NUREG-1080
Vol. 1

Long-Range Research Plan

FY 1985-FY 1989

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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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. Staff should not engage in research merely to postpone tackling difficult regulatory issues.

3. The highest priority for NRC research efforts will be light water reactor safety. Staff should be prepared to evaluate and explore the inherent safety characteristics of new reactor types.

4. The severe accident research program must be supportive of 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. 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. Advanced reactor concepts shall be pursued consistent with programs adopted by the Executive Branch, the Congress or a focused private sector effort alone or in combination with the government. Special emphasis should be given to the HTGR.

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*This is the section on research in NUREG-0885, Issue 3, "Policy and Planning Guidance, 1984."

3. 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. Research undertaken by the staff will be consistent with the long-range research plan.

4. The staff should continue to provide to the Commission an annual report which lists regulations likely to be substantively modified or substantiated by the research programs. Target dates for review of these regulations and the completion of changes to them should be specified. The particular research programs that relate to each of these regulations should be identified. Any remaining research programs should be listed along with a brief explanation of their purpose. Resources allocated to the latter category should also be provided.

5. 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 industry.

6. The staff will: (1) conduct annual assessments, with input from appropriate user offices, of the progress and usefulness of specific research topics; (2) consider the marginal utility of additional research given the licensing status of plants; and (3) make greater use of Research Review Groups.

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. Some of these broad issues are designated as Unresolved Safety Issues or TMI Action Plan Items. Any of these that are referenced in the LRRP are listed in Appendix A with the sections containing the reference identified. An overview of the LRRP in graphic form is included in Appendix C. 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 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.

RES has established general program priorities by ordering the research areas covered by this plan using the analysis described in Appendix B. Based on this analysis, the following issues appear to be the most pressing:

1. Severe Accidents (Chapter 6). The regulatory issue is the realistic treatment of reactor accidents and fission product behavior. A significant research effort will be applied during the next few years to support the reassessment of the regulatory treatment of severe accidents, i.e., the loads on the containment resulting from release of energy and substances such as hydrogen and the fission product release and transport. One of the objectives is a realistic treatment of fission product behavior to provide assurance that the real problem is being addressed. Work on this issue includes studies of transients leading to fuel or cladding damage, the behavior of damaged fuel, fuel melt, fission product release and transport, and concepts for mitigating severe accidents.

2. Risk Analysis (Chapter 4). The regulatory issue is the introduction and continuing use of probabilistic risk assessment (PRA) as a decisionmaking tool in the licensing review process and in the regulatory review process. In the licensing review process, PRA is used to place in perspective new and recurring unresolved safety issues and generic safety issues in addition to its direct use in licensing decisions. In the regulatory review process, PRA is used as part of the regulatory analysis accompanying changes in NRC regulations, guides, and standards and for setting reactor safety research priorities.

3. Human Factors (Chapter 4). The safety issue is the application of human factors engineering in the design, operation, and maintenance of nuclear facilities. Included in this program is work on control room design and

evaluation criteria, personnel qualification and staffing, management and organizational criteria, plant procedures, human reliability, and emergency preparedness.

4. Operating Reactor Inspection, Maintenance, and Repair (Chapter 1). The safety issue is the regulation of operating nuclear power plants as they become older to ensure that they continue to meet health and safety requirements. The work is directed toward an understanding of the mechanisms of aging and degradation and toward methods for examining and testing to determine the condition of components. A significant part of this work is concerned with the effect of radiation-induced embrittlement on the structural integrity of reactor vessels in support of work on the pressurized thermal shock issue.

5. Equipment Qualification (Chapter 2). The safety issued is the adequacy of the methods used for qualifying equipment used in nuclear power plants. Of concern are 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 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.

The proposed funding levels for the major research program areas described in the LRRP for Fiscal Years 1985-1989 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.

*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

LONG-RANGE RESEARCH PLAN

FUNDING LEVELS*

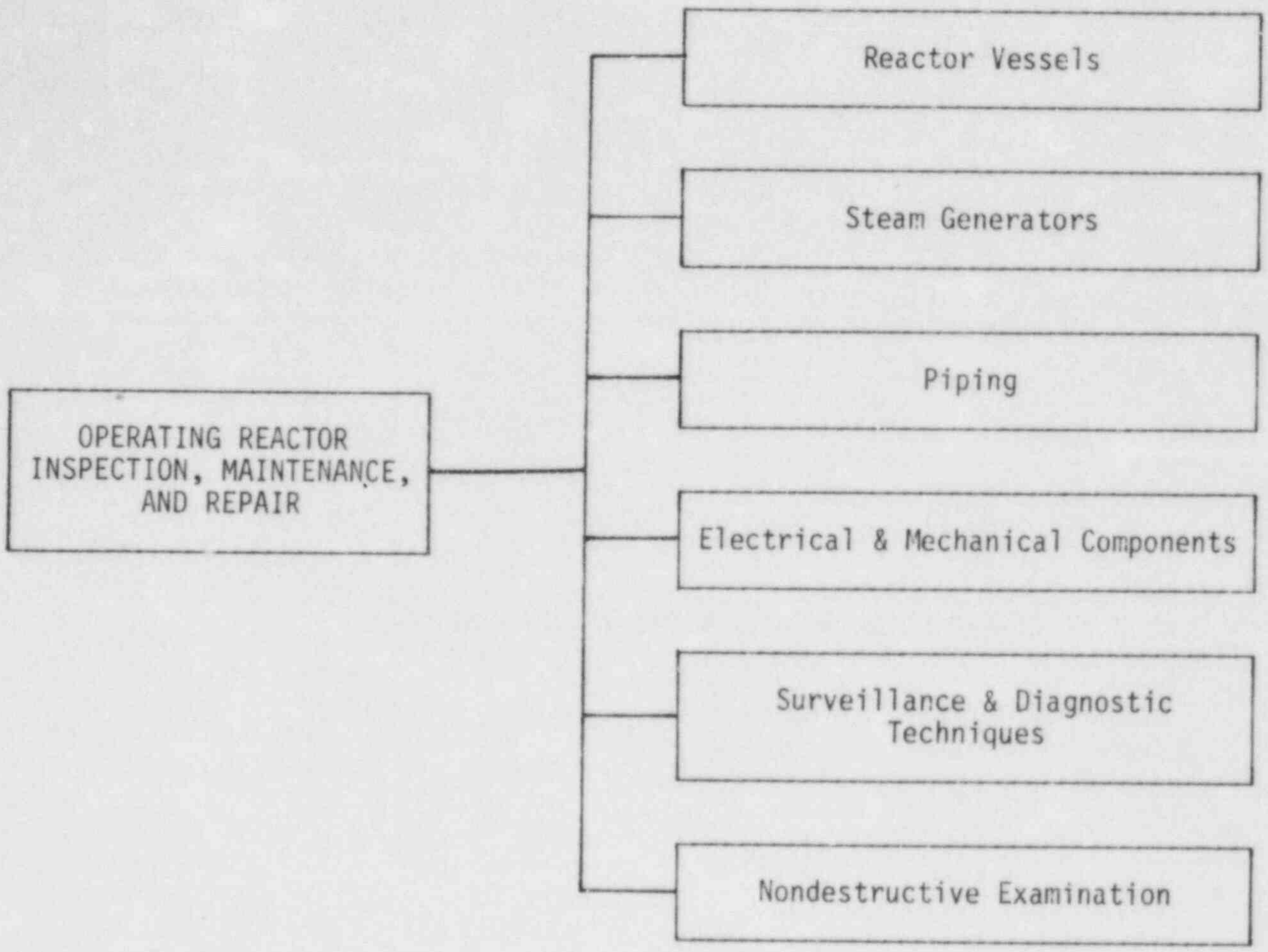
FY 1985-FY 1989

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR	$ 24.4	$ 30.1	$ 29.0	$ 30.0	$ 31.0
EQUIPMENT QUALIFICATION	5.6	6.8	8.4	9.0	8.0
SEISMIC RESEARCH	10.0	12.9	18.0	18.0	17.0
REACTOR OPERATIONS AND RISK	15.5	20.7	17.0	16.0	15.0
THERMAL-HYDRAULIC TRANSIENTS	27.3	30.3	43.3	39.5	37.8
SEVERE ACCIDENTS	50.9	48.3	48.5	35.0	30.0
ADVANCED CONCEPTS	5.1	5.1	5.1	5.1	5.1
RADIATION PROTECTION AND HEALTH EFFECTS	2.8	3.6	2.8	2.8	2.8
WASTE MANAGEMENT	9.4	10.8	13.0	14.0	14.0
TOTAL	$151.0	$168.6	$185.1	$169.4	$160.7

*Dollars in millions.

