will be verified, using experimental data to be obtained from tests on core-melt/concrete interactions, high-pressure melt ejection, hydrogen burning, and rapid steam generation from which aerosol source term information will also be generated.

The major research products will be:

1. a. CONTAIN code verification (1986).

   b. Integration interface between CONTAIN code and other codes such as TRAP-MELT for fission product and energy source data (1987).

   c. Integration interface between CONTAIN and fission product dispersion code for computing offsite doses (1986).

   d. CONTAIN code maintenance, user supports, and model improvements (1986-1990).

6.7 Fission Product Release and Transport

The fission product release and transport research program is directed at developing models and obtaining experimental data (to support development and assessment of these models) to determine the potential radiological source term released from LWR plants during severe accidents. The findings from this research program will be reviewed for usefulness in improving the response capability of the NRC Operations Center. This research includes studies on radionuclide release from the fuel, on transport and attenuation of radionuclides within the reactor coolant system, and on attenuation (depletion) within the containment vessel.

6.7.1 Major Regulatory Needs and Their Justifications

1. Determination of amounts, chemical forms, and aerosol characteristics of fission products released in-vessel and ex-vessel and removed in the reactor coolant system and containment under severe accident environments, for use in confirming regulatory decisions concerning conditions beyond the design basis (1987).
   Justification: In-pile measurement of the fission product release rates from prototypic fuel will be used to benchmark out-of-pile separate effects tests on fission product release from fuel under severe accident conditions. Given such information, the NRC will be able to answer more precisely the question of the amount of the fission products that exist in the containment.

2. Improvements in PRA consequence calculational methods (1986).
   Justification: The present state of risk assessment techniques suffers from limitations both in the methodology and in the incompleteness of the phenomenological data base. A better understanding of the phenomenology of accidents involving fuel damage and the corresponding fission product release will permit realistic treatment and should much improve the usefulness of risk assessment.

3. Safety goal evaluation and emergency preparedness (1986).
   Justification: Nuclear power plant emergency preparedness requirements have the potential of alleviating some consequences of the impact upon the public of accidental releases of radionuclides. Modification of the radiological source released from the plant during severe accidents could have some impact upon the bases for any new requirements.

6.7.2 Research Program Description

The strategy of the fission product release and transport program is to develop an experimental data base, scientific information, and computer codes for predicting with reduced uncertainty release, chemical reaction, deposition, and transport behavior of radionuclides under severe LWR accident conditions. This will allow the technical bases for licensing, inspection, and regulatory practices to be substantiated. Computer models are being developed and validated to assess fission product and aerosol release from the fuel during the in-vessel heatup and melting phase and during ex-vessel interactions of fuel debris with reactor cavity materials (e.g., concrete). Models are being validated and improved to assess the transport, reaction, and deposition of radionuclides within the reactor coolant system components and piping and within the main containment compartment(s). Models for quantifying the effectiveness of engineered safety features (ESFs) in mitigating fission products under severe accident (fission product and aerosol) loadings and environmental conditions are discussed in Section 6.9.

Laboratory-scale separate effects experiments are under way to provide data for model development and validation in a number of areas, including fission product release from fuel and from core-concrete interactions, fission product chemistry, fission product interactions with prototypic surface materials, and fission product aerosol behavior in the containment. Data on the behavior of radionuclides in the lanthanide and actinide series are of particular importance.

Multiple effects experiments and in-pile separate effects experiments are being conducted to assess the validity of the computer models. These experiments include the small-scale and large-scale in-pile fission product release experiments to be carried out in the ACRR and NRU, respectively, and the large-scale fission product and aerosol transport tests conducted at the Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL), and Marviken facility in Sweden.

Using the models developed within this element, periodic analyses will be conducted to refine best-estimate, release-from-plant, radionuclide source terms for severe LWR accident sequences. A major source term assessment was completed in 1985 (NUREG-0956).

The major research products will be:

1. a. Data report for developing and improving in-vessel fission product release models (FASTGRASS, VICTORIA, and CORSOR) from out-of-pile tests up to 2800K (1986).

   b. Data report for benchmarking models for in-vessel release of fission products from in-pile experiments (1986).

2. Large-scale fission product and aerosol transport test results to be used to assess TRAP-MELT code, VANESA code, and CONTAIN code (1986).

3. Severe accident source term uncertainty report (1986).

6.7.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

Models and codes for high burnup (>60,000 MWD/MTM) fuel or for recycle (MOx) fuel need to be validated.

6.8 Containment Failure Mode

This element treats three possible failure modes: valve failure, materials failure in electrical penetrations due to high temperature, and mechanical failure of the containment due to either excessive local deformation at major penetrations or structural failure. Both assessments of the risk posed by loads outside the design basis, such as hydrogen burns or basemat melting, and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment could fail. However, this element does not address the failure mode arising from the failure to isolate the containment because of improper valve positioning. Both the utilities and the NRC address this part of the problem through quality assurance practices, inspection and enforcement, and other administrative and management techniques. Scenarios that bypass containment via penetrations are also treated in Sections 6.2 and 6.11. The findings of the containment failure mode research program will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.8.1 Major Regulatory Needs and Their Justifications

1. The capability to predict, with a high degree of confidence, the pressure and temperature environment that can be sustained by any of the great variety of containment structure designs before the rate of leakage becomes unacceptably high (1988).
   Justification: State-of-the-art methods cannot reliably predict whether leakage will begin around penetrations or in the membrane region of the shell. If leakage at penetrations is critical, the effects of aging on gasket performance will be of significance. The technical problems involve developing an ability to predict deformations for the wide array of containment types and relating deformations of containment structures to leak behavior. In addition to failures of the shell structure or penetrations, possible leakage paths exist through isolation valves and electrical penetrations. The staff must have the capability to evaluate the leak rate estimates made of the capabilities of a wide range of containment designs.

2. The development of simplified computational models, suitable for use in risk analyses, that adequately represent the variability of containment performance under severe loadings (1987).
   Justification: The impleme tation of a safety goal would require, as part of the PRA calculations, computational models describing the performance of containments. In particular, the implementation of a containment performance criterion, i.e., conditional probability that a containment will function given an accident, would require an ability to relate variability in leakage behavior to variability in structural parameters and accident conditions.

3. An ability to assess the extent to which containment performance may be degraded in accidents initiated by extreme external events such as a major earthquake (1990).
   Justification: The first generation of PRAs for nuclear power plants indicates that severe environmental events are likely initiators of severe accidents. The current practice is, effectively, to assume that containment performance is the same, whatever the sequence of events leading up to a severe accident. More realistic models will require that the effects of the initiating event, as well as those of the ensuing accident, be considered.

6.8.2 Research Program Description

The research effort will focus on five areas:

1. Model tests of containment structures aimed at verifying computational methods for predicting deformations and failure.

2. Experiments to understand the behavior of seals and gaskets when subjected to severe accident conditions.

3. Experiments on models of penetrations to relate leakage behavior to local deformations and pressure-temperature environments.

4. Experiments on leakage behavior of electrical penetrations in high-temperature environments.

5. Experiments on the performance of containment valves when subjected to severe accident loadings.

There is, and will continue to be, significant interaction with other NRC-sponsored programs related to the severe accident research program. Particularly close coordination will be maintained with the programs on hydrogen generation and control, fuel-structure interaction, and containment analysis (see Sections 6.4, 6.5, and 6.6). In addition, there will be interactions with the risk code development program (see Section 6.10). There will also be interaction with other national and international programs. Contributions to this program from EPRI-sponsored work are anticipated in the way of analytical predictions of capacity to be compared against test results and in the localized behavior of concrete containments under severe accident loadings.

Two foreign programs have been identified as sources of information. One is the effort on prestressed concrete containments being conducted in France. The other is the planned testing, on a shake table in Japan, of containment models to simulate seismic response.

Experiments involving tests of steel containment models under static pressure will be completed in 1985. Reinforced concrete models will be tested in 1986.

The tests to understand the behavior of seals and gaskets under simulated accident pressure-temperature profiles are being performed in the 1984-1985 period. The shell-penetration interaction effects will be studied in steel model tests and concrete model tests. Other large-scale testing of penetrations (containment airlocks, equipment hatch, BWR containment head, etc.) will be planned in 1985 and performed in 1986-1987.

Planning for simulated seismic testing of containment models will begin in 1985. The actual testing depends, in great measure, on the extent of cooperation developed with the Japanese research program on seismic testing. The three options currently under consideration are cooperative testing using the large Japanese shake table facility at the Nuclear Power Engineering Test Center in Japan; simulation of earthquake ground motion by phased explosive arrays; and quasi-dynamic loading using hydraulic actuators. The first seismic tests are anticipated in 1988.

Valve performance tests and tests on electrical penetrations have been or will be performed in 1984 and 1985.

The major research products will be:

1. a. Results of experiments on valve performance and electrical penetrations (1986).

   b. Results of final test series on seals and gaskets of major penetrations (1986).

c. Results of tests on models of major penetrations (1986-1987).

d. Comparison of predicted capacities for prestressed and reinforced concrete containments with experiments under static pressure (1987).

2. a. Comparison of predictions of steel containment capacity under dynamic pressure loads with experimental results (1986).

b. Comparison of predictions of capacity for reinforced and prestressed concrete containments under dynamic pressure loads with experimental results (1987).

3. Results of initial tests of containment models under simulated seismic loading (1988).

6.9 Fission Product Control

The fission product control program was developed to evaluate the effectiveness of engineered-safety-feature (ESF) systems under severe conditions as a part of broad research needs to support the reassessment of the regulatory assumptions of severe accidents and to develop a technical basis for decisions concerning the response of ESF systems to the source term reassessment and to severe accidents. The findings will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.9.1 Major Regulatory Need and Its Justification

Validation of the effectiveness of ESF systems under realistic estimate of revised fission product source terms to provide input for (1) severe accident policy and regulation questions and (2) source term reassessment for basic LWRs (1987).
Justification: Evaluation of the impacts of revised source terms on the design and effectiveness of ESF systems for a spectrum of accident conditions is needed as part of the information base for formulating policies and strategies to mitigate the postulated fission product loadings in a severe accident.

A concern arising from past regulatory emphasis on emitted radioactive iodine is that this practice may have resulted in a misplaced emphasis on ESF-system design. A review of mitigative ESF systems used in current LWR designs shows that the combination of ESF systems used in contemporary power reactors results in effective mitigation of all currently postulated accident sequences within the design basis accident (DBA) envelope. For the DBA, conservatisms exist in the form of simplifying assumptions and underestimates of some of ESF-system effectiveness. Most ESF systems are likely to be functional for postulated accidents substantially more severe than the DBA. There is, however, substantial variation in the effectiveness of fission product removal of various ESF systems under conditions exceeding their design basis.

The results of this research are expected to produce significant new information that will permit evaluating ESF-system design and effectiveness for the full spectrum of accidents and are therefore expected to contribute to future regulatory decisions.

6.9.2 Research Program Description

The strategy for the research is to obtain and develop technical information that will assist in providing best estimates of the spectrum of chemical and physical properties of the severe environments expected to be imposed on the ESF systems and to evaluate and predict ESF-system performance under such conditions.

The research is closely coordinated with the other NRC severe accident programs as well as with those conducted by other foreign countries and the United States nuclear industry (EPRI, General Electric, Westinghouse). The existing and expected research results are and will be extensively used to achieve the objectives of this program.

Research will concentrate on the prediction of the extent of the removal effectiveness and the depletion of aerosols and other fission products by ESF systems, such as containment sprays, suppression pools, ice beds, and filter systems, on the quantification of the effectiveness of ESF and other mitigation features in reducing the potential fission product escape from containment, on the evaluation of hydrogen burning on the performance and the effectiveness of ESF and aerosol concentrations under such conditions, on an evaluation of the existing design features under expected aerosol loadings, and on the development of simulated conditions and design and operational features of ESF for a generic evaluation for standardized nuclear facilities.

Codes will be developed and verified for ESF-system reliability, accelerated aging, and evaluation of safety/technical benefits as well as cost benefit for alternatives to some of the existing ESF systems.

The major research products will be:

1. a. Verification based on experimental data to be provided by EPRI (1986).

   b. Code (ICEDF) verification for evaluation of PWR ice-condenser effectiveness (1986).

   c. Code development and verification for performance and effectiveness of PWR/BWR filtration systems under predicted aerosol loadings (in-containment systems, auxiliary buildings, fuel handling buildings, standby gas treatment systems, and double containment annulus venting; 1986). Evaluation of alternatives to those systems, including technical/cost-benefit analysis (1986).

   d. Code modification and verification for performance and effectiveness of PWR/BWR containment sprays under severe accident conditions (1987).

   e. Evaluation of existing PWR/BWR ESF-system design (1986).

   f. Code development and verification for evaluation of generic design of ESF systems for standardized nuclear facilities (1987-1989).

g. ESF-reliability-testing code development and verification (1987-1988).

h. Code development and verification for aging aspects of selected generic ESF systems (1987-1988).

6.10 Risk Code Development

The risk code development work described in this section has as its purpose the periodic improvement of the present set of computer codes used in analyzing severe accident physical processes for PRA. These risk codes are distinguished from codes discussed in other sections by their simplistic, faster-running, and more integral character. Such characteristics are necessary for PRA because of the need to perform analyses of many accident sequences from initiation to final environmental effects. These characteristics also lead to the use of risk codes in regulatory areas where such broad accident perspectives are important. For severe accident regulatory considerations, these codes are used directly to produce the bottom-line technical products, i.e., the risk estimate and value/impact assessments discussed in Sections 6.11 and 6.12. Risk code development will be completed in 1986. Activities will concentrate on maintenance of the code and on applications to regulatory issues. Some of the effort to validate the risk code (including comparison to mechanistic code predictions) is addressed in Sections 6.2 and 6.3.

6.10.1 Major Regulatory Need and Its Justification

Development for regulatory use of a risk code more readily understandable and amenable to modification, to include the capability to assess the uncertainty associated with risk estimates (1986).
Justification: Item II.B.8 of the TMI Action Plan (NUREG-0660) discusses NRC efforts concerning a long-term rulemaking on the need to consider severe accidents in the regulatory process. The analyses discussed in Sections 6.11 and 6.12 are intended to provide the technical data needed to address this action plan item. The risk codes described in this section are the principal codes to be used in the analyses of Sections 6.11 and 6.12. Correction of known deficiencies in these codes is necessary prior to their use in this context. An evaluation of the uncertainties associated with the results of calculations made with these codes is necessary for improved decisionmaking using risk perspectives. The work of this section will provide the required code modifications.

6.10.2 Research Program Description

This element relates to the development of computer codes for use in PRA to analyze the phenomenological processes associated with severe accidents. Because of the need in PRA studies for the analysis of many accident sequences, these codes are to be relatively simple and fast-running. They will thus be the more approximate and quick counterparts to the more mechanistic codes being developed in parallel in other research elements and will provide the means by which the detailed analytical and experimental program results can be reflected in risk studies.

The code development work in this element is being undertaken in order to correct identified deficiencies in existing risk codes (MARCH, CORRAL/MATADOR, and CRAC). These deficiencies relate both to the modeling of physical processes within the codes and to the actual structure of the code.