1

Operating Reactor Inspection, Maintenance, and Repair



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR	$24.4	$30.1	$29.0	$30.0	$31.0

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.

3. Experimentally validated dosimetry methods to allow accurate predictions of neutron fluence 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 (1986).

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 one 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. The research will reduce the error band in this calculation methodology 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 in situ 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 in situ 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 essentially completed by the end of FY 1985 with the longer, confirmatory research reaching completion by FY 1988.

The major research products will be:

1. a. Unified elastic and elastic-plastic fracture mechanics analysis procedure for licensing evaluations of pressurized thermal shock in plants (1985).

   b. Large-scale verification of unified methodology by thermal shock and pressurized thermal shock experiments, PTSE-2 (1985).

   c. Completion and experimental validation of analytical model for prediction of potential for initiation, propagation, and arrest of finite geometry ("real") flaws under postulated pressurized overcooling scenarios with consideration of cladding effects completed (1985).

d. Validation, by large-scale tests, of unified fracture mechanics methodology for licensing evaluation of pressurized thermal shock (1985).

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

f. Complete development of unirradiated fracture toughness data base for new, high-strength materials proposed for next generation reactor pressure vessels (1989).

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

   b. Confirmation of K_{Ic} curve for present-practice steels in ASME Section XI (1985).

   c. Technical basis for proposed licensing criteria and standards for in situ annealing of commercial reactor vessels (1985).

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

   e. Irradiated specimen test completed to validate ASME Section XI fracture toughness curves (1986).

3. 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 and support column embrittlement predictions (1985). (Also applies to Needs 1 and 2.)

4. a. Revision of fatigue curves for ferritic materials in ASME Section III (1986).

   b. Final revisions recommended for the environmentally assisted fatigue crack growth curves in ASME Section XI (1987).

   c. Revisions recommended for ASME Section XI fracture toughness curves for newer higher-strength materials (1988).

   d. Technical bases for revising 10 CFR Part 50 governing reactor fracture toughness requirements under both normal and accident operating conditions (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 errors must be reduced, and signals must be carefully correlated to exact defect 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 1.6.) 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 (1985). (Also applies to Need 2.)

   b. Validation of current and advanced NDE results through examination of removed tubes (1985). (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 (1985).

b. Demonstration of generator cleaning and decontamination as basis for action on licensing applications (1986).

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 plan (1986).†
   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 guillotine break 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 (1985-1987).
   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 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 (1986).
   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 cast duplex (austenitic and ferrite) stainless steels from long-time aging at reactor operating temperature, to be the basis for developing a regulatory guide and revising the standard review plan (1987).
   Justification: A certain level of toughness is required to ensure safety in piping and other primary system components, especially to resist failure if flaws snould develop in service and under accident loading. Long-term time-at-temperature can cause the ferrite in the duplex stainless steel to transform to another phase 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 pipe whip restraints and asymmetric loss-of-coolant accidents (LOCAs) and load combinations for reactor coolant loop piping, components, and supports (1986).
   Justification: The understanding of the reliability of reactor coolant loop piping of each United States vendor must be improved.

7. Confirmation of licensing criteria for evaluating pipe impact resulting from pipe rupture and the analytical capability to predict the results of pipe-to-pipe impact (1985).
   Justification: Current acceptance criteria are based on engineering judgment and logic, and overly conservative designs may have been the result. This condition may not be best for normal operation.

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 FY 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.

Pipe rupture investigations deal with probabilistic assessments of directly and indirectly induced double-ended guillotine breaks in PWR and BWR reactor coolant loop piping. Direct mechanisms depend on advanced fracture mechanics evaluations and modeling intergranular stress corrosion cracking. Indirect mechanisms include evaluations of component support failures and other failures that in turn lead to pipe ruptures. Another aspect of the pipe rupture studies explores the behavior of whipping pipes that impact on adjacent pipes. Both theoretical studies, involving finite-element computer codes, and experimental investigations, involving full-scale impacts of pressurized prototypical piping, support the research on whipping pipes.

The major research products will be:

1. a. Evaluation of pipe cracking predictive models, proposed fixes, and weld repair criteria (1985). (Also applies to Needs 2 and 3.)

   b. Initial findings on toughness of cast stainless steels for use in leak-before-break study (1985). (Also applies to Need 4.)

   c. Data and conclusion regarding the effectiveness of short-term fixes for pipe cracking (1985). (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 (1986).

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 (1987).

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 (1985).

   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 (1987).

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

5. a. Evaluation of the effect of hydrogen on stress corrosion cracking (SCC) of piping materials, of thermal aging on ferritic materials, and of irradiation on SCC of piping and internal materials (1987).

   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. Information regarding need for pipe whip restraints and jet impingement shields (1985).

   b. Information regarding the postulation of asymmetric LOCA for piping in the safety assessment (1986).

   c. Information regarding the combining of pipe rupture with earthquakes in the design basis (1985).

   d. 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 (1985).

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

7. Summary report on pipe-to-pipe impact studies (1985).

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) establish a data base for conducting a comprehensive aging assessment of critical components and (2) develop the technical basis for monitoring the condition of components so that significant functional degradation can be anticipated and corrected before major safety problems occur. 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 (1985-1987). 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 (1987-1989).

   Justification: 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. Licensing guidance on dry interim spent fuel storage prior to its ultimate disposition (1989).
   Justification: The National Waste Policy Act of 1982 requires that interim spent fuel storage and monitored retrievable storage provided by either the utility or DOE shall be licensed.

5. 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 (1989).
   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.

This is a relatively new research program, the initial phase of which will provide inputs 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:

- o 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.
- o 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.
- o 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.

The major research products will be:

1. a. General methodology for comprehensive assessment of aging of electrical and mechanical components (1985).

   b. 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 (1985).

   c. 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) (1985-1987).

2. a. Identification of practical and cost-effective techniques for monitoring equipment or service wear effects for selected components (1985).

   b. Assessment of effectiveness of surveillance monitoring in supplementing artificial pre-aging (prior to qualification testing) for selected electrical equipment (1986) and mechanical equipment (1987).

   c. Assessment of necessity to modify surveillance/testing intervals in plant technical specifications to account for age (1987).

   d. Assessment of methodologies for predicting service lifetime of equipment (1988).

   e. Evaluation criteria for surveillance, maintenance, and replacement programs for selected components (1988).

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 bases on spent fuel conditions that will occur during the dry interim storage of spent fuel (1988).

5. a. Information on actual decommissioning of nuclear facilities to be used to develop updated reports on the technology, safety, and costs of decommissioning reactors, spent fuel storage facilities, and nonfuel-cycle facilities (1988).

   b. Information on decommissioning facilitation to be used to develop an upgraded data base (1988).

1.5 Surveillance and Diagnostic Techniques

This element provides research needed to evaluate surveillance and diagnostic techniques to help prevent undesirable plant transients or accidents and to help avoid damage to equipment important to safety in reactor systems.