The MELCOR development program is intended to replace the MARCH, CORRAL/MATADOR, and CRAC computer codes for use in risk studies.* One fundamental characteristic of this code is that it is to be developed using a "data management system" and a modular structure. Unification of the subject areas of the present three codes under MELCOR is being undertaken to permit direct assessment of the entire course of a severe accident, a feature particularly important to uncertainty analyses.

The major research product will be:

- MELCOR code documented for users such as the Severe Accident Research Program (SARP) and the Risk Methodology Integration and Evaluation Program (RMIEP) (1986).

6.11 Accident Consequence and Risk Reevaluation

In this section, the risk codes discussed in Section 6.10 are being applied in concert with products of other sections (e.g., Sections 6.1 and 6.8) to produce up-to-date assessments of the consequences and risk of severe accidents in LWRs. The findings will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.11.1 Major Regulatory Need and Its Justification

Up-to-date analyses of the predicted consequences and risk from severe accidents in LWRs using the revised source terms associated with such accidents (1986-1988).
Justification: The consequence and risk reevaluations described in this section provide a baseline level of risk from which severe accident decisions and risk-reduction analyses (see Section 6.12) can be performed. They require use of improved understanding of the phenomenology of severe accidents, including results of updated estimates of the associated source terms.

6.11.2 Research Program Description

The research to be conducted under this element relates to the application of advanced versions of risk codes to the reanalysis of the consequences of important accident sequences. That is, as the severe accident physical process risk codes are developed (as discussed in Section 6.10) and improved source term analyses are completed, they will be put to use to reanalyze the consequences of accident sequences determined to be important in previous risk studies. Further, as these consequence analyses of specific sequences are

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*The MARCH code analyzes inplant accident thermal hydraulics, the CORRAL code analyzes in-plant radionuclide transport behavior, and the CRAC code is used to analyze ex-plant radionuclide dispersion and resulting effects (e.g., property damage, health effects). In the future, it is planned that the MELCOR code will replace these three codes.

completed, they are to be convoluted with their associated sequence likelihood, thereby providing a redefinition of the risk of studied plants. In this way, previously completed risk studies can be periodically updated to reflect the latest advances in accident likelihood and consequence analysis.

The major research product will be:

- o Consequence and risk evaluations performed iteratively at roughly 1-year intervals (1986-1988).

6.12 Risk Reduction and Cost Analysis

In this element, methods are being developed and analyses performed to permit more systematic evaluations to be made of the cost effectiveness of current or proposed regulatory requirements, alternative concepts for reactor design and operation, and decisions on backfitting.

6.12.1 Regulatory Needs and Their Justification

1. Methods to identify, on a generic basis, the potential for risk reduction of alternative design concepts for all classes of power reactors to support severe accident decisions (1986).

2. Improved estimates of the costs associated with adopting alternative designs, safety features, and operating procedures on both existing plants and plants still in the design stage, to support decisions on backfitting safety features to existing plants and to evaluate and implement the benefit-cost guidelines of the proposed Commission Safety Goals (1986).

3. Documentation of uniform procedures for carrying out improved value/impact analyses for use by both NRC and the industry (1986).
   Justification of Above Needs: NRC must decide whether to require augmented safety features on those plants being reviewed under the ISAP or as a result of severe accident policy decisions and how the benefit-cost guidelines are to be applied in future licensing actions. A prerequisite for such decisions is the availability of reliable methods that will permit the staff to evaluate and compare the risk-reduction potential and costs of practical options for preventing and mitigating the effects of severe accidents. Methods and procedures must be developed and documented to ensure that these options can be comparatively evaluated and that such evaluations are made on a consistent basis.

6.12.2 Research Program Description

This program will establish standardized methods for the analysis of the risk-reduction effectiveness of generic safety system modifications to support decisions on major proposed regulatory requirements. In the longer term, the risk-reduction research program will apply these evaluation methods to the review of a few standard plant designs in order to establish the feasibility of applying benefit-cost guidelines to new CP applications. Follow-on work will apply the results of these studies to a systematic review of the regulations dealing with reactor safety.

The major research products will be:

1. Generic assessment of costs and risk-reduction potential of alternative safety features applicable to specific classes of LWRs (1986-1987).

2. Evaluation of feasibility of using PRA to improve reliability of existing plant systems (1986). (Also applies to Need 1.)

3. Establishment of standardized procedures for the application of PRA results and value/impact analyses to the decisionmaking process for proposed rules, guides, etc. (1986).

7

7. RADIATION PROTECTION AND HEALTH EFFECTS

The goal of radiation protection is to ensure that any individual and societal risks of health damage resulting from licensed activities are not unacceptably high and are as low as is reasonably achievable (ALARA) in light of available technologies, the dollar costs of reducing risks, and other considerations in the public interest.

Achieving this goal requires a technical and scientific data base that permits accurate identification and measurement of sources of radiation exposure and clearly defines the relationship between exposure and potential health effects, thus allowing issuance of radiation exposure limits with a sound scientific basis for use in the work place and in the general environment. Insufficient information in these areas creates uncertainties, resulting in public health protection and safety regulatory policies that can be either overly protective and therefore uneconomical or too lax, thereby exposing workers and the public to unacceptable risks. Significant uncertainties remain in some areas of radionuclide metabolism and internal dosimetry, in the characteristics of dose-effect relationships, and in radiological dose measurements in the work place.

The radiation protection and health effects research program is limited to studies that have immediate application to regulatory issues. The annual health effects research budget of the NRC constitutes only about 3 percent of the total Federal research budget devoted to addressing questions on the biological effects of ionizing radiation. This means that the NRC relies on the larger biomedical research programs of DOE and the National Institutes of Health (NIH) and on other national and international sources of information to provide most of the necessary scientific base for its use. NRC research is done on relatively narrow and specific issues directly related to NRC programs.

The NRC is an active participant on the Committee on Interagency Radiation Research and Policy Coordination (CIRRPC), which was formed in April 1984. This committee replaces both the Interagency Radiation Research Committee and the Radiation Policy Council. The CIRRPC and its associated Science Panel provide for coordination of Federal research on radiation and uniformity in Federal radiation protection policies.

Discussed in this chapter are research plans in the areas of metabolism and internal dosimetry, health risk estimation, and occupational radiation protection.

7.1 Metabolism and Internal Dosimetry

The radiation dose to the human body following the intake of radionuclides through inhalation, ingestion, or absorption through the skin and wounds will depend on the physiological and metabolic processes that determine the distribution and retention of radionuclides in various body organs and on the type, energy, and spatial distribution of the radiation emitted from these internally deposited radionuclides.

Results from these investigations provide the basic data found in national and international radiation protection publications. These data serve as references for Commission licensing decisions and regulations. They also serve as the means by which estimates of doses received are made from bioassay data, which are vital to determining doses for an individual worker and thus confirming compliance with NRC standards.

7.1.1 Major Regulatory Needs and Their Justifications

1. Reduction of the uncertainties in data on metabolic behavior of reactor fuel materials, to be the basis for regulatory guides on bioassay and dose estimation (1987).
   Justification: Workers, especially those at uranium mills and fuel fabrication plants, are exposed to internal radiation doses from materials generated during the production of reactor fuel that contain naturally occurring uranium, thorium, and their decay products. Unlike the case for some fission products such as strontium-90, cesium-137, or iodine-131 and some activation products such as cobalt-60 or iron-59, the metabolism of and internal dose from these materials are not well understood. The amount of radioactive material deposited and retained in the lung after an intake by inhalation is very dependent on the chemical and physical state of the inhaled aerosol. This, in turn, changes during the milling and processing of ores and the fabrication of fuel. Better definition of the metabolic and dosimetric characteristics of these components will improve NRC dose and risk assessments and therefore the protection provided for workers exposed to these naturally occurring radionuclides.

2. Reduction of the uncertainties in data on metabolic behavior of transuranic elements, to be the basis for regulatory guides and amendments to 10 CFR Part 20 (1987).†
   Justification: Recent studies on the gastrointestinal absorption of plutonium suggest problems with the transfer factors on which present standards are based. Because of differences in physical and chemical states, industrially produced materials are likely to display different properties than the laboratory materials used most often in research and upon which current standards for protection are based. These uncertainties must be resolved in order to ensure the validity of present radiation protection standards for transuranics. This has application to the management of nuclear waste and the potential future licensing of reactors using mixed oxides or plutonium fuels.

3. Validation of methods for calculating internal doses used in implementing recommendations of the International Commission on Radiological Protection (ICRP) (1989).
   Justification: The ICRP-26 dose limitation system based on health risk necessitates practical calculational methods for internal and external doses. The current data published by the ICRP for this purpose applies only to workers (adults). In order to more precisely estimate the dose and health risks to the general population as a result of effluents from NRC-licensed facilities, data on the principal radionuclides are required for a number of age groups.

†A modification of the regulations may result from the research.

7.1.2 Research Program Description

The NRC research effort relies on the data generated by the more extensive programs supported by DOE to the maximum extent; NRC funds only those research projects with objectives that are specific to its regulatory needs and not otherwise addressed.

The NRC research program will develop metabolic models or provide metabolic distribution and retention values for reactor fuel materials and for the transuranic elements. This is accomplished by administering the materials to animals and measuring the amounts concentrated in various organs or excreted. Information on metabolism may be obtained by administering known amounts of materials to animals and measuring the amounts concentrated in various organs or excreted, by external counting of persons exposed to these radionuclides, and, when possible, by radiochemical assay of their tissues at autopsy. Age- and sex-dependent dose conversion factors will be determined, enabling more realistic dose estimates for infants and children as well as accommodations for sex differences.

The major research products will be:

1. Metabolic model for inhaled yellowcake (1987).

2. Values for the gastrointestinal absorption factor for the actinides (1986-1987).

3. a. Production of a modified internal dosimetry code (1988).

   b. Compilation of age- and sex-specific dose conversion factors (1988).

7.2 Health Effects and Risk Estimation

Appropriate radiation protection measures depend on our ability to identify and estimate the adverse health effects resulting from exposure to ionizing radiation (both external and internal) and to quantify the health risks related to the use or release of radiation and radioactive materials. Such quantitative relationships are estimated from experimental and epidemiological studies and analyses designed to assess the dose-effect relationships (risk coefficients) for types of radiation and levels of exposure that might be encountered in the work place, in the environment, and following potential major accidents. This research effort provides basic data, improved methods of data analysis, and models for predicting health effects. These results strengthen the confidence in health risk assessment and thereby improve the scientific bases for rules and licensing decisions, including those underlying safety goals.

7.2.1 Major Regulatory Needs and Their Justifications

1. Reduction of uncertainties in health risk from exposure to low levels of low-LET (linear energy transfer) radiation (i.e., gamma and beta rays), to be used in the resolution of the general and basic question of what degree of control constitutes adequate protection of public health and safety (1989).

Justification: Current NRC standards are based for the most part on the cautious acceptance of a direct proportionality between absorbed dose and health damage that has no threshold dose for damage and is dose rate independent. This dose response model, when generally used, probably overestimates the risk for low-LET radiation at low levels; low-LET radiation predominates in the work place and in the environment. Quantitative determination of such an overestimation can have a major regulatory implication on risk management programs, especially ALARA programs that depend solely on tradeoffs between individual and collective doses and cost of controls. Because this relationship between dose and effect is so dependent on the outcome of the dose reassessment of the exposure estimates of Hiroshima and Nagasaki atomic bomb survivors, it will be necessary to reexamine NRC's basic acceptance of this direct proportionality and change its risk management programs accordingly.

2. Reduction of uncertainties in health risk from exposure to low levels of high-LET radiation (i.e., neutron and alpha particles), to be used in the resolution of the general and basic question of what degree of control constitutes adequate protection of public health and safety (1986).
Justification: Considerable uncertainty exists regarding the relative carcinogenic and mutagenic potential (i.e., quality factor) of low levels of neutrons that might be encountered in nuclear power plants and other NRC-licensed facilities. This uncertainty is presently even more pronounced because of the likelihood of significant changes in the estimates of neutron exposure to Hiroshima atomic bomb survivors.

Resolution of these uncertainties will have considerable impact on NRC regulations regarding allowable neutron exposures of workers at nuclear power plants and on the management requirements for tailing piles in controlling emissions from NRC-licensed uranium mills. Quality factors assumed for neutrons and alpha particles (presently 10x and 20x, respectively) affect these regulations and management requirements. Considerable uncertainty exists regarding the biological effects of internally deposited alpha emitters. The biological effects of radon decay products at levels encountered in the vicinity of tailing piles are being estimated from data gathered on uranium miners. The applicability of these risk estimates to the general public is highly questionable, particularly in the case of estimating the risk of radiation-induced lung cancer for a nonsmoking population.

3. Models and parameter values for predicting early health effects of inhaled radionuclides from major accidents, to be used in risk assessment (1986).
Justification: According to accident risk assessment studies performed by or for the NRC (e.g., Reactor Safety Study), a potential exists for large atmospheric releases of radioactive materials. Quantitative prediction of consequences of major accidents in terms of early mortality and morbidity resulting from inhalation of radioactive materials that can be accompanied by large (but sublethal) external radiation is currently based on very limited data. This contributes considerably to overall uncertainties in accident risk assessments. Better estimates of early health effects will improve emergency planning and preparedness, and a more realistic assessment of consequences will permit better definition and setting of priorities for reactor safety requirements.

4. Implementation of ICRP 26/30 system, requiring development of regulatory guides to explain to licensees how their radiation protection practices can be in compliance with NRC regulations (1989).
   Justification: Although the ICRP system relied upon data from numerous epidemiological studies, it would be prudent to follow the patterns of mortality demonstrated by nuclear power plant workers. This would require a feasibility study for establishing a nuclear worker registry. Finally, it would be desirable to have a method for comparing the risks from nuclear power with those from other fuel cycles.

7.2.2 Research Program Description

The strategy in the NRC health effects research program is to limit its support to studies that have direct application to regulatory requirements or questions and to rely heavily on the larger biomedical research programs of NIH and DOE and on other national and international sources for the extensive scientific base necessary.

An evaluation of the relative merits of recently developed statistical procedures for dose-response modeling will be undertaken (FY 1986). These procedures are presently being applied to estimate dose-response relationships and risk coefficients for radiation-exposed populations such as the Hiroshima and Nagasaki survivors and the U.S. uranium miners. Since the results of these investigations could well have an impact on NRC's radiation protection standards, it is necessary to know the strengths and weaknesses of the methodologies applied.

An epidemiological study of inhaled radon decay products and associated lung cancer risk in radium dial painters was initiated in FY 1984. This study will provide information on the contribution of confounding variables (e.g., smoking, mineral fiber exposure) to the risk of radon-induced lung cancer, thereby allowing more precise extrapolation to risk in the general public. They will remove existing uncertainties in our basic radon health effects data (uranium miners) and will allow development of a more logical radiation protection policy based on lung cancer/radon decay product exposure estimates for nonsmoking populations rather than on total population statistics.

Research to determine the quality factor for neutron exposures involves exposing mice to gamma rays and to fission neutrons at levels of exposure comparable to present occupational standards. Both life-shortening (with causative pathology) and genetic effects of exposures will be observed as end points of the effects of radiation.

The assessment of human risk from internally deposited alpha emitters, based on review of relevant epidemiological and radiobiological data, will begin in FY 1985.

To better define the mechanisms and the risks of acute injury from large atmospheric releases of radioactive materials under accident conditions, animals (rats and dogs) are being exposed to a variety of radionuclides representative of such releases, with and without external exposure. Mortality and morbidity patterns will be determined.

Regulatory guides for implementing the revision of 10 CFR Part 20, "Standards for Protection Against Radiation," will include those on worker exposure reports and on protection of the embryo/fetus. The feasibility of establishing a worker registry will involve assessing the information currently available in employee files. Comparative health risks will be determined by analyzing data on deaths and illnesses suffered by workers in the electricity-generating industry and by the general public exposed to effluents from the plants.

The major research products will be:

1. Improved statistical procedures that will find application in other hazard evaluation studies (1986).

2. a. Revised (or reaffirmed) values for neutron quality factor (1986).

   b. Improved estimate of dose-response function for radon daughter exposure/ lung cancer risk in the general public, including nonsmoking populations (1987).

   c. Risk coefficients for internally deposited alpha emitters (1987).

   d. Revised risk coefficients for low-level, low-LET radiation (1989).

3. a. Verification and improvement of models for early mortality resulting from radionuclide inhalation (1986).

   b. Development of models for morbidity resulting from radionuclide inhalation (1986).

4. a. Guidelines on establishing a worker registry (1989).

   b. Development of health risk coefficients for cost/benefit analyses (1989).

7.2.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

As a result of budget constraints in FY 1986, the studies involved in implementing 10 CFR Part 20 will be deferred until FY 1987. Consequently, the regulatory guides for radiation protection, the guidelines on establishing a nuclear worker registry, and the development of health risk coefficients for cost/benefit analyses will be deferred beyond FY 1989.

7.3 Occupational Radiation Protection

Research in the occupational radiation protection program is intended to provide information needed to help ensure an adequate degree of radiation protection for workers in NRC-licensed facilities and activities. Application of the results from this research through NRC regulations and guidance promotes consistency with national and international advances in radiation protection methodology.

7.3.1 Major Regulatory Needs and Their Justifications

1. Monitoring and evaluating the extensive research being conducted by the nuclear power industry and DOE on ALARA engineering technology to reduce

occupational exposure at nuclear power plants, monitoring applications of existing dose-reduction technology, and evaluating waste processing associated with this technology, so as to provide the basis for revising regulations or developing regulatory guides as warranted (1986-1989).
Justification: As nuclear power plants grow older, plant modification, inservice inspections, and major repairs tend to subject workers to higher radiation exposures. Effective and safe dose-reduction techniques and safe processing of the resultant wastes are needed to arrest this trend. Close, critical observance of industry decontaminations and DOE research programs by the NRC is necessary to obtain assurance that these activities and programs are adequately comprehensive and effective and to obtain the results from them on a timely basis for use in the NRC regulatory program as appropriate.

2. Improvements in occupational health physics technology to be the basis for implementing 10 CFR Part 20 and for developing regulatory guides as warranted (1986-1989).
Justification: Where the NRC has required measures to protect workers against radiation, further action should be carried out by the NRC to ensure that these measures are taken in a competent manner. Thus, standards of performance are needed, either as regulations or guides. Requirements and recommendations that would be practical and effective are sometimes not readily available so that supportive research is necessary.

3. Radiological health protection information extracted from inspection reports, preliminary notifications, overexposure reports, licensee employee complaints, applications for licenses and amendments, requests for major task approval, and LERs, to be collected at a central location and computerized within a specialized format permitting data retrieval and analysis within highly specific radiation protection subdisciplines (1986-1988).
Justification: The Office of Nuclear Reactor Regulation is responsible for evaluating on a continuous basis the performance of nuclear power plant licensees, including their occupational radiation protection programs, and for taking corrective action as needed. At present, the information available on a systematic basis is limited to occupational dose data (annual dose statistics by plant, collective dose data for broad task categories, termination dose reports for individual workers). Additional radiological health protection information is needed if specific problem areas and unfavorable trends are to be identified and corrected.

7.3.2 Research Program Description

The strategy in ALARA engineering technology is to closely monitor the extensive research being conducted by the nuclear power industry and the DOE, to perform a continuing analysis of these programs to provide information needed to meet NRC objectives, and to bring identified problems to the attention of appropriate personnel. For example, during FY 1983 and FY 1984, the funding level for such projects exceeded $33.5M.* The NRC staff, supported as

*"Radiation Dose Reduction--Working-Group Report," comment draft prepared by the DOE Dose Reduction Working Group, July 1, 1983.

necessary by contractors, will develop working relationships on a national and international basis with organizations that are funding and conducting research on dose reduction at LWRs through engineered methods and will acquire and maintain a high degree of familiarity with the details of these research projects, as well as DOE-funded projects, through personal contacts, site visits, meetings, and literature review. The staff will develop and implement procedures for keeping affected personnel in NRR, RES, IE, and the NRC Regional Offices informed regarding pertinent details of these research activities, including recommendations for any indicated changes in existing NRC regulations and regulatory guides. In addition, the NRC staff will perform a continuing analysis of this overall research program to identify dose-reduction opportunities that are receiving inadequate attention. Also, action will be taken to develop and implement information and methods for bringing identified problems to the attention of appropriate personnel in industry and DOE and to seek corrective action in an appropriate manner.

Regarding existing dose-reduction technology, in-plant observations and measurements are being conducted at nuclear power plants undergoing decontamination. These include evaluation of radiation fields, decontamination data, dose-rate reduction, worker exposure (including waste handling), and postdecontamination conditions.

If solidification of decontamination solution wastes leaves free water or has other waste-form problems or if interface problems occur between these wastes and their containers, handling, shipping, and disposal can be affected. The effect of chelating agents on solidification and containers is also a potential concern. For these reasons, the NRC needs to establish a position on the solidification processes and the containers used. Under a contract, laboratory tests on solidification processes and containers used with decontamination solutions are now being performed. This program will be completed in FY 1989.

The major research products will be:

1. a. Evaluation of processes for producing acceptable decontamination waste disposal forms (1989).

   b. Assessment of decontamination projects, including recontamination rates, at selected nuclear power plants with regard to overall effectiveness and safety (1988).

2. Data needed for modification of respirator protection factors as shown to be needed by field-testing with workers (1986).

3. Computerized system for accessing and analyzing radiological health protection information (1988).

7.3.3 Deferred/Unfunded Research Needs (as of [illegible] 85)

As a result of budgetary reductions in FY 1986, the program to evaluate processes for producing acceptable decontamination waste disposal forms will be unfunded.

8

8. WASTE MANAGEMENT

Regulation of radioactive waste management requires a technical capability to assess compliance of a waste management system with the regulatory requirements for operational safety, occupational radiological protection, and long-term waste isolation.

8.1 High-Level Waste

High-level-waste (HLW) management includes the regulation of operational safety, occupational radiological protection, and the long-term isolation of HLW. DOE has the responsibility to design, construct, and operate an HLW repository and to demonstrate that it complies with the standards and regulations established or to be established by the EPA (40 CFR Part 191) and the NRC (10 CFR Part 60) and in accordance with the Nuclear Waste Policy Act of 1982. Regulation of geologic disposal of HLW requires that NRC perform an independent assessment of DOE compliance sufficient to provide reasonable assurance of safety. Such an assessment must be based on a thorough understanding of the relevant phenomena and processes that affect the performance of a geologic repository both during and after waste emplacement operations. Effective regulation also requires providing timely guidance to DOE, especially in consultations during the time prior to submittal of a license application. As the results of NRC HLW research become available, they are provided to the licensing staff to aid in providing such guidance to DOE.

8.1.1 Major Regulatory Needs and Their Justifications

1. Evaluation of Releases to Accessible Environment--capability to evaluate DOE's safety analysis reports (SARs) to assess compliance with 10 CFR Part 60 and 40 CFR Part 191 (release to the accessible environment), to be used in reviewing DOE's license application (1988).
   Justification: Research into the physical phenomena relevant to repository performance, including investigation of the scientific and technical bases for the DOE performance assessment, is needed to enable NRC to evaluate DOE's demonstration of compliance with Part 60. Included is research into methods to allow the licensing process to deal with the uncertainties associated with predicting future changes to the site that might affect waste isolation.

   Part 60 requires that the SARs contain descriptions and analyses of the repository site's geologic barrier to radionuclide migration. The hydrology, geology, and geochemistry of the site will have to be analyzed. Research is needed so that NRC's licensing staff will understand how ground water transports radionuclides through the geologic barrier to the accessible environment and how geochemical effects can retard radionuclide migration.

2. Evaluation of Containment Requirement--capability to assess the DOE demonstration that HLW packages will comply with long-term radionuclide containment requirements defined in 10 CFR Part 60, as part of the assessment of DOE's license application (1988).

Justification: The potential hazard posed by HLW will last thousands of years. It is NRC's policy that containment of HLW must be substantially complete during the period when radiation and thermal conditions in the underground facility are dominated by fission product decay and that any release of radionuclides from the engineered barrier system will be a gradual process that results in small fractional releases to the geologic setting over long periods of time. (See performance objectives in 10 CFR Part 60.)

Research is needed to understand mechanisms of waste package degradation and failure and to identify the uncertainties associated in predicting both the behavior of waste form and package material in repository environments and the nature of those environments. The results of this research will facilitate an independent NRC assessment of the validity of the methods and tests used by DOE to predict long-term waste package performance.

3. Evaluation of Release Rate Requirement on Engineered Facility--capability to assess DOE's demonstration that the engineered facility and waste package will comply with the release rate criterion of 10 CFR Part 60 as part of the assessment of DOE's license application (1988).
Justification: 10 CFR Part 60 requires that following loss of containment the release rate of radionuclides from the underground facility must not exceed one part in 10^5 per year of the inventory present 1000 years after permanent closure. To perform an independent review of DOE's prediction of repository performance, NRC needs to know and understand the phenomena that control the rate of radionuclide release from the facility. Among the phenomena of concern are the impact of thermal perturbations on the geochemical environment, the very-near-field hydrological flow conditions, and the functioning of backfills to control influx and chemistry of ground water and sorption of radionuclides.

Research is needed to understand the chemical process by which radionuclides enter the ground-water system of a repository. Since there will be no opportunity to observe actual dissolution leaching of radionuclides from a waste form over very long periods, the results of short-term tests will have to be extrapolated. Further, the actual waste form will have undergone an aging process before the leaching process is expected to begin. As with waste package failure mechanisms, understanding of experimental and testing methods is needed to allow confident review of scaling to the long periods required for acceptable repository performance.

Of particular relevance to extrapolating laboratory experiments to the long periods of time required for repository performance is research on natural systems and materials exposed to conditions analogous to specific aspects of the repository environment for very long times--times on the order of those required for isolation of HLW. These "natural analogues" greatly increase our confidence that NRC licensing tools and DOE analytical and field studies will be adequate representations of expected repository performance.

8.1.2 Research Program Description

The strategy of the HLW research program is to identify and develop an understanding of basic phenomena and processes that determine the performance of an HLW geologic repository. Such identification and understanding form the technical bases for assessing DOE license submittals for construction and operation of HLW repositories so that the NRC can determine whether there is confidence that long-term performance objectives of 10 CFR Part 60 and 40 CFR Part 191 will be met. Coordinated program efforts of laboratory and field experimentation and theoretical studies will provide the identification and understanding of the processes and conditions that control the long-term performance of the system. Relevant research sponsored by DOE, EPRI, Department of Interior, Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD), and foreign governments is being factored into the planning of the NRC waste management research program in order to avoid unnecessary duplication and to maximize research effectiveness.

Essential elements in understanding the relevant phenomena are the identification and assessment of uncertainties pertaining to both the performance and assessment of performance of a geologic repository. Included are considerations of waste package and engineered system performance (e.g., overpack corrosion, failure of weldments, waste form leaching); geochemical interactions with respect to radionuclide migration and ground-water transport of radionuclides (e.g., chemical speciation, absorption, diffusion); effects on geological stability and isolation from excavation and impacts of emplaced wastes (e.g., thermal propagation of fractures, resaturation, performance of shaft seals); response of the engineered barrier system to changes in the repository environment (e.g., climatic changes, resaturation, wet-dry cycling, coupled interactions of thermal-mechanical-chemical-hydraulic systems); and models used to predict overall system performance. The HLW research program also addresses monitoring methods and instrumentation reliability.

Fundamental work in understanding important phenomena is often factored into the regulatory program through predictive models. An essential determinant of the usefulness of these models in the licensing process is their reliability as established by field validation work that both confirms and clearly defines limits of utility of these models. In addition to establishing the models of individual components and phenomena, a systematic and comprehensive integration of the understandings supporting the models must be carried out to ensure a uniform and complete approach to assessing repository performance. This "systems integration" is key to ensuring that RES resources are allocated according to level of uncertainty and importance of repository performance--there is no point to intense study of trivial problems. Finally, off-normal conditions that stress repository systems beyond their design need to be studied to probe the safety margin in the natural and engineered repository systems and ensure that the repository will perform acceptably even under excursions from expected repository conditions without application of undue conservatisms.

The major research products will be:

1. Evaluation of Releases to Accessible Environment

a. Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes (1987).

b. Report on field validation studies of radionuclide transport model (1989).

c. Parameters and processes important to evaluation of effectiveness of borehole plugging and sealing and shaft sealing techniques for salt (1987).

d. Parameters and processes important to evaluation of effects of resaturation in basalt and cyclical re-wetting in tuff on borehole and shaft seals (1989).

e. Report on integration of subsystem predictive knowledge into systematic repository evaluation capability (1990).

f. Assessment of effects of likely off-normal conditions that might result in degraded repository performance (1990).

2. Evaluation of Containment Requirement

a. Identification and description of the relationship among parameters important in container manufacturing and expected long-term container performance (1987).

b. Identification and description of methods for predicting long-term performance of waste packages in basalt and tuff (including waste form, container, and overpack) (1988).

c. Identification and description of waste package performance in the environment of a salt repository (1989).

3. Evaluation of Release Rate Requirement on Engineered Facility

a. Parameters and physical relationships important to methods for evaluating long-term performance of backfill systems proposed by DOE (1988).

b. Applicability of hydrothermal and geothermal data and predictive techniques to waste isolation performance assessments (1987).

c. Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes (1987).

d. Report on field validation studies of radionuclide transport model (1989).

8.1.3 <u>Deferred/Unfunded Research Needs (as of May 1, 1985)</u>

The waste management research effort in HLW is attempting to maintain progress in the three indicated areas: Evaluation of Releases to Accessible Environment;

Evaluation of Containment Requirement; and Evaluation of Release Rate Requirement on Engineered Facility. However, at present funding levels there are insufficient resources to adequately address all the technical issues sufficiently well to confidently support projected licensing schedules. Specifically, systems integration work has been removed from the list of projected accomplishments, and critical work on field validation of flow and transport models has been listed for funding in future years anticipating higher budget levels. The lack of timely results from field validation work could delay the licensing process. Additional topics that have been deferred include: waste package performance in salt; borehole and shaft sealing in basalt and tuff; cyclical re-wetting effects on radionuclide transport; and repository response to off-normal conditions. In each case, the availability of information in these areas is critical to the licensing activity. Inadequate funding levels may result in heightened uncertainty on critical licensing issues and subsequent significant licensing delays.