1.5.1 Major Regulatory Needs and Their Justifications

1. Capability for early detection and diagnosis of nuclear power plant operating anomalies such as early indication of impending failure of mechanical equipment or a potential for fuel cladding breach. Such indication will help (1) prevent damage to equipment and systems important to safety or avoid unnecessary radiation exposure of personnel and (2) prevent or mitigate accidents. This research will provide a technical basis for licensing criteria and regulatory actions for such capability (1988).
   Justification: Surveillance and diagnostic techniques and instrumentation (e.g., noise diagnostics) can provide warning of impending failures or malfunctions with possible impairment of safety so that early corrective action can be taken to prevent those failures. Research is necessary to assess the need for and the practicality of such techniques.

2. Evaluation of the effectiveness of current surveillance test frequencies of engineered-safety-feature actuation systems (ESFAS) and reactor trip systems (RTS) in ensuring availability of these systems important to safety and development of a methodology for arriving at optimum test frequencies, for use in the revision of Regulatory Guide 1.118 (1988). (See Section 1.4 on the aging of electrical and mechanical components, a program closely coordinated with this need.)
   Justification: Frequent testing of ESFAS and RTS equipment could increase the probability of introducing errors or causing component degradation.

1.5.2 Research Program Description

The strategy for this research is to evaluate various surveillance and diagnostic techniques and alternative means of diagnosing undesired events so that potentially harmful plant transients, equipment damage, and accidents can be avoided or their consequences minimized. Included in this research are evaluations of continuous on-line surveillance and diagnostic systems, diagnostic instrumentation, ESFAS and RTS test frequencies, and response time testing.

An evaluation of ESFAS and RTS test frequencies will be performed in an effort to develop a methodology for arriving at optimum ESFAS and RTS test frequencies.

On-line testing techniques for important instrument performance characteristics (e.g., response time) will be examined to ensure their adequacy.

The major research products will be:

1. Surveillance and diagnostic techniques using reactor neutron, pressure, and temperature noise signals for early detection and diagnosis of nuclear power plant anomalies (1987).

2. Methodology for arriving at optimum test frequencies for ESFAS and RTS (1986).

1.6 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 1985-1987 period.

1.6.1 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, to contribute to the resolution of USI A-14, "Nondestructive Examination," and 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 contribute to the resolution of USI A-14 and 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.

1.6.2 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 B&PV 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 accuracy 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 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 1.2. Here, accurate knowledge of the flaw sizes and of the extent of degradation measured in tubes taken from the generator provides the ultimate means to ensure calibration of the inspection method. Continuous monitoring to detect the onset of cracking or leakage through use of acoustic emission is validated by tests on large-scale pressure vessel burst tests and by studies on loops in 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 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 accuracy.

A most important use of the research results is as the basis for criteria for qualification of personnel, equipment, and methods, especially for ultrasonic test inspection of piping and other primary system components. It is because of the insights gained into the effects on detection and evaluation accuracy 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 B&PV Code, Section XI, requirements for ultrasonic inspection of vessel plate and forging, to be used to improve methods for through-weld and stainless steel inspection (1985).

   b. Completion of tests of current and advanced eddy current NDE methods, using retired steam generator (1985).

   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 thick sections, welds, and multimetal joints (1985-1986).

   d. Code acceptance for the improved SAFT-UT method for flaw evaluation and detection, and for continuous acoustic emission (AE) leak monitoring (1985-1986). (Also applies to Need 2.)

   e. Improved inspection plan for implementation in licensing actions for inservice inspection of steam generator tubing (1985-1986).

   f. Recommendations for ASME Code acceptance of new and improved methods for ultrasonic and eddy current inservice inspections and for continuous AE monitoring (1985-1987). (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).

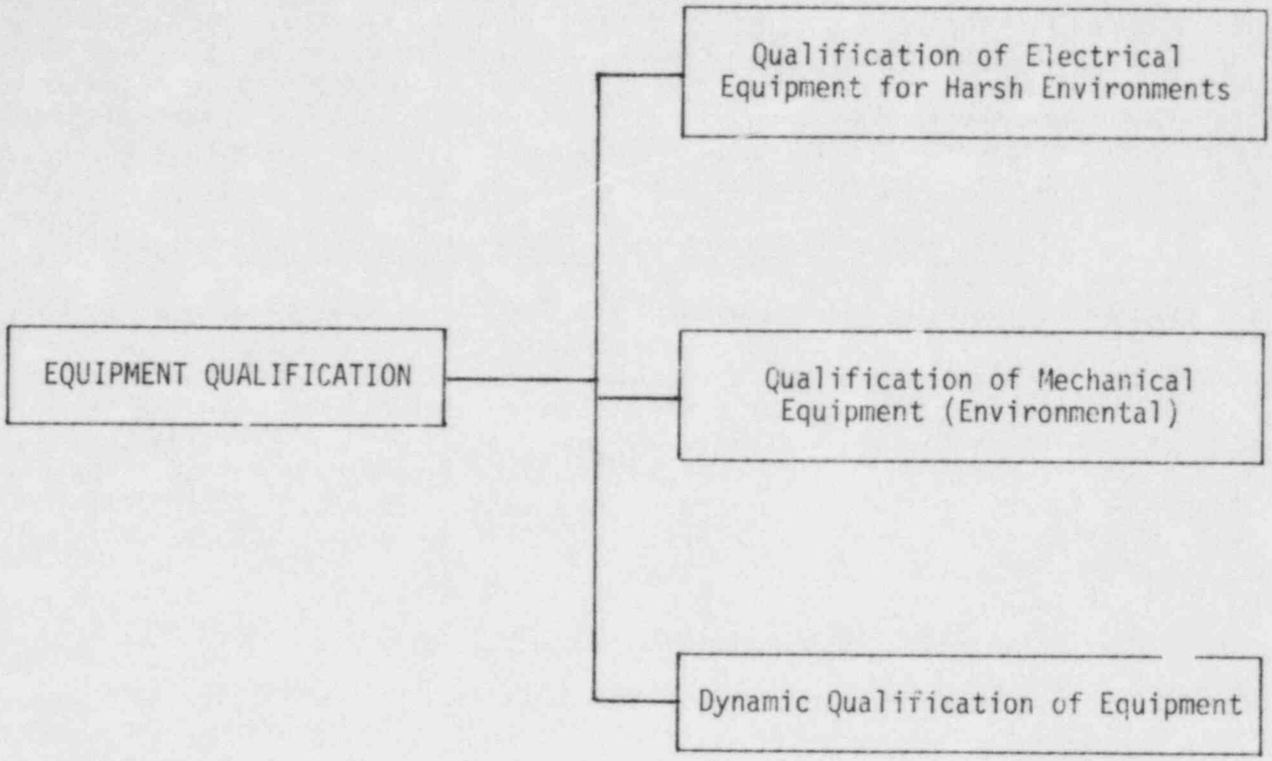
   h. Code acceptance of recommendations for improved inservice and continuous monitoring inspections (1986-1988). (Also applies to Need 2.)

   i. Capability in ultrasonic testing, acoustic emission, eddy current, and related NDE methods for prompt reaction to inspection problems (1985-1989). (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 (1985).

2

Equipment Qualification



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
EQUIPMENT QUALIFICATION	$5.6	$6.8	$8.4	$9.0	$8.0

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 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 will be described in a 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).
<u>Justification</u>: 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 used to simulate the end-of-life condition to permit qualification testing on a shorter time span. 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 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 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).
<u>Justification</u>: 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 further joint arrangements are under consideration with the Japanese Atomic Energy Research Institute (JAERI) and German Gesellschaft für Reaktorsicherheit (GRS) establishments. The cost of this research is thus being minimized where possible by international cooperative agreements.

4. Identification of the potential failure or degradation modes that can result from practices used in the design, manufacture, installation,

operation, periodic testing, and maintenance of instrumentation and control (I&C) and electrical equipment important to safety (1985).
Justification: Regulatory review of Class 1E I&C and electrical power system component qualification includes assurance that the severity of the qualification methods equal or exceed the maximum anticipated service requirements and conditions and that any extrapolation or inference be justified by allowances for known potential failure modes and the mechanism leading to these failure modes. The research is needed to identify the potential failure or degradation modes that can result from practices used in the design, manufacture, installation, operation, periodic testing, and maintenance of the components and to identify the means by which these practices can be modified to minimize the failure or degradation modes.

5. Identification of the potential failure or degradation modes that can result from practices used in the design, manufacture, installation, operation, periodic testing, and maintenance of heating, ventilating, and air conditioning (HVAC) equipment important to ensuring control room habitability, to be the basis for regulatory guidance on the design, manufacture, installation, operation, periodic testing, and maintenance of control room HVAC equipment (1985).
Justification: The Advisory Committee on Reactor Safeguards (ACRS) has expressed concern that improvements in control room habitability are needed for safety reasons. Among the ACRS concerns was the need for a testing protocol and for acceptance criteria for control room HVAC systems based on conditions that permit continuing equipment functionability and human comfort during prolonged emergency situations. An NRR task force is assessing changes needed in staff and licensee practices related to the control room habitability safety concerns expressed by the ACRS. The performance of control room HVAC equipment under accident conditions needs to be evaluated in support of the task force review.

6. Assessment of the use of solid-state motor controllers in nuclear power plants from a safety standpoint, to be the basis for regulatory decisions on the use of solid-state motor controllers (1988).
Justification: Solid-state motor controllers are being used in fossil energy plants and may be proposed for use in nuclear power plants. The use of solid-state motor controllers in nuclear power plants needs to be investigated to determine if their use would be acceptable from a safety standpoint.

7. Development of the technical basis for regulatory guidance on 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 (1985).
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 safety functions (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.

8. Experimental data base to quantify the safety margins inherent in the current fire protection criteria, to be used in resolving safety issues arising out of the staff's experience in performing fire protection reviews (1989).
   Justification: In addition to assisting in the resolution of licensing reviews, the data base will be valuable in developing probabilistic fire risk assessment modeling, described in Section 4.1.1. This program will reduce the uncertainties in the assessment of safety margins by quantifying the characteristics of credible fires, by defining damage thresholds of equipment, and by determining the fire environment in the vicinity of the safety equipment.

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 between breaking bonds by 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. Samples of a large number of polymers are being tested in a joint program with the French CEA using the LICA facility at Sandia for aging and the CESAR LOCA test facility in France. The importance of the sequence of radiation aging and thermal aging is being tested. The sequential exposure of the aged polymers to accident radiation followed by LOCA steam versus simultaneous exposure to radiation and steam is also being studied in these tests. The results will provide an extensive data base on how different electrical equipment materials behave in an accident and will aid in determining the conservatism of various test sequences. 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 being pursued under cooperative agreements with the French CEA and the Japanese 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.

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. In addition, degradation and failure modes that can lead to decreased accuracy and response time will be identified for these components.

Control room HVAC equipment will be evaluated under accident conditions for equipment reliability and performance capability in maintaining control room habitability. Specifically, the HVAC equipment specification, design, manufacture, testing, installation, and maintenance procedures will be examined to determine failure modes and weaknesses, if any, that may affect control room habitability.

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.

Evaluations of the individual instruments, control, and electrical components will be performed to examine practices in design, manufacture, installation, operation, periodic testing, and maintenance that are needed to ensure proper operation when required. Individual generic I&C and electrical components that are used in a variety of systems will be evaluated first (e.g., pressure transducers used for measuring absolute pressure, differential pressure, level, and flow). Other components to be evaluated include solenoid- and motor-operated control valves.

The major research products will be:

1. a. Evaluation of methods for simulating accident sequence and aging in the qualification of electrical equipment (1985). (Also applies to Need 3.)

   b. Reassessment of the basis for determining equipment radiation levels in accidents and the adequacy of simulations (1985). (Also applies to Need 3.)

c. Acceptable methods for aging and accident simulation qualification tests of electric penetrations (1985), cables (1985), and electric motors (1986).

d. 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 (1985).

b. Understanding of the mechanisms of aging in polymers under normal and accident conditions (1985).

c. Criteria and a basis for regulatory requirements for aging in equipment qualification (1986).

3. a. Understanding of electrical polymer behavior during aging and accidents (1985). (Also applies to Need 1.)

b. Evaluation of importance of beta radiation and the gamma damage equivalence in accident simulation (1985). (Also applies to Need 1.)

c. Assessment of degradation of gasket and s i materials and lubricants in the radiation and environment accompanying a LOCA or severe accident (1985).

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

4. Recommended guidelines for design, manufacture, installation, operation, periodic testing, and maintenance of pressure transducers and solenoid- and motor-operated control valves (1985).

5. Recommended guidelines for design, manufacture, installation, operation, and periodic testing of HVAC equipment important to control room habitability during accident conditions (1985).

6. Criteria for the use of solid-state motor controllers (1987).

7. Technical basis for regulatory guidance for instrumentation, control, and electrical power systems important to safety but not safety related (1985).

8. a. Compilation of data on energy and mass-release characteristics of credible fires (1985).

b. Compilation of data on fire damage thresholds of safety equipment (1985-1986).

c. Definition of a set of design basis fires (1986).

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

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

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

g. 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 (1988).
   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 (1986).†
   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. a. Identification of significant environmental parameters (1985).

   b. Standard review plan to include criteria for environmental qualification (1986).

2. Acceptable methodology for environmental qualification of mechanical equipment (1987).

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/or 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 (1985).
   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 fragilities, to be the basis for ensuring the seismic capability of mechanical and electrical equipment in new and operating plants (1985).
   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 (1985).
   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 (1986).†
   Justification: Evaluation of methods of mechanical and electrical equipment qualification for dynamic (including environmental) 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.

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 (including environmental) 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 (including environmental) loads. The first step in this process is to identify what components should be considered as candidates for qualification testing and/or analysis in operating plants, plants under construction, and new standardized plants. Techniques such as probabilistic risk analyses will be used to identify those components and equipment where implementation of qualification procedures would reduce the overall risk to the public and indicate the margins necessary to account for uncertainties. Previous studies such as the Seismic Safety Margins Research Program and technical assistance studies sponsored by NRR will be used to the maximum extent in this effort. The next step will be to identify the dynamic (including environmental) loads that may be imposed on these components and equipment during normal operations and postulated accidents, including design basis events. Following this effort, those loads that are significant in establishing equipment qualification procedures will be evaluated to determine if the assumptions currently used are correct and if there are any synergistic effects when those loads are combined with other mechanical loads and environmental loads defined in Section 2.2.

The adequacy of existing qualification procedures set forth in effective and draft national consensus standards will be evaluated. Changes to these standards and the need for additional standards will be identified. In order to evaluate qualification procedures, the following technical issues will be addressed.

Valve and pump operability will be evaluated through experimental programs and will include closing torque requirements, leak integrity under accident conditions, flow-induced vibrations, and scaling effects. This experimental effort will, in part, provide information regarding containment leak integrity to the severe accident research program. Efforts are being made to establish a cooperative agreement with Japanese agencies to obtain detailed test results from their pump and valve experimental programs.

Dynamic (including seismic) loads will be characterized and evaluated in terms of how they should be simulated in a qualification test. This will include evaluation of the input spectra, wave form (sinusoidal, random, etc.); need for multiaxial testing; and duration and amplitude factors (margin) to account for uncertainties and degradation. Experience data on equipment similar to that found in a nuclear plant that has been installed in facilities that have experienced large earthquakes will also be examined. This effort will be in cooperation with the Seismic Qualification Utilities Group (SQUG) that is sponsoring a detailed study of large facilities that have undergone large earthquakes. Additionally, test data from commercial qualification laboratories will be used to the extent that proprietary limitations will allow.