8.2 Low-Level Waste

Low-level-waste (LLW) management includes occupational radiation protection, protection of the general population from releases of radioactivity, and environmental protection. Land disposal of LLW must be dealt with in compliance with the NRC's 10 CFR Part 61. Congress has mandated (Public Law 96-573) that each State have the capability of disposing of LLW generated within that State by 1986. This has accelerated the selection of suitable sites. Most States are either writing compacts to serve regional LLW disposal needs or are planning in-State disposal. In some States, alternatives to shallow-land burial for LLW disposal are being investigated. The NRC will license disposal in non-Agreement States and will provide technical assistance to Agreement States. Disposal of low-level wastes licensed by Agreement States must be carried out in a manner compatible with NRC safety practice and requirements. In order to determine compliance with 10 CFR Part 61, and in particular to evaluate the long-term performance of specific disposal facilities, as well as to evaluate alternatives to shallow-land burial, the NRC must know and understand the phenomena and processes that affect the stability of disposal sites, the isolation and movement of radionuclides, and the associated uncertainties that enter into any demonstration that release criteria are met.

8.2.1 Major Regulatory Needs and Their Justifications

1. Methods and procedures for determining the long-term stability of packaged LLW and analyses of the long-term performance of disposed wastes, to be used in evaluating waste form and long-term performance (1987).
   Justification: Section 61.56 of 10 CFR Part 61 specifies a series of minimum requirements to ensure LLW stability. The regulation is written in general terms and does not provide detailed prescriptive requirements. Research is needed to determine that this requirement can be met over the design life of the stable waste packages. Also needed is research to provide additional guidance (1) to applicants on meeting the requirements of § 61.56, (2) to the staff on evaluating applicants' methods for ensuring waste stability, and (3) to better understand the long-term performance behavior of disposed wastes.

2. Capability to assess facility engineering for land disposal of LLW for determining compliance with § 61.51 of 10 CFR Part 61 (1987).
Justification: Investigation has shown that a significant path for water to enter trenches is through the trench cap. Further, trench cap subsidence has resulted from degradation and compaction of waste packages in addition to inadequate waste burial procedures and inadequate trench cap designs. Research is needed to evaluate the effectiveness of various facility designs to control water movement into waste trenches and to improve the stability of trench caps.

3. Capability to assess the effectiveness of monitoring plans for the preoperational, operational, and postoperational periods, including evaluation of the data, results, and conclusions from the monitoring programs, to be the basis for determining compliance with applicable regulations (1988).
Justification: Monitoring methods are needed to verify successful performance of shallow-land burial both during operations and following closure and to warn of incipient failure before radionuclides begin migration off site.

4. Capability to assess the licensee's demonstration that the concentrations of radioactive material that may be released from a land burial site to the general environment will meet the criteria in § 61.41 of 10 CFR Part 61, to be used in evaluating applications (1986).
Justification: In order to determine whether the criteria in § 61.41 are met, the NRC needs to understand the methods that can be used for predicting radionuclide transport, including how radionuclides become available for transport from the waste form or package. The NRC must understand the geochemical-hydrological interactions that control radionuclide transport in land burial sites. This requires investigation of coupled geochemical-hydrological radionuclide transport models that predict transport through both saturated and unsaturated geologic media.

5. Capability to assess alternatives to shallow-land burial of low-level wastes, including wastes that have higher concentrations than are acceptable for Class C wastes, to be the basis for rulemaking, developing regulatory guides, and evaluating applications (1988).†
Justification: 10 CFR Part 61 applies to land disposal of LLW. However, technical requirements now contained in Subpart D apply specifically to near-surface disposal. Disposal methods other than near-surface disposal, including aboveground vaults and mined cavity disposal, are likely to be presented as part of a license application. Research is needed to (1) develop performance assessment capability for licensing alternative methods that are different from near-surface disposal and (2) develop an understanding of the long-term performance of engineering materials, including interaction with the LLW that may be employed in a given alternative.

6. Criteria for characterizing sites for LLW disposal facilities to be used in developing programs for the acquisition of site-specific data for demonstration of compliance with NRC and EPA requirements (1990).

†A modification of the regulations may result from the research.

Justification: Acquisition of information on a potential LLW disposal site that is necessary in order to demonstrate compliance with NRC and EPA regulations can be complicated and costly. Criteria need to be developed to allow efficient collection and analyses of relevant data with a high degree of confidence that all licensing issues will have been given adequate consideration to allow firm and consistent licensing decisions to be made.

8.2.2 Research Program Description

The strategy for LLW-disposal research is to use field data and laboratory experiments to understand the phenomena that determine the performance of LLW land disposal facilities. This will be useful in guiding disposers of LLW, in assessing compliance with NRC requirements, and in evaluating the resulting level of protection achieved relative to public health and safety.

While the LLW research program is primarily directed toward near-surface disposal, i.e., to support the regulatory requirements of 10 CFR Part 61, it also includes research related to alternatives to shallow-land burial, including the disposal of wastes exceeding the limits for Class C low-level wastes. In addition, the program studies the problems identified through experience with existing LLW disposal facilities in order to evaluate and resolve the uncertainties important to licensing.

In particular, the LLW research program is developing information that can be used to understand the factors that influence long-term trench cap stability, water infiltration through trench caps, and waste form degradation (which can induce trench cap failure). It will identify means for minimizing uncertainties in predicting the release of radionuclides into the unrestricted environment. In addition, it is determining (1) the mechanisms that allow the release of radionuclides from the waste forms or waste packages, (2) the geochemical changes that occur when radioactive wastes interact with soils, (3) the radionuclides and their chemical forms that migrate through soils, and (4) the nonradiologic hazardous chemicals that are contained in or accompany low-level wastes. It will both test the chemical composition of wastes and develop data on materials that could be added to disposal trenches to fix or retard the movement of radionuclides. The research will develop and test geochemical/hydrological transport models for predicting water movement and radionuclide attenuation in this water for the various media through which it may pass.

A complement to the separate parts of this program will be an effort to bring these parts together and integrate them into a systematic approach to site and facility evaluation. This "systems integration" effort should begin in 1986 but has not yet been planned because funding projections do not indicate adequate support.

The major research products will be:

1. a. Evaluation of long-term performance of wastes and containers produced through currently available processing and containment technologies (1987).

   b. Specification of source terms from major LLW streams for assessment of site and facility performance (1988).

2. Assessment of means to enhance facility engineering performance in the control of water entry into below-grade burial trenches (1986).

3. a. Assessment of effectiveness of monitoring methods, particularly through the use of environmental methods applicable to the postclosure period (1988).

   b. Technical basis for LLW disposal site closure and monitoring criteria (1989).

4. a. Assessment of interaction of radionuclides with soils to predict LLW-disposal-facility performance (1986).

   b. Description of coupled geochemical/hydrological phenomena relevant to migration of radionuclides from shallow-land burial facilities (1986).

   c. Evaluation of properties of and release of chelating agents (organic complexants) from solidified decontamination wastes (1986).

5. Assessment of alternatives to shallow-land burial of LLW (1988).

6. Identification of data and information necessary to support definitive licensing decisions for an LLW site and facility (1989).

8.2.3 Deferred/Unfunded Research Needs (as of May 1, 1985)

The systems integration, site closure and monitoring criteria, and site characterization criteria will not be funded in FY 1986. Even with an anticipated increase in funding in FY 1987, the needed systems integration work is not projected to start sooner than FY 1988. Products for source term specification and examination of alternatives to shallow-land burial will be provided on a delayed schedule. Although States are actively seeking to solve their individual LLW problems, the idealized solution of a "half dozen" LLW sites seems to be disappearing amid general confusion and dissension among potential compact States. There may be a resultant proliferation of sites and designs for which the present research program is insufficiently funded to provide adequate and timely support to the licensing office.

A

Research Program	Research Elements	Typical Products
Chapter 1 **Operating Reactor Inspection, Management, and Repair**		
This research program will study time-related issues such as the mechanisms of aging and degradation, methods of examination and testing to determine the condition of components, and the interpretation of the results of these tests for appropriate action. This work will provide the bases by which the staff can assess with confidence industry test and examination methods and results. These assessments in turn provide bases for licensing decisions on whether operating plants continue to meet health and safety requirements in effect at the time of licensing and subsequently imposed health and safety requirements.	1.1 Reactor Vessels. This research element applies to the structural integrity of pressure vessels especially as affected by irradiation embrittlement and growth of postulated cracks in service.	- Validated model for predicting the potential for initiation, propagation, and arrest of flaws under scenarios involving pressurized overcooling. - Basis for fracture toughness requirements under conditions of thermal shock to reactor vessels. - Standard procedures for neutron dosimetry for surveillance irradiation. - Technical basis for proposed revisions to the environmentally assisted fatigue crack growth curves. - Validated methodology for recovering fracture toughness properties of reactor vessel steel by in situ annealing.
	1.2 Steam Generators. This research element deals with corrosion, cracking, and degradation of steam generator tubing during service; integrity of tubing as degraded by the water and stress environment during normal operation and upset conditions; and integrity of tubing over the long term as affected by decontamination and tube bundle cleaning and by other causes.	- Results of burst and leak rate testing of tubes removed from steam generator. - Validation of results from nondestructive examination through examination of removed tubes. - Correlation of remaining tube integrity obtained from burst and leak tests with results of nondestructive examination. - Evaluation and recommendations on possible tube vibration and damage during operation after chemical cleaning.
	1.3 Piping. This research element applies to the structural integrity of piping degraded during service by the water, stress, and temperature environment. This degradation is in the form of stress corrosion cracking, fatigue and cyclic crack growth, and toughness loss due to long-time aging at temperature. Evaluation of the factors causing cracking and of proposed fixes is included. Pipe rupture investigations are also a part of this research program.	- Evaluation of pipe cracking predictive models, proposed fixes, and weld repair criteria. - Technical basis for licensing decision on acceptance of the leak-before-break concept in LWR piping systems. - Predictive models on initial sensitization and intergranular stress corrosion cracking developed for evaluation of welding and repair-welding stainless steels. - Technical basis for establishing limits on environmental variables to control pipe cracking in LWR piping systems. - The effects of hydrogen and irradiation on stress corrosion cracking of piping and other materials and the effect of thermal aging on ferritic materials. - Information on the need for pipe whip restraints and jet impingement shields. - Summary report on pipe-to-pipe impact studies.
	1.4 Electrical and Mechanical Components. This research element deals with the time-related degradation of electrical and mechanical components during service and the potential impacts on public safety of the degradation of plant systems involving these components.	- General methodology for comprehensive assessment of aging of electrical and mechanical components. - Comprehensive assessment of the aging of selected plant components and systems. - Practical and cost-effective techniques for monitoring equipment for service wear effects. - Criteria for evaluating surveillance, maintenance, and replacement programs for selected components.
	1.5 Nondestructive Examination. This research element applies to the validation of reliable and reproducible NDE techniques for detecting and characterizing cracks and flaws for pressure vessels, piping, and steam generator tubing as well as the associated interpretation and analysis for decisionmaking.	- Technical basis for new requirements for ultrasonic inspection of vessel plate and forging to improve methods for through-weld and stainless steel inspection. - Validation in field tests of an improved ultrasonic testing method for flaw detection and evaluation using a synthetic aperture focusing technique. - Technical basis for improvements in the inservice inspection of steam generator tubing. - Validation of acoustic emission for leak detection and for continuous monitoring for cracks.

Research Program

Research Elements

Typical Products

Chapter 2
Equipment Qualification

This research program will evaluate the methods used for qualifying equipment used in nuclear power plants. The program will take into account such factors as effects of synergism, order or sequence of tests, accelerated aging techniques, and methods for simulating accident environments. Methods will be validated and new methods developed as appropriate to ensure that qualification test results reported by applicants and licensees provide a basis for licensing decisions that ensure protection of the public health and safety. It will also evaluate the survival of electrical equipment exposed to fire and severe accident environments, including hydrogen burns.

2.1 Qualification of Electrical Equipment for Harsh Environments. This research element will study the methods for qualifying safety-related electrical equipment to demonstrate the equipment's ability to function both during and following design basis accidents that produce harsh environments, including high radiation, temperature, pressure, and humidity, and to identify ways of reducing the likelihood of undesired failure modes.

- Acceptable methods for simulating aging and accident sequences in qualification tests of electric cables, penetrations, solenoid valves, etc.
- Data on thresholds of safety equipment failure due to fire.
- Determination of environments resulting from credible fires.

2.2 Qualification of Mechanical Equipment (Environmental). This research element will provide the technical basis for developing requirements for environmental qualification of mechanical components. Environmental parameters include temperature, pressure, humidity, radiation, chemicals, and submergence but not dynamic loads.

- Criteria for environmental qualification.
- Acceptable methodology for environmental qualification of mechanical equipment containing nonmetallic materials.
- Identification of significant environmental parameters affecting the functioning of equipment.

2.3 Dynamic Qualification of Equipment. This research element will provide the technical basis for developing the qualification requirements involving dynamic loads that originate either outside the equipment (e.g., seismic or other transmitted vibration) or inside the equipment (e.g., dynamic effects from process flow) for electrical and mechanical equipment.

- A data base of experience for use in assessing current qualification criteria and methods.
- Development of methods regarding the use of safety margins in equipment qualification.
- Identification of load combinations and sequences for use in the qualification of equipment.
- Criteria for extrapolating test results from one size component to another.

Research Program	Research Elements	Typical Products
Chapter 3 Seismic Research		
This research program will study earthquakes, which are among the most severe of the natural hazards faced by nuclear power plants, in terms of seismic hazard, seismic risk, and seismic margin.	3.1 Seismic Hazard. This research element will study the seismic hazard of nuclear power plants on the eastern seaboard of the United States to determine whether an earthquake of the magnitude of the Charleston earthquake could occur elsewhere on the eastern seaboard and result in a hazard to a nuclear power plant.	- Seismographic network data used in licensing, PRAs, and seismic hazard characterization. Geophysical data for determining crustal structure. Data from the in situ stress measurement program in the Northeastern United States. - Techniques for calculating site-specific response spectra. - Probabilistic sensitivity study of probable ground-motion dependence on the various proposed causes of seismicity in the Eastern United States. - Preliminary results of field investigations of the Meers fault in Oklahoma. - Validation of the DESRA code and recommendations regarding the use and criteria for the use and importance of SSI analytical methods for layered soil, high water tables, and liftoff effects.
	3.2 Seismic Risk. This research element will study the public risk associated with seismic events in excess of the current design basis, the safe shutdown earthquake.	- Summary reports describing and evaluating the seismic risk methodology as it relates to specific plants. - Nonlinear response data from HDR shake tests and earthquake response data from EPRI's scale-model containment in Taiwan. - Evaluation of the seismic safety margins research program's system analysis methods.
	3.3 Seismic Margins. This research element will study the margin inherent in the seismic design of older plants. New information available on seismicity, soil-structure interaction, and plant seismic response can result in changes in the predicted loads on structures, systems, and equipment. These changes must be compared to the inherent margin of the original design.	- Experimental validation of assumptions and analytical methods used to predict the fundamental natural frequency, accelerations, and displacements of structures. - Determination of the changes in floor response spectra, structural frequency, acceleration, displacement, and damping for input motion resulting in elastic and inelastic behavior. - Failure modes and failure level data for various concrete configurations. - Recommendations for structural load combination criteria. - Fragility data from component testing. - Recommendations for damping values to be used in piping analysis and risk comparisons among piping systems designed to various proposed new criteria.

Research Program	Research Elements	Typical Products
Chapter 4 Reactor Operations and Risk		
This research program supports the development of probabilistic risk assessment (PRA) methods and their use within the regulatory structure to identify those elements of reactor operations that are the most significant contributors to risk, to identify the causal factors associated with them, and to permit comparative evaluations of risk levels associated with various regulatory actions. Past efforts in this area have identified the man-machine interactions as an area of significant uncertainty and therefore a potentially large contributor to risk. This work includes the development and trial use of models, methods, procedures, and other analyses required to support Commission decisions on a broad range of critical issues relating to power reactor safety and the acquisition of data to support the application of PRA methods to the regulatory process.	<u>4.1 Reliability and Risk Methodology</u>. This research element is directed toward developing, testing, documenting, and, to the extent possible, validating methods for estimating the probabilities and consequences of severe reactor accidents and toward evaluating and reducing the uncertainties in such estimates.	Identification and review of accident sequences and their likelihood and related probabilities. A procedure for incorporating dependency analyses, including common-cause failures, into a PRA. Procedures for incorporating results from the seismic safety margin research program into PRA methodology.
	<u>4.2 Data Base Development and Evaluation</u>. This research element consists of data collection and analysis efforts that serve as a foundation for the application of risk methods.	A data base on component and system reliability developed from evaluations of plant operating data, licensee event reports (LERs), and vendor information. Testing of integrated methods using improved data involving internal, external, and common-cause risk assessment techniques in the RMIEP for better identification and display of quantitative uncertainties.
	<u>4.3 Regulatory and Inspection Applications</u>. This research element will develop methods of analysis to permit more systematic evaluations to be made of the cost effectiveness of current or proposed regulatory requirements, alternative concepts for reactor design and operation, and decisions on backfitting.	Results of the demonstration of and recommendations for the reliability assurance program. Information based on risk assessment insights and updated analysis of dominant accident sequences for PRA analyses to aid in developing and setting priorities for IE inspection activities. Plant-specific analyses of accident management strategies and a methods application guide for use on other plants. Evaluation of emergency action level identification.

Research Program

Research Elements

Typical Products

Chapter 5 Thermal-Hydraulic Transients

This research program provides the experimental data and analytical methods needed to predict and understand primary and secondary coolant systems during all types of plant transients, including the full range in the sizes of pipe ruptures. The resulting analytical methods are used to quantify margins of Appendix K to 10 CFR Part 50, to assist the regulatory assessment of operator guidelines for accident management, and to analyze complex plant system transients.

<u>5.1 Separate Effects Experiments and Model Development</u>. This research element consists of experiments designed to provide data specific to various phenomena such as two-phase (steam/liquid) heat transfer, downcomer thermal mixing, and flow characteristics in the range of conditions that occur in reactors during transients and accidents.

- Evaluation of the scaling of integral test data and of scaling methods at alternative reactor system test facilities.
- Revision of heat transfer packages in RELAP5 and TRAC based on assessment results.
- Validated steam generator data from MB-2 and improved models for iodine transport through the secondary side.
- Critical flow models for pipe cracks.
- Thermal mixing model to predict fluid temperature fluctuations at the HPI and cold-leg interface.

<u>5.2 Integral Systems Experiments</u>. This research element includes experimental simulations of integral thermal-hydraulic systems of light water reactors. Transients simulated include the full break-size spectrum of loss-of-coolant accidents, loss of feedwater, steam line and feedwater line breaks, steam generator tube rupture, anticipated transients without scram, and various safety and control system failures.

- Semiscale data to be used to assess calculations of steam line and feedwater line breaks, small-break LOCAs without HPI, and analysis codes for large-break LOCAs in plants equipped with upper-head injection.
- Full-scale fluid-fluid mixing and ECC bypass data from UPTF and SCTF.
- Experimental tube rupture data for once-through steam generators from OTIS and MIST, including integral natural circulation data, integral small-break LOCA tests, and integral feed-and-bleed tests in MIST.
- Recommendations on meeting NRC's long-term needs for integral facilities.

<u>5.3 Code Assessment and Application</u>. This research element involves the application of computer codes to the analysis of transients in full-scale light water reactors and the assessment of these analytic capabilities against experimental data.

- Development of statistical uncertainty for large thermal-hydraulic analysis codes.
- Summary status report on ECCS research.
- Summary reports on assessment, accuracy, and limitations of frozen versions of TRAC and RELAP5 for PWRs and BWRs, including BWR-2 plants and PWRs with UHI and UPI.
- Final major versions of the advanced multidimensional two-fluid transient anlaysis codes TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1.
- Assessment of TRAC-BF1 by DOE laboratories.

<u>5.4 Plant Analyzer and Data Bank</u>. This research element includes improvements in computational techniques and the development of user-oriented capabilities for using advanced versions of these codes in the form of an interactive plant analyzer. It also includes the acquisition and manipulation of plant data needed to develop input specifications for plant-specific analyses.

- Increasing the speed of TRAC by vectorization and two-step numerics in vessel components on CRAY-1 and of RELAP5 by conversion to CRAY-1.
- First general user version of PWR system plant analyzer completed and demonstrated for use by NRC personnel.
- Advanced version of PWR system plant analyzer with complete developmental assessment.
- First general user version of BWR system plant analyzer with complete developmental assessment demonstrated for use by NRC personnel.

Research Program

Chapter 6
Severe Accidents

This research program supports the reassessment of the regulatory treatment of severe accidents in nuclear power plants. It includes the coordinated phenomenological research programs needed to develop a sound technical basis for NRC decisions concerning the ability of reactors to cope with these accidents.

Research Elements

Typical Products

Accident Analysis

6.1 Accident Likelihood Evaluation
6.2 Severe Accident Sequence Analysis

- Identification, review, and delineation of accident sequences.
- Evaluations of plant operating data, LERs, and vendor information for component and system reliability.
- Technical bases for improved risk assessments.
- Basis for ex-containment source term as distinct from source term release to containment.

Fuel & Fission Products

6.3 Behavior of Damaged Fuel
6.5 Fuel Structure Interaction
6.7 Fission Product Release & Transport
6.9 Fission Product Control

- Report on NRU SFD full-length verification tests.
- Report on the analysis of power burst facility tests.
- The final report on the independent examination of selected TMI-2 core samples.
- The MELPROG-Mod 1 code released.
- Reports on ACRR debris formation tests and the melt progression separate effects tests with irradiated fuel.
- Assessment of the MELPROG code with PBF and ACRR results.
- Assessment of the final version of SCDAP with PBF, NRU, and ACRR results.
- Report on limited sensitivity studies using MELCOR/SCDAP/MELPROG.
- Verification of fuel-melt/concrete interaction models in CORCON to be obtained from large-scale test facilities.
- Verification of the CORCON and CONTAIN codes.
- Experimental data for modeling the interaction of hot solidified melts with concrete.
- Predictive mechanistic models for the explosive and nonexplosive interaction of thermite melts with water.
- Results from high pressure ejection tests.
- For steam explosions, quantification of functional relationship between energy conversion ratio and mass of molten material.
- Data for developing and improving in-vessel fission product release models (FASTGRASS, VICTORIA, and CORSOR).
- Data for benchmarking models for in-vessel release of fission products from in-pile experiments.
- Results from large-scale fission product and aerosol transport tests.
- Severe accident source term uncertainty data.
- Verification of the ICEDF code for evaluating the PWR ice condenser.
- Development and verification of codes for the performance of filtration systems under aerosol loadings.
- Modification and verification of a code for evaluating the performance and effectiveness of PWR/BWR containment sprays under severe accident conditions.
- Evaluation of the design of existing PWR/BWR engineered safety features.
- Development and verification of codes for evaluating the reliability and aging aspects of engineered safety feature systems.

Containment

6.6 Containment Analysis
6.8 Containment Failure Mode
6.4 Hydrogen Generation and Control

- Verification of the CONTAIN code.
- Integration interface between CONTAIN code and other codes such as TRAP-MEL for fission product and energy source data.
- Integration interface between CONTAIN and fission product dispersion code for computing offsite doses.
- Results of tests on valves and electrical penetrations.
- Results of final test series on seals and gaskets.
- Results of tests on models of major penetrations.
- Comparisons of predicted capacities of containments under static and dynamic pressure loads with test results.
- Results of initial tests of containment models under simulated seismic loading.
- Analysis of three to five specific plants for degraded core and severe accidents.
- Preliminary hydrogen diffusion model and flame acceleration model.

Risk

6.10 Risk Code Development
6.11 Accident Consequence & Risk Reevaluation
6.12 Risk Reduction & Cost Analysis

- The MELCOR code documented for use in such programs as the severe accident research program and the risk methodology integration and evaluation program.
- Iterative consequence and risk evaluations of important accident sequences.
- Generic assessment of costs and risk-reduction potential of alternative safety features applicable to specific classes of LWRs.
- Evaluation of the feasibility of using PRA to improve reliability of existing plant systems.

Research Program

Chapter 7
Radiation Protection and Health Effects

This research program will provide a technical and scientific data base that permits accurate identification and measurement of sources of radiation exposure and clearly defines the relationship between exposure and potential health effects, thus allowing issuance of radiation exposure limits with a sound scientific basis for use in the work place and in the general environment. There are significant uncertainties in some areas of radionuclide metabolism and internal dosimetry, in the characteristics of dose-effect relationships, and in radiological dose measurements in the work place.

Research Elements

7.1 Metabolism and Internal Dosimetry. This research element will study the physiological and metabolic processes that determine the distribution and retention of radionuclides in various body organs and the type, energy, and spatial distribution of the radiation emitted from these internally deposited radionuclides to determine the radiation dose to the human body following the intake of radionuclides through inhalation, ingestion, or absorption through the skin and wounds.

7.2 Health Effects and Risk Estimation. This research element will provide basic data, improved methods of data analysis, and models for predicting the adverse health effects resulting from exposure to ionizing radiation (both external and internal) and for quantifying the health risks related to the use or release of radiation and radioactive materials. The element will consist of experimental and epidemiological studies and analyses designed to assess the dose-effect relationships (risk coefficients) for types of radiation and levels of exposure that might be encountered in the work place, in the environment, and following potential major accidents.

7.3 Occupational Radiation Protection. This research element will provide information needed to help ensure an adequate degree of radiation protection for workers in NRC-licensed facilities and activities.

Typical Products

7.1:
- Metabolic model for inhaled yellowcake.
- Values of the gastrointestinal absorption factor for the actinides.
- Production of a modified internal dosimetry code.
- Compilation of age- and sex-specific dose conversion factors.

7.2:
- Improved statistical procedures that will find application in other hazard evaluation studies (1986).
- Revised (or reaffirmed) values for neutron quality factor.
- Improved estimate of the dose/response function for radon daughter exposure/lung cancer risk in the general public, including nonsmoking populations.
- Revised risk coefficients for low-level, low-LET radiation and risk coefficients for internally deposited alpha emitters.
- Development of models for morbidity and verification and improvement of models for early mortality resulting from radionuclide inhalation.

7.3:
- Assessment of decontamination projects, including recontamination rates, at selected nuclear power plants with regard to overall effectiveness and safety.
- Data needed for modification of respirator protection factors shown to be needed by field-testing with workers.
- Computerized system for accessing and analyzing radiological health protection information.

Research Program	Research Elements	Typical Products

Chapter 8
Waste Management

This research program will provide a technical capability to assess compliance of a waste management system with the regulatory requirements for operational safety, occupational radiological protection, and long-term waste isolation.

8.1 High-Level Waste. This research element will provide an understanding of the relevant phenomena and processes that affect the performance of a geologic repository both during and after waste emplacement operations to permit an independent assessment of DOE compliance with EPA and NRC regulations sufficient to provide reasonable assurance of safety.

- Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes.
- Identification and description of the relationship among parameters important in container manufacturing and expected long-term container performance.
- Identification and description of methods for predicting long-term performance of waste packages in basalt and tuff (including waste form, container, and overpack).
- Parameters and physical relationships important to methods for evaluating long-term performance of backfill systems.
- Applicability of hydrothermal and geothermal data and predictive techniques to waste isolation performance assessments.

8.2 Low-Level Waste. This research element will identify and provide an understanding of the phenomena and processes that affect the stability of disposal sites for low-level waste, the isolation and mobility of radionuclides, and the associated uncertainties.

- Evaluation of long-term performance of wastes and containers produced by available technologies.
- Assessment of means to enhance the control of water entry into below-grade burial trenches.
- Assessment of monitoring methods, particularly through the use of environmental methods applicable to the postclosure period.
- Assessment of the interaction of radionuclides with soils to predict the performance of an LLW disposal facility.
- Description of coupled geochemical/hydrological phenomena relevant to migration of radionuclides from shallow-land burial facilities.
- Evaluation of properties of and release of chelating agents (organic complexants) from solidified decontamination wastes (1986).

INTRODUCTION

This appendix has been prepared to supplement the Long-Range Research Plan (LRRP) by listing the specific uses of research in the regulatory process for the years 1984 and 1985.

Uses of Research

The Office of Nuclear Regulatory Research was established by specific provisions of the Energy Reorganization Act of 1974. By this action, Congress recognized the desirability of providing NRC with an independent capability to develop technical information free from potential biases or conflicts of interest. The purpose of the research program is to provide the technical basis for rule-making and regulatory decisions; to support licensing and inspection activities; to assess the feasibility and effectiveness of safety improvement concepts; and to increase our understanding of phenomena for which analytical methods are needed in regulatory activities.

A research program will very often serve several of these regulatory purposes. The research program supports more than the establishment of rules, guides, and standards. The research program also develops data, methods, and procedures whereby the licensing and inspection and enforcement staffs can improve their assessment techniques, reduce subjectivity in staff judgments, improve the efficiency of the licensing process, and strengthen our inspection procedures. These improved techniques are often incorporated into the regulatory process through revisions to licensing standard review plans and inspection modules. These regulatory actions fall within the areas of responsibility of the other line offices in NRC; thus they often do not appear explicitly as outputs from the RES program. The rules, guides, and other regulatory actions comprise interlocking regulatory solutions flowing from the research programs.

Research results also support the Commission's policymaking activities, again not always explicitly. For example, the promulgation of the draft Safety Goal implicitly reflects the progress that has been made in providing better quantification of the risk presented by the present generation of nuclear power plants.