The sequence and combination of loads to be applied during qualification testing will be investigated. The objective of this effort will be to identify the simplest test sequence and load combinations necessary to demonstrate qualification.

In addition, the ability to test small-sized equipment and components and to extrapolate the results to larger sizes will be evaluated. Extrapolation limits will be established along with guidelines for analytical evaluation in lieu of testing.

Experimental programs 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. Another example is the use of the HDR facility in the Federal Republic of Germany. 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.

The major research products will be:

1. a. Identification of significant loads or combinations of loads (1985).

   b. Criteria for qualification by test (1986).

2. Establishment of a data base of equipment responses, failure modes, and fragilities to determine safety margins available in existing equipment (1986).

3. Evaluation of safety margins available in existing mechanical and electrical equipment subjected to the SSE (1986).

4. Acceptable methodology for dynamic qualification of mechanical equipment (1987).

5. Evaluation of criteria to extrapolate test results from one-sized component to another (1987).

Seismic Research

SEISMIC RESEARCH

FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
SEISMIC RESEARCH	$10.0	$12.9	$18.0	$18.0	$17.0

3. SEISMIC RESEARCH

Earthquakes can be the most severe of the natural hazards faced by nuclear power plants. In order to assess seismic risks and to establish appropriate regulatory requirements, it is necessary to determine the seismic hazard (earthquake magnitude and occurrence intervals) for a power plant site and to predict the response of the site and the facility to earthquakes within the range of magnitude appropriate to the site. Current estimates of seismic risk not only contain large uncertainties that stem primarily from a lack of earthquake records, failure experience, and fragility test data but also include significant difficulties associated with analytically modeling such phenomena as soil-structure interaction and response of piping systems to seismic motions.

3.1 Major Regulatory Needs and Their Justifications

1. a. Data concerning seismic source zones in the Eastern United States, including Charleston, New Madrid, New England, and others, 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 major faults and associated stress fields that drive the faults are poorly known.

   b. An information base for the development of site-specific spectra, to be the basis for amendments to Appendix A to 10 CFR Part 100 and for developing a regulatory guide (1987).†
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 States. Analysis of these records will address important regulatory questions concerning the interpretation of this type of record and its use in licensing decisions.

   c. Methods for handling the uncertainties in assessing the potential risk from seismic hazards, 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.

2. a. 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.

   b. Verification of methods for predicting seismic soil settlement and soil-structure interaction (1985)
Justification: The consequences of soil settlement, liquefaction, or other soil failures at high earthquake levels (above Safe Shutdown Earthquake (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. The interaction between soil and buildings has an important effect on inplant seismic response. Although several analytic techniques have been developed for soil-structure interaction, the limitations of these techniques and their associated uncertainties have not been quantified.

3. Data to predict the nonlinear response and failure modes of nuclear shear wall structures (1985-1989).
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.

4. An improved seismic fragility data base for mechanical and electrical equipment (1987).
Justification: The fragility data used in current PRAs rely heavily on expert opinion and thus are subject to large modeling uncertainties. An increased knowledge of equipment failure modes and levels is needed as a basis for equipment qualification decisionmaking and for standards leading to more balanced plant design.

5. Validation of current seismic PRA methods (1985-1989).
Justification: The Seismic Safety Margin Research Program (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.

6. Assessment of the margins inherent in the seismic design of structures, piping, and equipment in older nuclear power plants (plants designed

prior to the mid-1970s), and quantification of seismic design margins using simplified approaches in terms of risks associated with seismic events beyond the design basis (1986).
Justification: The most urgent NRR need for seismic research is to assess the margins inherent in the seismic design of older plants. 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.

3.2 Research Program Description

3.2.1 Seismic Hazard

Uncertainty in seismic hazard analysis is the 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. A probabilistic sensitivity study is also under way to develop methods to better deal with uncertainties in seismic hazard analysis.

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, and studies of recent crustal movements. 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 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 major research products will be:

1. a. 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 (1985-1988); geophysical data for determining crustal structure in areas of suspicious geologic structures (1985); and data from the in situ stress measurement program in the Northeastern United States (1986).

b. Techniques for calculating site-specific response spectra (1986).

c. Seismic hazard characterization and probabilistic sensitivity study of probable ground-motion dependence on the various proposed causes of seismicity in the Eastern United States (1985).

3.2.2 Soil Failure and Soil-Structure Interaction

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 benchmarking of the Structural Engineering Computer Codes Program at Brookhaven National Laboratory is currently comparing and evaluating soil-structure-interaction methods using physical data, including response records from the Fukushima plant, EPRI's SIMQUAKE test results, and Department of Defense test data. The CLASS I Code, used by the SSMRP, will be validated in this study.

The major research products will be:

2. a. Workshop on soil liquefaction (1984).

   b. Validation of the DESRA code and recommendations regarding the use and limitations of soil-structure-interaction computer codes (1985); determination of the seismic risk contribution from dam and embankment failure (1986).

3.2.3 Structural Response and Fragility

The Seismic Category I structures program will provide experimental data on shear wall structures subjected to earthquake loadings that will cause linear and nonlinear behavior. Sensitivities of structural behavior and changes in amplified response spectra and damping, resulting from variations in structural configuration, design practice, input motion, etc., will be determined as the models are loaded from the linear to the nonlinear range. Failure modes and levels will be determined for each configuration.

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.

The major research products will be:

3. a. Determination of the changes in floor response spectra and damping for input motion resulting in linear and nonlinear behavior (1985-1988).

b. Initial failure modes and failure level data for various concrete configurations (1985).

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

3.2.4 Equipment Response and Fragility

The NRC piping research program is addressing a number of issues for the purpose of finding ways to increase overall piping reliability. The nature and consequence of seismic loading on piping is being investigated in the load combinations and stiff versus flexible piping programs and in pipe damping studies.

Information regarding seismic failure modes and levels for equipment will come as an offshoot of the equipment qualification program. Additional data will likely be needed and will be obtained through a separate data acquisition and test program. The SSMRP Zion analysis and sensitivity studies will serve to set priorities in establishing research directed at reducing uncertainties in this program's fragility data bases.

The major research products will be:

4. a. Risk-based recommendations for decoupling SSE and LOCA loads in primary piping design criteria (1985).

   b. Recommendations regarding the possible removal of pipe whip restraints and jet shields (1985).

   c. Fragility data from equipment testing (1985-1986).

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

   e. Risk comparison between stiff and flexible piping analysis (1986).

3.2.5 Seismic Risk Validation

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 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 will be pursued.

In the past, the inability to correlate input spectra with damage experience and the multidisciplinary aspects of seismic risk evaluation made it necessary to approach SSMRP validation through addressing separately the various modeling and calculational steps. The SSMRP sensitivity studies will provide an identification of important contributors to seismic risk that will be used in setting validation priorities.

The techniques and data used for predicting the seismic hazard at sites will be validated through the seismotectonic program (Section 3.2.1). 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's SMACS computer code (and the data bases and modeling assumptions used by it) will be partially validated by the response and fragility research outlined in Sections 3.2.2 through 3.2.4. Additional research will be performed to validate the unique probabilistic and interdependent nature of the structural evaluation. An integrated program plan to address this is being developed.

The Risk Methodology Integration and Evaluation Program (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 major research product will be:

5. RMIEP's evaluation of the SSMRP's system analysis methods (1985).

3.2.6 Quantification of Seismic Margins

During 1984, a plan to address the generic method and its associated uncertainies needed to quantify plant-specific seismic design margins will be developed. The plan will include techniques to quantify margins with respect to (1) soil-structure interactions, (2) response and failure modes for buildings, and (3) response and failure modes for piping and equipment. The scope of this program as presently envisioned will include doing the following:

- Identify the failure conditions assumed in the old design process. Compare these conditions to current knowledge on seismic failure experience and fragility data, and quantify the differences.