Finally, research is also carried out to validate or confirm the adequacy of fundamental data that have in the past formed the basis for key Commission policy and licensing decisions. The need to conduct such confirmatory research was explicitly recognized in the language of the House-Senate Conference Committee Report on the Energy Reorganization Act of 1974.

In summary, research results have many broad applications in the regulatory process. Many research results are used to directly support the development of rules, standards, or guides; such regulatory products are set forth in this report. However, because the other interlocking regulatory solutions flowing from the research program, the research supporting policymaking decisions, and the confirmatory research often all come from the same research program, it is not possible to attribute x research dollars to y rules or guides.

Organization of this Appendix

This appendix follows the format of the LRRP. There are eight chapters, which correspond to the eight chapters of the LRRP. These contain listings of the regulatory products* of the research program completed in 1984 and targeted for completion in 1985. A ninth chapter has been added to this appendix for listing regulatory products of research programs that are nearing completion and therefore are not included in planning for 1986 and beyond. The regulatory products that are regulations are identified with the symbol †. This symbol is also used in the LRRP to point out that regulations may result from the research program.

Table B-1 summarizes the changes in regulations that may result from the research described in the LRRP, i.e., research conducted in 1986 and beyond.

Table B-2 shows the funding and staffing levels at the section level of each chapter for the base year 1986.

*Regulatory products are those results of research that are used in the regulatory process. Research products, listed under LRRP program descriptions, result directly from research and are used as the basis for developing a regulatory product.

1. OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR

1.1 Reactor Vessels

1. Verification of prediction of crack initiation, propagation, and arrest, as well as warm prestressing, under pressurized thermal shock (PTS) conditions to validate the presently used NRC methodology for predicting the effects on the structural integrity of reactor vessels subjected to a PTS scenario (1985).

2. New data on importance of the sensitivity to irradiation damage of the residual and alloying elements copper and nickel in pressure vessel steels, to be used for revising Regulatory Guide 1.99 (1985).

3. New data on environmentally assisted fatigue crack growth of pressure vessel steels to be incorporated in revisions of the ASME B&PV Code, Section XI, Appendix A (1985).

4. Validation of benchmark dosimetry methodologies and data base used to predict neutron fluence and radiation damage in reactor pressure vessel steels (1984, 1985).

1.2 Steam Generators

Data provided to NRR to validate current licensing positions on:

1. Burst pressures of corrosion cracked tubes (1985).

2. Correlation of eddy current flaw evaluation to actual flaws and corresponding failure pressures (1984, 1985).

3. Evaluation of reliability of both current and advanced eddy current nondestructive examination (NDE) methods for detecting and characterizing flaws and degradation in steam generator tubing (1985).

1.3 Piping

1. Information to support the resolution of USI B-6 concerning decoupling of seismic loads and double-end guillotine breaks in primary piping (1984, 1985).

2. Probabilistic information to assist in the resolution of USI A-2 concerning asymmetric LOCA (1984).

3. Probabilistic analytical models of intergranular stress corrosion cracking (IGSCC) in BWR recirculation loop piping, to be used for making regulatory decisions on safe operation of cracked piping (1984).

4. Draft regulatory guide on anchoring components and structural supports to concrete (1984).

5. Test data to serve as the basis for confirming acceptance criteria for pipe-to-pipe impact in Section 3.6.2 of Standard Review Plan (SRP) and the analytical basis for justifying the amplification factor for pipe restraint design in Section 3.6.2 of SRP (1985).

6. Information to validate licensing positions on leak before break and on BWR pipe cracks:

   a. Ductile fracture properties for piping materials (1984, 1985).

   b. Pipe fracture tests to validate fracture mechanics analyses and to evaluate ASME Code flawed-pipe evaluation rules in IWB-3640 (1984, 1985).

   c. Evaluations of short-term and long-term fixes for stress corrosion cracking in BWR piping for revision of NUREG-0313 (1984, 1985).

   d. Data on susceptibility of piping materials to stress corrosion cracking and rate of crack growth in spectrum of BWR water chemistries (1984, 1985).

   e. Information for NRC Piping Review Committee on proposed regulatory positions related to piping integrity (1984, 1985).

7. Probabilistic fracture mechanics and systems engineering calculations to support rulemaking to modify GDC-4 dealing with the codification of leak-before-break technology (1985).

1.4 Electrical and Mechanical Components

1. Nuclear Plant Aging Research (NPAR)

   a. Two technical reports describing findings of workshops and surveys of operating experiences from Licensee Event Reports (LERs) to identify aging trends in vital electrical and mechanical equipment (1984).

   b. Evaluation of signature analysis technique to assess degradation and misadjustments of motor-operated valves, to support NRR in resolving Generic Issue II.E.6.1, "In Situ Testing of Valves" (1985).

   c. Three technical reports, based on operating experience reviews and engineering analysis, identifying failure modes and causes and assessing the influence of aging and service wear of electric motors, motor-operated valves, and mechanical and hydraulic snubbers (1985).

2. Spent Fuel Storage

   a. Report on the proceedings of an NRC workshop on the effects on cladding and fuel during dry storage, to be used for information purposes in evaluating design concepts during licensing evaluations (1984).

   b. Three technical reports on nondestructive and destructive examination of spent fuel stored under dry conditions (1984-1985).

   c. Regulatory Guide 3.54 on spent fuel heat generation (1984).

   d. Draft of 10 CFR Part 72 (1985).†

   e. Draft regulatory guide on design of independent spent fuel storage installation (dry type) (1985).

3. Decommissioning

   a. Five technical reports containing data bases for evaluating contents of decommissioning plans for nuclear reactors and fuel cycle facilities (1984-1985).

   b. Two technical reports to serve as data base for preparing decommissioning rule (1984-1985).†

   c. Three technical reports on data collection during decommissioning and decontamination of reactors (1984-1985).

   d. Proposed amendments to 10 CFR Parts 30, 40, 50, 51, 70, and 72 on decommissioning criteria for nuclear facilities (1985).†

   e. Regulatory guide on assuring funds for decommissioning material licenses (1985).

1.5 Nondestructive Examination

1. Data transmitted to NRR on NDE for sizing of IGSCC and fatigue cracks for use in safety analyses (1984, 1985).

2. Recommendations and test results transmitted to NRR on improved eddy current testing of steam generator tubing (1984).

3. Independent ultrasonic inservice inspection (ISI) of pipe welds in two LWRs to referee other ISI results in disagreement over the presence of cracks (1984).

4. Regulatory guide to document current procedures for ISI of piping, as well as to upgrade those procedures as required (1984, 1985).

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†The regulatory product involves rulemaking action.

5. Development and submittal for consideration to ASME Code Committee of improved qualification criteria for performance demonstration of ultrasonic inspection personnel, equipment, and procedures for stainless steel piping (1984, 1985).

2. EQUIPMENT QUALIFICATION

2.1 Qualification of Electrical Equipment for Harsh Environments

1. Revision to Regulatory Guide 1.89 on environmental qualification of electric equipment (1984).

2. Evaluation of sequential versus simultaneous simulation of aging and loss-of-coolant accident/main steam line break conditions for a large number of polymer materials used in safety-related cables, gaskets, seals, terminal boards, and connectors to provide data for licensing decisions (1984).

3. Evaluated qualification of EPR-insulated cables to determine if deficiencies identified in IE Notice No. 84-44 would be expected to result in failure (1984).

4. Technical report evaluating the effect of LOCA-simulation procedures on cross-linked polyolefine cable performance, to provide NRR with verification data for use in licensing decisions (1984).

5. Four technical reports describing 12-year-old station batteries that were removed from nuclear power plants and evaluated for the potential for common mode failure and for the significance of battery case cracks, to resolve concerns identified in IE Notice No. 84-43 (1985).

6. Reevaluation of qualification procedures for EPR-insulated multi-conductors using a superheated steam LOCA profile, to verify adequacy of industry standard (1985).

7. Revision to Regulatory Guide 1.63 to improve the design, testing, and circuit protection requirements for electric penetration assemblies (1985).

8. Regulatory guide on qualifying connector assemblies for nuclear power plants to ensure that electrical connectors used inside containments have been shown (qualified) to be capable of functioning under the stresses of design basis LOCAs (1985).

9. Fire Protection Research Program

   a. Analysis by computer simulation of probable fire environments in control rooms and validation of computer model for predicting fire damage in control rooms, for use in making licensing decisions (1985).

   b. Determination of fire failure thresholds of safety-related equipment, for use in making licensing decisions (1985).

10. Regulatory guide on criteria for programmable digital computer systems of nuclear power plants (1985).

11. Draft regulatory guide on quality assurance of computer-based protection system software (1985).

12. Evaluation of safety impact of failures in non-safety-grade control systems for Oconee and Calvert Cliffs to support resolution of USI A-47 (1985).

2.2 Qualification of Mechanical Equipment (Environmental)

Data will be provided to NRR to validate or establish licensing positions on:

1. Leakage threshold and leakage rates of large containment penetration seals when subjected to pressures and temperatures typical of normal and severe accident conditions (1984).

2. Integrity of elastomeric pump shaft seal (1984).

3. Operability of purge and vent valves during design basis LOCA environments (1985).

4. Leakage integrity of purge and vent valves during severe LOCA accident environments (1985).

2.3 Dynamic Qualification of Equipment

1. Development of basis for purge valve qualification by extrapolation, data to be supplied to NRR to validate or establish licensing positions (1985).

2. Development of method for using safety margins in equipment qualification (1985).

3. SEISMIC RESEARCH

3.1 Seismic Hazard

1. Technical reports containing seismographic network data for the Eastern United States used by the NRR staff in preparing safety evaluation reports and at board hearings on a regular basis. Recent examples include Summer, Vogtle, Grand Gulf, Midland, Seabrook, Shearon Harris, etc., PRAs, rulemaking decisions, and engineering research projects (1984, 1985).

2. Technical reports and computer codes from the seismic hazard characterization of the Eastern United States project, a joint project with NRR to develop a screening tool to be used by NRR to address the issue of the occurrence of a "Charleston earthquake" anywhere on the Atlantic seaboard (1985).

3. Technical reports describing preliminary results from a geological and geophysical investigation of the Ramapo Fault System of New York, New Jersey, and Pennsylvania, to be used to understand seismotectonics of the northeastern region and to understand seismogenic mechanisms for Eastern United States tectonic provinces for nuclear power plant licensing decisions (1985).

4. Technical reports and summaries from in situ stress measuring project to provide insight to NRR on the fault systems that might be reactivated as current sources of seismicity and to aid NRR in routine decisionmaking process (1985).

5. Technical reports on soil failure modeling with particular emphasis on soil liquefaction (1984, 1985).

3.2 Seismic Risk

1. Technical report (5 volumes) on engineering characterization of earthquake ground motion (including consideration of wave passage and soil-structure interaction) and inelastic response of structures (1985).

2. Simplified PWR seismic risk methodology to serve as benchmark for future seismic risk analyses and reviews (1985).

3.3 Seismic Margins

1. Application of detailed analytical methods and data bases developed in the Seismic Safety Margin Research Program (SSMRP) to determine seismic risk at the Zion nuclear power plant (1984).

2. SSMRP analyses used as benchmarks in studies leading to NRC Piping Review Committee recommendations on pipe damping, peak shifting, and multiple-spectra methods (1984).

4. REACTOR OPERATIONS AND RISK

4.1 Reliability and Risk Methodology

1. Methods to assist NRR in evaluating industry initiatives on nuclear power plant personnel training programs to support NRC policy statement on training (1984).

2. Eleven technical reports on human reliability for use in establishing a data base and review process for human behavior and errors in operating nuclear power plants (1984).

3. Methods that will assist NRR in implementing the Integrated Safety Assessment Program (1985).

4. Effectiveness of nuclear power plant operator licensing examinations to support NRC policy decisions on simulator examination requirements (1985).

5. Effectiveness of function-based emergency operating procedures in nuclear power plants (1985).

6. Final Commission Policy Statement on fitness for duty of personnel with unescorted access to protected areas (1985).

4.2 Data Base Development and Evaluation

1. Priorities set for resolution of outstanding safety issues (1985).

2. Assessment of risk importance of operating experience to identify potential generic safety issues (1985).

4.3 Regulatory and Inspection Applications

1. Recommendations for implementing the Commission's safety goal policy (1985).

2. Assessment of risk importance of present regulatory requirements (10 CFR Part 50), in particular those affected by revised source terms (1985).

3. Methods to revise technical specifications for operating reactors (LCOs, surveillance testing requirements, and AOTs) based on risk significance (1985).

4. Methods and procedures to assist IE in establishing priorities for IE inspection activities based on risk-important measures (1985).

5. Statistical methods to permit establishment of criteria for licensee's reliability assurance program (1985).

6. Emergency Preparedness

   a. Rulemaking on refining emergency preparedness regulations (use of new source term information) (1985).†

   b. Proposed amendments to 10 CFR Parts 30, 40, 70, and 72 regarding emergency preparedness for fuel cycle and byproduct material licensees (1985).†

   c. Rulemaking to delete the "unusual event" emergency classifications (1985).†

   d. Rulemaking to address the consideration of earthquakes in the context of emergency planning (1985).†

7. Guidelines for value-impact analysis for agencywide use to provide cost-benefit information in support of major regulatory actions (1984).

8. Improved methods for use in performing value-impact analysis on major regulatory actions and value-impact information data base to support regulatory analysis and rulemaking control (1985).

9. Reference methods to be used to evaluate safety goal applications, to support generic rulemaking using probabilistic risk assessment (PRA) as one factor to be considered and to evaluate the adequacy of PRAs submitted by licensees as required by the licensing process (1985).

†The regulatory product involves rulemaking action.

5. THERMAL-HYDRAULIC TRANSIENTS

5.1 Separate Effects Experiments and Model Development

1. A level detector device based on existing thermal neutron detectors developed by Penn State and tested at LOFT to demonstrate the possibility of using these devices in operating reactors to detect accident conditions (1984).

2. Computer model of downcomer and cold-leg fluid mixing for use in evaluating pressurized thermal shock (1984).

3. Thermal-Hydraulic Systems Analysis handbook completed for use by AEOD and IE (1984).

4. Data on critical flow from horizontal pipes to be generated by the Thermal-Hydraulic Loop at INEL under a cooperative program with EPRI for use in helping NRR to evaluate critical flow under separate flow conditions (1984).

5. Completion of modifications to the CITADEL code for use by NRR in performing iodine transport analyses (1984).

6. Thermal fluid mixing and pressure vessel fluid heat transfer models for NRR use in pressurized thermal shock analyses for licensing applications (1985).

7. Data from testing at MB-2 Steam Generator Facility for loss of feedwater, steam generator tube rupture (SGTR), and steam line break, to be used to show the effectiveness of steam separators and other hardware in reducing iodine release during SGTR accidents over the release currently assumed in licensing calculations (1985).

5.2 Integral Systems Experiments

1. The effect of pressurizer spray on controlling transients, including feed and bleed scenarios, was studied in the Semiscale facility. The results of these tests are being used by NRR to help evaluate these procedures in Combustion Engineering and other plants (1984).

2. Assessment of advanced computer codes using LOFT thermal-hydraulic test data to provide confidence in analyzing similar events in full-scale LWRs (1984).

3. The downcomer injection test in Cylindrical Core Test Facility can be used to evaluate proposed U.S. vendor design changes in next generation PWRs (1984).

4. Simulations of BWR transients, including large-break LOCA, small-break LOCA, ATWS, and steam line break, were performed at the BWR

Full Integral Simulation Test (FIST) facility. These data were used in support of GE Safer evaluation model approval (1985).

5. Steam line and feed line break series and special liquid holdup tests have been completed for SGTR and power loss test series at Semiscale. The results are being used to assess steam and feed line break calculations (1985).

6. MIST program construction is 75 percent completed on schedule and within cost. Analyses of OTIS tests are completed and are being reviewed by program partners (NRC, EPRI, B&W, and B&W Owners' Group) (1985).

5.3 Code Assessment and Application

1. Computer code analyses of several pressurized thermal shock transients that were input to fracture mechanics calculations to establish the probability of vessel failure for individual plants that may fail the screening criteria (1984).

2. Interim fast-running version of TRAC-BWR (TRAC-BFO) available for use on NRC programs (1985).

3. Assessment of TRAC-PD2, TRAC-PF1, and TRAC-BD1 completed, these computer codes to be applied to actual licensing cases (1985).

5.4 Plant Analyzer and Data Bank

1. Integral loop test results, plant data, and computer code calculations displayed in a loop configuration to help NRC engineers better visualize and understand the time-dependent position of fluid in reactor systems (1984).

2. Demonstration of the BNL plant analyzer using the latest hardware and software to significantly speed up computer calculations. The purpose of this concept is to reduce the cost and time of calculations, to allow visualization of the accident on a screen while in progress, and to make numerous sensitivity studies practical (1984).

3. Addition of major capabilities to the plant analyzer version of the TRAC-PWR code permits fast-running calculations (1985).

4. Addition of small-break LOCA capability in steam line, for use in I&E training drills (1985).

5. Collection of all transient analysis descriptions and sample microfiche results from contractors for indexing in the nuclear plant analyzer, and training for conducting a search provided to NRR (1985).

6. SEVERE ACCIDENTS

6.1 Accident Likelihood Evaluation

1. Methods for systematic identification and evaluation of principal reactor accident sequences to support decisions on the severe accident rule (1984, 1985).

2. Techniques for incorporating, qualitatively and quantitatively, the contribution of common-cause failures and system interactions into PRA methods to support risk assessments of significant safety issues (1985).

3. Methods for quantifying the effects of severe natural phenomena (e.g., seismic activity, floods) and human factors on assessments of reactor risk to support risk assessments of significant safety issues (1985).

6.2 Severe Accident Sequence Analysis

1. SASA studies of inadvertent boron dilution event to support the resolution of Generic Issue 22 (1984).

2. Greater understanding of the capability of operating PWRs to remove decay heat using feed and bleed following loss of all secondary cooling, to provide assistance to NRR in evaluating USI A-45 (1984).

6.3 Behavior of Damaged Fuel

1. Results have been obtained on core damage progression, hydrogen generation, and fission product release and transport under realistic in-vessel severe accident conditions in the final two severe fuel damage (SFD) tests in PBF with 28-rod bundles of high-burnup fuel. These and the previous two PBF SFD tests have provided the only large integral (multi-effect) data up to near fuel-melt temperature on these subjects, and these data have provided a major part of the technical basis for decisions on source term reassessment and severe accident policy (1984, 1985).

2. Results from PBF SFD tests have been used as the technical basis for interpretation of the results of the DOE examination of TMI-2 core debris (1985).

3. Results of two small-scale ACRR debris formation and relocation experiments have provided new data on fuel damage development, fuel liquefaction, fuel relocation, and core melt progression, along with discovery of the generation of a dense tin aerosol at relatively low fuel temperatures. These results have been used in the development and assessment of the SCDAP, MELPROG, and TRAP-MELT mechanistic accident analysis codes (1984, 1985).

4. Results of two ACRR damaged-core-coolability experiments have provided data on the dryout coolability limits of LWR core debris up to 2,600 psi. These results have generally agreed with the existing advanced LMFBR core-debris coolability models and provide validation for the use of these coolability models for LWR safety assessment (1984).

5. The QUEST source term uncertainty analysis for severe accidents showed that the large source term uncertainties (about two orders of magnitude) are dominated by uncertainties in the in-vessel fission product release rates and the in-vessel core melt progression, which strongly affects both the in-vessel fission product release and the ex-vessel threat to containment. These results have been used both in the source term reassessment and in directing the severe accident research program (1984).

6. Results of the first full-length cooldown test in NRU have given good agreement with SCDAP modeling of the rapid oxidation (and hydrogen generation) transient and on the transient threshold. The results are significant for hydrogen generation in the regulatory implementation of the source term reassessment and for accident management planning (1985).

7. The SCDAP mechanistic fuel damage code has been used to benchmark the MARCH analyses in the NRC source term reassessment, in analysis of the TMI-2 accident, and in analysis of in-pile SFD experiments in PBF, ACRR, and NRU (1985).

8. In support of NRR, the COMMIX detailed thermal-hydraulics code has been used for analysis of the threat to the integrity of the steam generator tubes in core-uncovery accidents by stratified natural-convection flow in the hot leg and in the steam generator tubes themselves (1985).

9. The newly operational linked MELPROG-TRAC two-dimensional in-vessel core melt progression and thermal-hydraulics code has been used for analysis of the in-vessel natural convection and heat-transfer processes and their effect on severe accident sequences and consequences. This is particularly important in the station blackout sequence and for consideration of direct containment heating. The results will be used in the regulatory implementation of the source term reassessment (1985).

6.4 Hydrogen Generation and Control

1. Analysis of hydrogen combustion potential of three to five specific plants with various containment types. This work is directed toward resolving the issue of local and global combustion during severe accidents (1984).

2. As part of the cooperative program with EPRI, a number of hydrogen combustion and equipment survival experiments have been conducted to assess the risk of hydrogen burning in large, dry containments and to

assess the efficiency of ignitors in pressure-suppression containments in support of the final Hydrogen Control Rule (1985).

3. Data base for resolving licensing condition at Sequoyah concerning operability of Tayco igniter in water spray environment (1984).

4. Two technical reports, one on accident-generated jets and problems using deliberate flaring from high-point vents to eliminate hydrogen from primary containment (1984) and the other on hydrogen combustion experiments in the VGES facility (1984).

5. Revision to § 50.44 of 10 CFR Part 50 to require hydrogen control systems for Mark III BWRs and ice condenser PWRs that can handle the hydrogen from a degraded core accident. Essential systems must be able to function during and following a hydrogen burn if the containments are not inerted (1984).†

6. Two technical reports, one on the behavior of hydrogen igniters in the presence of water sprays and gas flows (1985) and the other on air currents driven by sprays in reactor containment buildings (1985).

7. Evaluation of equipment response and survival in support of a possible expansion of the Hydrogen Control Rule for a hydrogen burn in large, dry PWR containments (1985).

8. Development of equipment response and survival models for licensing assessment of responses to the Hydrogen Control Rule (1985).

9. Assessment of the potential for local detonations in large, dry PWR containments in support of the Hydrogen Control Rule (1985).

6.5 Fuel-Structure Interaction

1. Large-scale core/concrete interaction tests provided the data base for predictive modeling of concrete ablation in the CORCON code for the analysis of ex-vessel fuel/structure interaction in the assessment of risk at individual nuclear plants from severe accidents (1984).

2. Test results regarding noncondensible/combustible gas generation are also included in the CORCON code for analyses of their contribution to containment loading (1984).

3. A research information letter was issued on basemat penetration rates to be used for severe accident risk analysis (1984).

4. A research information letter was issued on heat generation and release to be used for risk analysis of containment performance during severe accidents (1984).

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†The regulatory product involves rulemaking action.

5. Experimental data on aerosol generation in the core/concrete interaction are used to develop analytical models in the VANESA code for prediction of ex-vessel fission product releases (1985).

6.6 Containment Analysis

1. Release of first version of CONTAIN 1.0 for general application by reactor safety community. The CONTAIN code is NRC's primary computational tool for predicting and analyzing atypical containment system loading imposed by severe accident conditions (1984).

2. Over 24 tapes of CONTAIN 1.0 distributed to laboratories throughout U.S. and in five foreign countries (1985).

6.7 Fission Product Release and Transport

All programs listed below will provide technical input to existing rulemaking and regulatory guides for the possible future adoption of a severe accident source term and will be used along with severe accident policy decision and safety goals:

1. Best-estimate assessment of the source term on the basis of current knowledge for the risk-dominant accident sequences in five representative plants, for use by the Commission in source term reassessment (1985).

2. Results from small-scale laboratory experiments to resolve questions regarding parametric effects on fission product release rates and chemical forms (1985).

3. Data report on severe accident control rod aerosol generation from out-of-pile fuel bundle experiments (1985).

4. Preliminary version of the mechanistic FASTGRASS computer code for calculating the severe accident release of volatile fission products from fuel (1985).

5. Best-estimate severe accident source terms revised for use by the Commission to reassess the source term currently used (1985).

6. Development of models in ice condensers and water-suppression pools for best-estimate accident source term calculations for ice condenser PWRs and for BWRs (1985).

6.8 Containment Failure Mode

1. Draft general revision of Appendix J to 10 CFR Part 50 on containment leak-testing (1985).†

†The regulatory product involves rulemaking action.

2. Evaluation of state-of-the-art calculation methods to predict structural behavior of steel containments near failure, to be used in licensing decisions (1985).

3. Identification of those containment penetration seal and gasket designs most susceptible to leakage under severe accident conditions and how such leakage would grow (1985).

6.9 Fission Product Control

Technical reports dealing with:

1. An investigation of fission product chemical forms (1984).

2. Data base assessment and suggested experimental program for fission product removal in ESF systems (1984).

3. Background information for predicting effectiveness of ESF system fission product retention (1984).

4. A user manual on the code for the aerosol/particle capture in ice compartments (ICEDF) (1985).

5. A user manual on the code for the aerosol/particle capture in suppression pools (SPARC) (1985).

6.10 Risk Code Development

1. Updated codes for use in risk estimation and risk-reduction analyses for source term rulemaking, for plant-specific licensing casework requiring risk assessments (e.g., Indian Point, Limerick), and for standard plant licensing evaluations (e.g., GESSAR, Washington SP-90) (1985).

2. Improved MARCH and MATADOR codes to support future site-specific evaluations of emergency planning needs and more general siting evaluations (1985).

6.11 Accident Consequence and Risk Reevaluation

1. Improved estimates for assessing the costs associated with adopting alternative designs, safety features, and operating procedures on both existing plants and plants still in design stage (1984).

2. Source term rule and associated regulatory implementation (1986) based on:

   a. Up-to-date evaluations of likelihood of severe accidents (1984, 1985).

   b. Up-to-date evaluations of consequences of severe accidents (1984, 1985).

c. Integration of accident likelihoods and consequences into statement-of-knowledge predictions (1985).

d. Evaluation of cost effectiveness of various means to reduce risk (1985).

6.12 <u>Risk Reduction and Cost Analysis</u>

1. Section 50.62 of 10 CFR Part 50 on reduction of risk posed by accidents involving anticipated transients without scram (ATWS) (1984).†

2. Proposed § 50.62 to Part 50 on additional requirements for Westinghouse reactors for accidents involving ATWS (1985).†

†The regulatory product involves rulemaking action.

7. RADIATION PROTECTION AND HEALTH EFFECTS

7.1 Metabolism and Internal Dosimetry

1. Two technical reports giving values for the gastrointestinal absorption factor for plutonium in mice, rats, and dogs (1985) and for thorium and neptunium in mice (1985), to be used in revising 10 CFR Part 20.†

2. A technical report on the metabolic parameters and dose/effect relationships for inhaled mixed (U, Pu) oxides, to be used in revising 10 CFR Part 20 (1985).†

7.2 Health Effects and Risk Estimation

1. A technical report on the genetic effects of fission neutrons and gamma rays, to be used in setting limits for exposure to neutrons in 10 CFR Part 20 (1985).

2. A technical report on dose-rate amelioration of chronic exposure to gamma rays for developing uniform health risk estimates, to be used for regulations, regulatory guides, policy statements, and news releases (1985).

3. A technical report on use of thermoluminescent dosimeters for teletherapy calibration, to be used in revising 10 CFR Part 35 (1985).†

7.3 Occupational Radiation Protection*

1. External Dose Control

   a. Two proposed rules on accreditation of personnel dosimetry processors and on industrial radiography audits and surveys (1984).†

   b. Four technical reports on optimization (1984) and dose reduction (1985) at nuclear power plants.

   c. Three technical reports on photon spectra and ALARA (1984) and on occupational exposures (1985) at nuclear power plants.

2. Internal Dose Control

   a. Two technical reports (1984) and one technical report (1985) on bioassay for internally deposited radionuclides.

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†The regulatory product involves rulemaking action.
*The technical reports listed in this section are being used as technical support documents for regulatory guides or as information reports to aid licensees in complying with regulations.

b. A technical report on personal air samplers for radioactive material (1985).

c. Two technical reports on respiratory protection (1984, 1985).

3. Training

   a. Two technical reports on health physics training at nuclear power plants and uranium mills (1984).

   b. Two technical reports on health physics training at medical facilities and uranium conversion and fuel fabrication plants (1985).

4. Instrumentation

   a. A draft regulatory guide on the calibration of health physics instruments (1984).

   b. A technical report on beta particle dosimetry and dose rate measurements (1984).

   c. A technical report on dose rate measurements at specified tissue depths (1985).

5. Three technical reports evaluating processes for dealing with chelating agents and for producing acceptable decontamination waste disposal forms (1984, 1985).

6. Technical reports assessing decontamination projects, including waste handling at selected nuclear power plants with regard to overall effectiveness and safety (1985).

8. WASTE MANAGEMENT

8.1 High-Level Waste

1. a. Proposed and final amendments to 10 CFR Part 60 related to disposal of HLW in the unsaturated zone (1985).†

   b. Proposed and final procedural amendments to 10 CFR Part 60 related to NWPA (1985).†

   c. Advanced Notice of Proposed Rulemaking on definition of high-level waste (1985).

   d. Revision 1 to Regulatory Guide 4.17, which provides standard format and content guidance for site characterization plans (1985).

2. Technical reports containing:

   a. Assessment of techniques for determining ground-water flow rate to provide a basis for evaluating data provided by the licensee on ground-water flow rates (1984).

   b. State of the art in HLW geochemistry research to provide the licensing staff with a comprehensive reference on current geochemical issues for use in licensee reviews (1984).

   c. Definition of relationship between gas conductivity and geochemistry of natural fracture for unsaturated zone characterization for use in assessing the importance of gas conductivity at fractured rock sites with respect to radionuclide transport (1984).

   d. Model for assessing degration of borosilicate glass waste forms based on solubility, to be used by licensing staff to evaluate waste form performance against 10 CFR Part 60 performance requirements (1985).

   e. Technical considerations of HLW disposal in unsaturated zone to provide a technical evaluation of issues with respect to proposed amendments to 10 CFR Part 60 on disposal in the unsaturated zone (1985).

   f. Model to simulate three-dimensional fluid flow and contaminant transport through a rock fracture system for use in performance assessment evaluations during license reviews (1985).