- Quantify the significant safety factors used in older designs, i.e., the required margins between annual failure conditions and acceptable design limits.

- Evaluate the differences between old and new improved seismic design methods. Identify those differences that need to be corrected in the old methods, and quantify this effect on seismic design.

- Quantify the margin inherent in the old design. Identify and quantify the uncertainties associated with the determination of the margin.

- Analyze selected PWR and BWR in terms of seismic margin to failure. Use realistic analysis to establish the seismicity level that is expected to cause core melt. Quantify the uncertainties associated with the prediction.

- Estimate the probability of occurrences of a seismic event that would be expected to cause core melt. Evaluate the uncertainties associated with this estimate.

The major research products will be:

6. a. Comparison of quantified differences between failure conditions assumed using old methods with those that can be derived from current knowledge on seismic failure experience and fragility data (1986).

   b. Quantification of significant safety factors used in older designs (1986).

   c. Identification of those differences between old and new seismic design methods that need to be corrected and quantification of their effect on seismic design (1986).

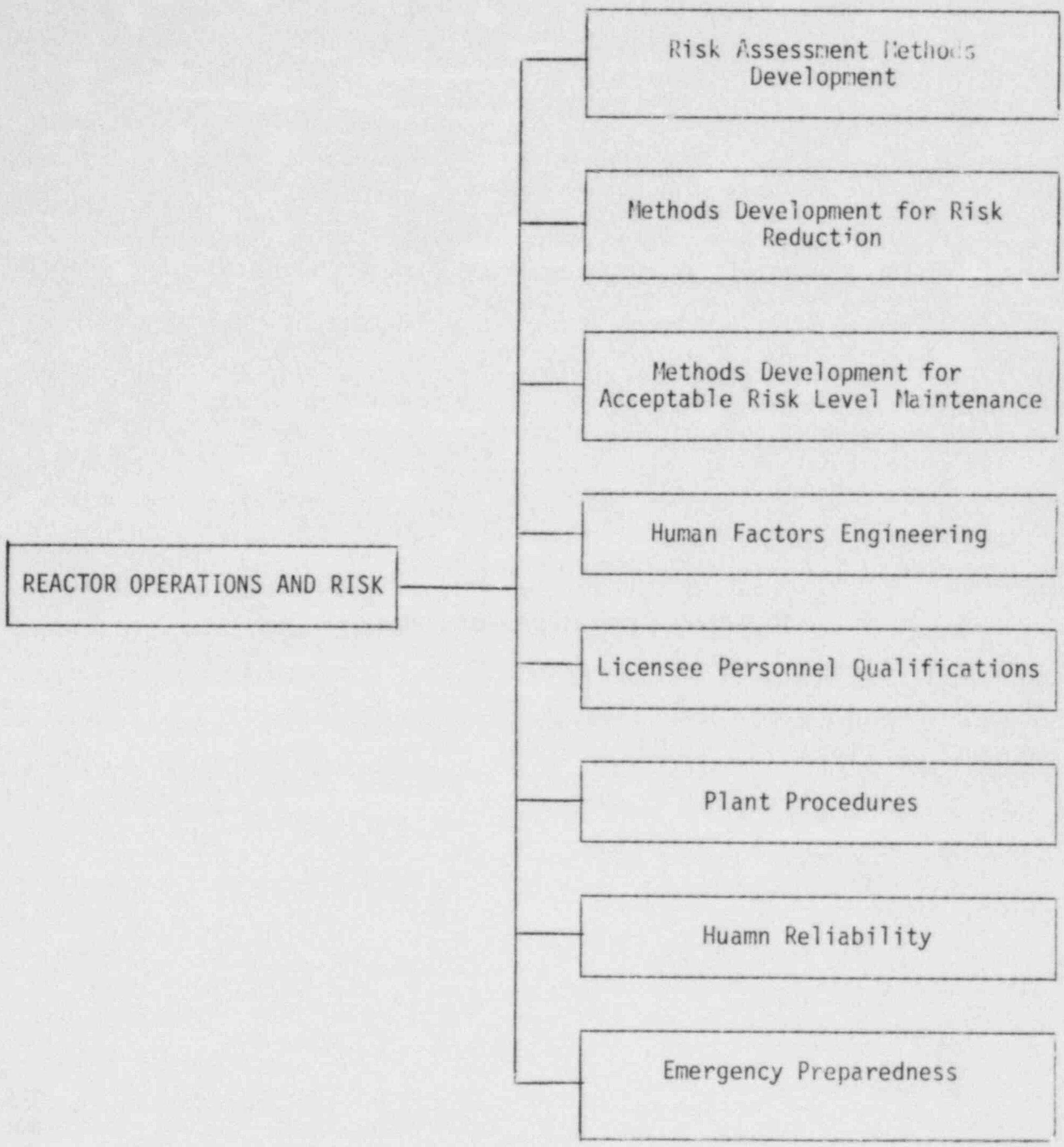
   d. Quantification of the inherent seismic margins with associated uncertainties in the older designs (1986).

   e. Computation of the seismic risk of a selected PWR and a BWR (1985).

   f. Estimation of the probability of occurrence of a seismic event that would cause core melt together with an evaluation of associated uncertainties (1986).

Reactor Operations and Risk

4



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
REACTOR OPERATIONS AND RISK	$15.5	$20.7	$17.0	$16.0	$15.0

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. 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. As a result, one of the major focuses of this program is the identification of opportunities for human error and of ways to improve plant reliability. 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 eight elements corresponding to the principal topical areas within this program, i.e., risk assessment methods development, methods development for risk reduction, methods for maintaining an acceptable level of risk, human factors engineering, licensee personnel qualifications, plant procedures, human reliability, and emergency preparedness.

4.1 Risk Assessment Methods Development

Research 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.

4.1.1 Major Regulatory Needs and Their Justification

1. Accident Likelihood

   a. Improved data base for estimating component and system failure rates, including harsh environment data, to support implementation of the Safety Goals and other licensing evaluations of significant safety issues (1985 and beyond as additional data become available).

   b. 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 (1985).

   c. 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 (1985-1989).

   d. Methods for quantifying the effects of severe natural phenomena (e.g., seismic activity (1985-1986) and external floods (1985-1987)), 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.

e. Methods to assess and display the quantitative uncertainties in PRA modeling and data in support of regulatory decisionmaking, including importance measures and sensitivity analyses (1985-1989).

f. Improved methodology for the quantitative prediction of human error likelihood, to support ISAP and other regulatory activities (1985-1986). (This effort will be coordinated with those described in Section 4.7.)

Justification of Above Needs: The requirement to develop quantitative assessments of the probabilities and consequences of severe reactor accidents is becoming an increasingly important element in the regulatory decisionmaking process. The Commission's Policy Statement on Safety Goals includes quantitative design objectives. Similarly, policies presently being considered by the Commission (SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation") and the proposed ISAP would 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 on 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) 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 in current PRA techniques that limit their usefulness. For example, a substantial amount of supporting research needs to be performed so that common-cause failures and human errors can be more comprehensively included in risk analyses. Considerable research is also required 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. Fire research is required to develop modeling approaches and data for fire risk analysis in all types of nuclear facilities. This research is coordinated with the fire protection research described in Section 2.1. In particular, this project will assist in determining the frequency of various fires, defining maximum credible fire sizes, and describing fire scenarios of interest. In turn, the research described in Section 2.1 will provide a data base for fire risk assessment modeling.

Additional refinements 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 research programs. Improved quantification of such uncertainties in likelihood and consequence prediction is needed to improve the use of PRA in NRC 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 is consistent with current Commission guidance concerning the evaluation and implementation of the Safety Goals and the ISAP. The basic strategy being followed in the program is first to establish a common data base for use in PRAs. Second, the program will provide detailed, documented methods to ensure consistency in PRA execution and will further augment existing techniques to account for new data. The third step calls for exercising and testing these methods in an integrated and realistic manner. Finally, improvements will be made in methods to facilitate the use of these techniques (i.e., by making them faster and simpler) by both NRC and the industry.