†The regulatory product involves rulemaking action.

g. Demonstration of computational methodology for assessing HLW repository performance in basalt to shown how the methodology could be applied to a basalt site during licensing review (1985).

h. Preliminary evaluation of the importance of coupled processes to evaluation of repository system performance (1985).

i. Evaluation of reliability of rock mass sealing technology for granite and basalt to assess DOE plans for sealing repository shafts and boreholes (1985).

j. Evaluation of performance characteristics of container and waste package materials as identified by DOE, to be used by licensing staff in evaluating DOE plans and designs to meet NRC containment requirements (1985).

k. Evaluation of geochemical phenomena important to the modeling of radionuclide transport, to be factored into the performance assessment methodologies to provide improved tools for review of license applications (1985).

3. Technical document to support licensing position on preliminary assessing of radionuclide transport as vapor through unsaturated fractured rock (1984).

8.2 <u>Low-level Waste</u>

1. Technical reports containing assessment of:

a. Effects of bioavailability of reactor LLW resulting from chemical changes in waste during transport through soil, to be used in assessing the need for stabilization of waste in engineered waste forms and in carrying out performance assessment of disposal systems (1984).

b. Field sampling designs and compositing schemes for cost-effective detection of spills and migration at contaminated sites (1984).

c. New York Nuclear Service Center in West Valley, New York, both geologically and hydrologically, to provide assistance to NMSS in its continuing role at West Valley with regard to understanding the geology and hydrology of the site and how they determine the transport of radionuclides from the site (1984).

d. Methods to ensure trench cap stability, the results to be used in evaluating operating and closure plans of LLW disposal sites against long-term performance requirements (1985).

e. Source term for migration, the results to be used in long-term performance assessment of LLW sites (1985).

f. Radionuclide migration through soils, the results to be used in predicting such movement as well as evaluating the predictions (1985).

g. Unsaturated zone hydrology and transport of tritium through plant transpiration stream at Maxey Flats, to be used to provide assistance at Maxey Flats and to provide insight into the movement of tritium through plants for use in evaluating any LLW site (1985).

h. Chemical characteristics of migrating radionuclides at commercial shallow land burial sites for use in providing information for the performance assessment of LLW disposal sites (1985).

i. Methods for extrapolation of short-term laboratory data into long-term in situ performance for waste forms and high integrity container materials (1985).

j. Modeling predictions compared to field validation studies, to provide for the licensing staff a model that has been field tested and is ready for staff use (1985).

9. FINAL PRODUCTS FROM COMPLETED ELEMENTS OF PAST LRRPs

Several elements included in past LRRPs are no longer part of the long-range planning because the research programs have essentially been completed. However, regulatory products are still being produced. These elements, together with their regulatory products, are provided in this section.

9.1 Safeguards

1. Revisions to eight regulatory guides dealing with measurement methods for material control and accounting (1984).

2. Ten technical reports on plant physical protection (1984).

3. Two technical reports on material control and accounting (1984).

9.2 Meteorology

1. Four technical reports dealing with tornado climatology and hazard probability assessments for use in PRAs and licensing decisions (1984).

2. Four technical reports concerning the atmospheric dispersion field test at Idaho National Engineering Laboratory and the results of real-time dispersion model validation studies for use in staff reviews of licensee emergency response and accident assessment capabilities and in selection or use of models that may be installed at NRC Operations Center (1985).

3. A technical report on an improved lightning strike climatology for the United States for use in PRAs and licensing decisions (1984).

4. Six technical reports on atmospheric dispersion and deposition related to modeling, laboratory measurements, and variations of dispersion with height and around building clusters for use in staff reviews of licensee emergency response, accident assessment capabilities, PRAs, and other licensing assessments and in the selection and use of models that may be installed at NRC Operations Center (1984, 1985).

9.3 Hydrology

Two technical reports, one describing ground-water mitigative methods and interdictive strategies for severe accidents at nuclear power plants for use in licensing decisions and emergency preparedness and the other on mathematical modeling of ultimate heat sink cooling pond performance for use in licensing assessments (1985).

9.4 Transportation Safety

Design criteria, brittle fracture criteria, and fabrication criteria for nodular iron shipping containers; the brittle fracture criteria for use in reviewing a full-scale drop test proposal for nodular iron shipping containers (1984).

9.5 Uranium Recovery

A 5-year program of research supporting the regulation of uranium milling operations was brought to an orderly close in 1984-1985, the results having been published in the form of 57 technical reports to make a comprehensive technical support base for the review of management of uranium mill tailings, the development of regulatory guides, and the implementation or modification of 10 CFR Part 40.

Table B-1

SUMMARY SHEET
LRRP INDICATIONS OF REGULATION CHANGES
AS RESULT OF RESEARCH CONDUCTED IN 1986 AND BEYOND

LRRP Section	Subject of Research	Targeted Date	LRRP Page
1.1	Reactor Vessels - PTS-validated fracture analysis methodology	1986	1-1
1.1	Reactor Vessels - PTS-fracture toughness and crack arrest toughness of irradiated vessel steel and weld metal	1986	1-1
1.3	Piping - validated analysis methodology for loading capacity of flawed and degraded piping; validation of leak-before-break concept; and data on true failure modes of cracked piping	1987	1-7
1.3	Piping - evaluation of aging and degradation in LWR materials	1989	1-8
2.1	Qualification of Electrical Equipment for Harsh Environments - evaluation criteria for environmental qualification testing of safety-related electrical equipment	1988	2-1
2.2	Qualification of Mechanical Equipment (Environmental) - evaluation of proposed methods of qualifying equipment for design basis events	1987	2-7
2.3	Dynamic Qualification of Equipment - evaluation of proposed methods of qualifying equipment for design basis events	1987	2-8
3.1	Seismic Hazard - data concerning seismic zones in Eastern U.S.	1988	3-1
3.1	Seismic Hazard - information base for developing site-specific spectra	1988	3-1

LRRP Section	Subject of Research	Targeted Date	LRRP Page
3.1	Seismic Hazard - methods for handling uncertainties in assessing potential risk from seismic hazards	1989	3-2
6.4	Hydrogen Generation and Control - data from all areas of hydrogen research such as generation, ignition conditions, and mixing	1986	6-7
7.1	Metabolism and Internal Dosimetry - reduction of uncertainties in data on metabolic behavior of transuranic elements	1987	7-2
8.2	Low-Level Waste - capability to assess alternatives to shallow-land burial of LLW	1988	8-6

Table B-2

FUNDING AND STAFFING FOR BASE YEAR FY 1986
(Dollars in Millions)

		FY 1986	Staff
1.	OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR	24.2	33
1.1	Reactor Vessels	8.6	
1.2	Steam Generators	1.4	
1.3	Piping	6.9	
1.4	Electrical and Mechanical Components	5.5	
1.5	Nondestructive Examination	1.8	
2.	EQUIPMENT QUALIFICATION	5.8	20
2.1	Qualification of Electrical Equipment for Harsh Environments	3.8	
2.2	Qualification of Mechanical Equipment (Environmental)	2.0	
2.3	Dynamic Qualification of Equipment	0	
3.	SEISMIC RESEARCH	10.0	21
3.1	Seismic Hazard	4.5	
3.2	Seismic Risk	5.5	
3.3	Seismic Margins		
4.	REACTOR OPERATIONS AND RISK	13.0	47
4.1	Reliability and Risk Methodology	4.0	
4.2	Data Base Development and Evaluation	2.1	
4.3	Regulatory and Inspection Applications	6.9	
5.	THERMAL-HYDRAULIC TRANSIENTS	20.9	19
5.1	Separate Effects Experiments and Model Development	6.5	
5.2	Integral Systems Experiments	7.3	
5.3	Code Assessment and Application	5.7	
5.4	Plant Analyzer and Data Bank	1.3	
6.	SEVERE ACCIDENTS	39.8	34
6.1	Accident Likelihood Evaluation	1.4	
6.2	Severe Accident Sequence Analysis	3.5	
6.3	Behavior of Damaged Fuel	11.3	
6.4	Hydrogen Generation and Control	1.6	
6.5	Fuel-Structure Interaction	1.4	
6.6	Containment Analysis	3.8	

Table B-2 (Continued)

		FY 1986	Staff
	6.7 Fission Product Release and Transport............	10.0	
	6.8 Containment Failure Mode.........................	4.9	
	6.9 Fission Product Control..........................	0.5	
	6.10 Risk Code Development...........................	0.3	
	6.11 Accident Consequence and Risk Reevaluation.......	0.7	
	6.12 Risk Reduction and Cost Analysis................	0.4	
7.	RADIATION PROTECTION AND HEALTH EFFECTS...............	2.5	16
	7.1 Metabolism and Internal Dosimetry................	0.4	
	7.2 Health Effects and Risk Estimation...............	1.3	
	7.3 Occupational Radiation Protection................	0.8	
8.	WASTE MANAGEMENT......................................	4.8	18
	8.1 High-Level Waste.................................	3.0	
	8.2 Low-Level Waste..................................	1.8	
		$121.0	208

Glossary

ACRONYMS AND INITIALISMS

ACRR	Annular Core Research Reactor
ACRS	Advisory Committee on Reactor Safeguards
AE	Acoustic emission
AEOD	(Office of) Analysis and Evaluation of Operational Data
ALARA	As low as is reasonably achievable
AOT	Anticipated operational transient
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated transient without scram
BNL	Brookhaven National Laboratory
BOP	Balance of plant
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
BWR	Boiling water reactor
CCFL	Concurrent flow limitation
CCTF	Cylindrical Core Test Facility
CEA	Commissariat à l`Energie Atomique, France
CFR	Code of Federal Regulations
CHF	Critical heat flux
CIRRPC	Committee on Interagency Radiation Research and Policy Coordination
COMMIX	Thermal-hydraulics code
CONTAIN	Containment analysis code

CORCON	Code to model interaction between molten core materials and concrete during core-melt accidents
CORRAL	Code to model behavior of fission products in containment atmosphere
CORSOR	Code to calculate core source term for severe accidents in light-water reactors
CP	Construction permit
CRAC	Code to calculate consequences of reactor accidents
CRAY	Type of computer
DBA	Design basis accident
DESRA	Code used by the U.S. Corps of Engineers to calculate seismic ground motion in loosely consolidated soils
DOE	Department of Energy
ECC	Emergency core cooling
ECCS	Emergency core cooling system
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
FAA	Federal Aviation Agency
FASTGRASS	Code to model fission product release from fuel
FEMA	Federal Emergency Management Agency
FIST	Full Integral Simulation Test
FITS	Fully Instrumented Test Series
GDC	General design criterion
HDR	Heissdampfreaktor (a decommissioned steam reactor in West Germany where reactor safety experiments are performed)
HLW	High-level waste

HPI	High-pressure injection
I&C	Instrumentation and control
ICEDF	Ice condenser system decontamination factor code
ICE S/E	Instrumentation, control, and electrical systems and equipment
ICRP	International Commission on Radiological Protection
IDCOR	Industry Degraded Core (Program)
IE	(Office of) Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular stress corrosion cracking
INEL	Idaho National Engineering Laboratory
IREP	Interim Reliability Evaluation Program
ISAP	Intergrated Safety Assessment Program
ISI	Inservice inspection
IST	Integral System Test
JAERI	Japanese Atomic Energy Research Institute
LCO	Limiting condition for operation
LER	Licensee event report
LET	Linear energy transfer
LLW	Low-level waste
LMFBR	Liquid-metal-cooled fast-breeder reactor
LOBI	Loop blowdown investigation facility in Italy where PWR physical phenomena and parameters that affect plant performance during small-break accidents are studied and computer models developed
LOCA	Loss-of-coolant accident
LOFT	Loss-of-Fluid Test
LRRP	Long-Range Research Plan

LWR	Light-water reactor
MAAP	Modular Accident Analysis Program (IDCOR's severe accident systems code)
MARCH	Code to analyze core meltdown phenomena
MATADOR	Code to model fission product behavior in LWR containments (replaces CORRAL code)
MELCOR	Code to model meltdown accident assessment (will replace MARCH, CRAC-2, and MATADOR codes)
MELPROG	Melt progression code
MIST	Multiloop Integral System Test
MIT	Massachusetts Institute of Technology
NASA	National Aeronautics and Space Administration
NDE	Nondestructive examination
NIH	National Institutes of Health
NPAR	Nuclear plant aging research
NRR	(Office of) Nuclear Reactor Regulation
NRU	Test reactor at Chalk River, Ontario (natural uranium, heavy-water moderated and cooled)
OECD	Organization for Economic Cooperation and Development
OL	Operating license
ORNL	Oak Ridge National Laboratory
OTIS	Once-Through Integral System
PBF	Power Burst Facility
PISC	Program for Inspection of Steel Components
PKL	Small-scale integrated test facility in West Germany
PMF	Probable maximum flood
PMH	Probable maximum hurricane

PPG	Policy and Planning Guidance
PRA	Probabilistic risk assessment
PTS	Pressurized thermal shock
PWR	Pressurized water reactor
RAP	Reliability Assurance Program
RELAP	Detailed model for thermal-hydraulic behavior in reactor coolant system during transient and loss-of-coolant accidents
RES	(Office of Nuclear Regulatory) Research
RMIEP	Risk Methodology Integration and Evaluation Program
ROSA	Rig of Safety Assessment (facility in Japan)
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Applications Program
SAFT-UT	Synthetic aperture focusing technique for ultrasonic testing
SAR	Safety analysis report
SARP	Severe Accident Research Program
SASA	Severe Accident Sequence Analysis
SAW	Submerged arc weld
SCC	Stress corrosion cracking
SCDAP	Severe Core Damage Analysis Package
SCTF	Slab Core Test Facility
SDMP	Seismic Design Margins Program
SEP	Systematic Evaluation Program
SFD	Severe fuel damage
SGTR	Steam generator tube rupture
SMAW	Shielded metal arc weld
SNL	Sandia National Laboratories
SPARC	Suppression Pool Aerosol Removal Code

SRP	Standard Review Plan
SSE	Safe shutdown earthquake
SSI	Soil-structure interaction
SSMRP	Seismic Safety Margin Research Program
TMI	Three Mile Island
TRAC	Code of model core reflood and quenching
TRAP-MELT	Code to analyze fission product behavior within LWR primary system under accident conditions up to and including fuel meltdown
UHI	Upper-head injection
UPI	Upper-plenum injection
UPTF	Upper Plenum Test Facility
USI	Unresolved safety issue
VANESA	Code to compute aerosol generation and fission product release during core debris interaction with concrete
VGES	Variable Geometry Experimental System
VICTORIA	Code to describe fission product release from fuel and transport in reactor coolant system
WPT	Wide Plate Test

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16. ABSTRACT (200 words or less)

The Long-Range Research Plan (LRRP) was prepared by the Office of Nuclear Regulatory Research (RES) to assist the NRC in coordinating its long-range research planning with the short-range budget cycles. The LRRP lays out programmatic approaches for research to help resolve regulatory issues. The plan will be updated annually.

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