The program has been designed to provide information and analyses needed to support short-term, high-priority Commission decisions such as the need to revise the regulatory treatment of severe reactor accidents (see Chapter 6) and the proposed ISAP and longer-range activities required to support the evaluation and application of the Safety Goals to standard plants and new construction permit (CP) applications.

As related to reactor systems, the safety-safeguards interface aspects at nuclear generating stations and the effectiveness of alternative or improved safeguards systems and procedures will provide the technical bases needed for proposed rule changes and regulatory guidance on physical protection. Major research projects will include refinement of techniques for determining vital equipment in support of ongoing rulemaking on the insider threat (1985-1986); guidance on adequate audits of safeguards programs by licensees; guidance for developing, inplementing, and maintaining computer-managed safeguards systems (1985); guidance on postincident recovery following a successful sabotage (1985); and human factors considerations in the areas of the insider threat, organization, response capabilities, and safeguards equipment and facilities (man-machine interface) (1985-1989).

The major research products will be:

1. a. Component and system reliability data base developed from evaluations of plant operating data, licensee event reports (LERs), harsh environment data, and vendor information (1985 with annual revision to reflect most recent data and operating experience).

   b. Identification and review of accident sequences and their likelihood and related probabilities, including accident precursors, using improved methodology (1985).

   c. Development of a procedure to incorporate dependency analyses, including common-cause failures, into a PRA (1986).

   d. Procedures for incorporating results of simplified Seismic Safety Margin Research Program methodology (see Chapter 3) into PRA methodology (1986). (To be developed and applied for trial use in the Risk Methodology Integration and Evaluation Program (RMIEP)).

e. 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 (1985).

f. Human factors research results to be incorporated into PRA Procedures Guide (1986).

4.2 Methods Development for Risk Reduction

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.

4.2.1 Regulatory Needs and Their Justifications

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 (1985-1986).

2. Improved estimates for assessing 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 (1985).

3. Studies of the risk significance of pressurized thermal shock (PTS) to confirm the screening criteria in proposed § 50.61 of 10 CFR Part 50, to help develop detailed guidance for plant-specific analyses of PTS, and to help develop acceptance criteria for corrective measures (1985).

4. Documentation of uniform procedures for carrying out value/impact analyses for use by both NRC and the industry (1985).
   Justification of Above Needs: As a result of the TMI accident, the NRC identified as an important element (Task II.B.8, NUREG-0660) the need to conduct a long-term study to reassess the regulatory treatment of severe reactor accidents. In this study, the Commission would consider whether current LWR designs and operating procedures need to be changed to explicitly consider the likelihood and possible consequences of severe reactor accidents.

   In addition, NRC must decide whether to require augmented safety features on those plants being reviewed under the proposed ISAP and how the benefit-cost guidelines of the Safety Goals are to be applied in future licensing actions. Also, NUREG-0058 documents the policy that value/impact analyses will be provided with all proposed changes in the regulations.

   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.

5. Determination of the adequacy of current standards for certain types of radioactive shipments for preventing potential high-consequence transport accidents, to be the basis for amendments to 10 CFR Part 71 and updating Environmental Statement on Transportation of Radioactive Material (NUREG-0170) (1985).†
Justification: During the next several years, the number of shipments of spent fuel and high-level radioactive wastes is expected to dramatically increase. Investigation of current radioactive materials package performance standards is needed to ensure that they technically and perceptually provide adequate protection against potential accidents associated with transportation of "high-hazard" radioactive materials.

4.2.2 Research Program Description

This program will establish methods for the analysis of the risk-reduction effectiveness of generic safety system modifications to support decisions on pressurized thermal shock, severe accident rulemaking (see Section 6.13), and other 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 (1985-1986).

2. Evaluation of feasibility of using PRA to improve reliability of existing plant systems (1985-1986). (Also applies to Need 1.)

3. Evaluation of the likelihood of PTS causing a through-wall crack in a reactor pressure vessel at three selected plants (1985).

4. Establishment of standardized procedures for the application of PRA results and value/impact analyses to the decisionmaking process for proposed rules, guides, etc. (1985).

5. Modal studies to define forces generated in severe accidents and predicted response of generic spent fuel shipping containers to these accident forces (1985).

4.3 Methods Development for Acceptable Risk Level Maintenance

This element describes the research programs that apply to the development 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 and to provide the technical basis for future Commission actions relative to operating plants.

4.3.1 Major Regulatory Needs and Their Justification

1. Establishment of a working group of experts to guide the development and execution of the reliability assurance program (RAP) (1985).

2. Demonstration of the feasibility, usefulness, and cost effectiveness of the selected reliability assurance methodology by applying it to two nuclear power plants, a PWR and a BWR (1986).

3. 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).

   Justification of Above Needs: 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.

   Specific needs to develop RAP elements applicable to commercial power reactors are described in the TMI Action Plan, the Indian Point Hearing Board Recommendations of October 24, 1983, the Safety Goal Evaluation Plan (SECY-83-428), the Salem Task Force Report (NUREG-1000), and the ATWS (anticipated transient without scram) rule Statement of Considerations.

4.3.2 Research Program Description

This program will develop the necessary management structure, procedures, methods, and requirements, using information gained from aerospace 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. In 1984, elements that appear effective for operating reactors will be developed based on a screening evaluation of practices in other industries. Development of the program will be guided by a working group of experts from other agencies (NASA, FAA, Navy, etc.), industry (EPRI, etc.), and NRC (IE, the regions, etc.) similar to the group that led to the PRA Procedures Guide (NUREG/CR-2300). In 1985-1986, a realistic trial application of the most promising elements will be conducted (on both a PWR and a BWR) to demonstrate the prospects for cost effectiveness, institutional compatibility, and technical coverage of potential reliability problems by 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 major research products will be:

1. Recommendations for a reliability assurance program to be made by working group (1985).

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

3. 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 starting in 1985).

4.4 Human Factors Engineering

The research provides the technical bases needed by NRC to evaluate the man-machine relationships at information and control stations and in control rooms, to assess and recommend human factors standards and guidelines for new or improved designs affecting the operator or maintainer, and to establish criteria for regulatory applications of human factors engineering. The information and data will be derived from empirical studies in laboratories and in field settings and from analysis and evaluation of relevant data sources. The ultimate objective is to better ensure the safe operation and maintenance of nuclear facilities and thereby enhance the safety of the public. The research program includes the development of methods and information requirements to assess and alleviate the effects of severe stress on operator performance.

4.4.1 Major Regulatory Needs and Their Justifications

1. Systematic data base reflecting human behaviors in a control room, and other manned stations in nuclear facilities, and the integration and use of engineering analyses and results of these analyses that reflect procedures, other task analyses, and severe accident sequence analysis (SASA) tasks (see Section 6.2), to be used to support regulatory actions related to human factors engineering, personnel staffing, qualifications of licensed personnel, training, procedures, job aids, and communications networks and to develop objective measures of individual operator and crew performance in nuclear power plant operations (1985-1989).
   Justification: This research was developed to establish safety requirements and to identify criteria and guidelines for the application of human factors engineering to licensed nuclear facilities. The NRC Human Factors

Program Plan (NUREG-0985), as approved by the Commission, identified near-term (1983-1985) human factor requirements. These near-term requirements explicitly imposed longer-range research tasks, which are discussed in this LRRP. Additionally, NRR requirements in "Functional Criteria for Emergency Response Facilities," NUREG-0696; "Guidelines for Control Room Design Reviews," NUREG-0700; "Clarification of the TMI Action Plan Requirements," Item I, NUREG-0737; and "Guidelines for the Preparation of Emergency Operating Procedures," NUREG-0899, directly and indirectly state the need for human factors engineering data and information that cut across several licensing areas. Objective measures of performance are necessary to determine the effectiveness of operator and crew training and examinations and to relate these to the control room design, shift crew operations, and safety-related procedures. A broad technical data base is necessary to evaluate man-machine systems and their contribution to safety in nuclear facilities. This effort provides information that supports the resolution of USI A-17 relative to human aspects of systems interactions.

2. Technical basis for advanced computer-based man-machine systems such as Safety Parameter Display System (SPDS) and for possible future systems to develop needed evaluation criteria and design guidelines for such equipment and systems as they come into use, to support regulatory actions in the man-machine topics of interest and to determine the acceptability of placing new designs in existing control rooms (1985-1989).
   Justification: The design characteristics of computer-based information and control systems, including artificial-intelligence-based display systems, affect the capability of the operator to safely use the large information retrieval and display features. The consequences of poor human factors engineering designs are excessive information overload, lack of operator control on automatic and semiautomatic operations, and a mismatch between machine and human functions and capabilities. This work will support near-term regulatory needs described in the NRC Human Factors Program Plan (NUREG-0985) for evaluation of computer-based data systems in current control rooms and will provide a technical basis for such man-machine systems in advanced reactors and in the fuel cycle and waste management facilities. Adequate consideration of human factors in new designs reduces risk associated with human errors of commission and omission. This effort will provide information that supports the resolution of A-49 regarding operator aids needed to address pressurized thermal shock conditions.

3. Assessment of the effects of a severe stress on human performance and decisionmaking and the identification of requirements for information displays and response capabilities, special training, and dedicated procedures and job performance aids, to provide the basis for regulatory actions that will ensure adequate operator actions under such conditions (1985-1987).
   Justification: This research was requested by the ACRS and supported by the Commission in their FY 1984 budget allocation to provide a systematic human-factor-oriented technical basis for regulatory decisions related to severe seismic events and other high-stress situations, e.g., toxic materials from onsite or offsite events, for personnel; man-machine interfaces;

and associated specialized training needs. This work will draw on other research projects as appropriate, including SASA (see Section 6.2) and SSMRP (see Chapter 3). In the near term, this research will address the ability of the operator to safely control the plant during and following a severe seismic event.

4. Evaluation of the suppressing of nuclear power plant alarms under anticipated accident sequences, to be the basis for regulatory guidance for the suppressing of alarms following anticipated accident sequences (1985). Justification: As a result of the TMI accident, the need to reduce the number of alarms presented to the operators became apparent. In order to determine the extent to which alarms can be suppressed, a typical plant alarm system, including operating procedures to be followed, should be evaluated for anticipated accident sequences to determine the set of least required alarms and their required timing.

4.4.2 Research Program Description

The strategy of the research is to develop the technical basis for evaluating the operator-machine interface of licensed nuclear facilities. Information and data to support changes to regulations (Appendix A to 10 CFR Part 50), regulatory guidelines, and standards will be the principal products of research. In addition, methods of analysis and techniques to evaluate man-machine systems that may be proposed by a licensee or applicant will be developed to assist the regulatory review processes. This research will be coordinated with the Institute of Nuclear Power Operations (INPO), EPRI, IEEE, and DOE.

The research program will emphasize an empirical approach to defining criteria and human performance. Analytical studies will be performed as needed to supplement the experiments. Both field studies and laboratory experiments will provide sources of data.

The major research products will be:

1. a. Comprehensive data base reflecting operator and crew behaviors in a variety of plant evolutions and accident sequences; this data base to be the result of the crew task analysis (1985).

   b. Effects of function allocation and automation on operator motivation, vigilance, and attitudes; assessment of need to preserve manual operation as backup to automatic system (1986). (Also applies to Need 2.)

   c. Evaluation of human factor design for maintainability and a data base to provide technical support for regulations and regulatory guides (date dependent on Maintenance Task Action Plan). (Also applies to Need 2.)

   d. Data and information base to help develop functional requirements and evaluation criteria for alarm filtering systems, disturbance analysis systems, and computerized procedures systems in the man-machine system (1988). (Also applies to Need 2.)

2. a. Guidelines for control room and display, control, and communication systems (1988).

   b. Guidelines for the use and human factor evaluation of artificial intelligence and other computer-based data display systems (1989).

3. Evaluation of effects of severe stress such as that due to seismic events and similar sources of stress on operations personnel; criteria to evaluate information display and control systems, procedures, effective decision-making, and training to be available (1988).

4. Guidance for advanced annunciator for improving the setting of priorities for alarms, including the suppression of lower-priority alarms (1985).

4.5 Licensee Personnel Qualifications

This element provides the research necessary to assess, develop, or confirm the technical basis for the guidance used by the NRC to establish and evaluate the qualifications of licensee personnel to safely operate and maintain a nuclear facility and reduce human-related risk. These qualifications include education, training, examination, experience, and requalification.

4.5.1 Major Regulatory Needs and Their Justifications

1. Evaluation of the effectiveness of implementing Section 306 of P.L. 97-425 as the basis for implementing regulatory decisions contained in proposed revisions to 10 CFR Part 55, new § 50.200, and proposed revisions to Regulatory Guides 1.8 and 1.149 (1985-1986).
   Justification: This work is needed to support the resolution of Section I.A.4.2 of the TMI Action Plan (NUREG-0660) and to provide the technical basis for implementing proposed regulatory decisions with regard to operator training (e.g., simulator capabilities, requiring the use of simulators in training programs, using simulators as examination tools, determining acceptable alternatives to simulator training, and the effectiveness of classroom instruction).

2. Determination of current status and staffing of nuclear power plant operations and nonlicensed support personnel (1985-1989).
   Justification: The current status of nuclear power plant licensed operators and other support personnel is needed to assess the effectiveness of regulatory actions and initiatives in this area.

3. Determination of the appropriate qualifications, training, examination, and licensing requirements for personnel at fuel cycle facilities, to be the basis for a regulatory guide (1988-1989).
   Justification: Historically, qualification, training, examination, and licensing requirements and guidance for personnel at fuel cycle facilities have been developed largely on the basis of best judgment. This research responds to Section IV.C of the TMI Action Plan (NUREG-0660) to extend the

lessons learned to other licensed activities. It will provide the technical basis necessary to assess the adequacy of the current NRC requirements and will determine whether there is a need for new requirements in this area.

4. Evaluation of the need for additional training for operators involved in the management (i.e., prevention, arrest, mitigation) of severe accidents with potential for severe core damage; development of operator training guidance for management of risk-dominant severe accident sequences (e.g., use of plant analyzers in conjunction with operator training on control room simulators), to be the basis for appropriate regulatory actions, e.g., a regulatory guide (1985-1989).
   Justification: This research is needed to provide a technical basis for criteria and guidance to be used by the regulatory staff to determine the adequacy of operator training for severe accident management.

5. Development of criteria and guidance for improving examination for individual job performance and for performance as team members as well as in overall plant operations, to provide a basis for regulatory decisions involving the contribution to crew performance beyond that due solely to individual training (1985-1987).
   Justification: This research is needed to provide a technical basis for regulatory decisions concerning team training in licensee training programs in order to ensure adequate crew performance of nuclear power plant personnel.

6. Determination of the value and impact of regulatory decisions (e.g., upgrade in qualification standards of nuclear power plant personnel job positions) that have consequences for manpower and staffing levels at nuclear power plants and development of manpower and staffing models to support regulatory analysis of regulatory actions (1989).
   Justification: This research will provide methods to assess the value and impact of regulatory actions addressing adequate manpower and staffing levels in nuclear power plants. Models are needed to quantify concerns such as an increase in job qualification standards decreasing the available pool of manpower upon which to staff nuclear power plant jobs.

7. Determination of training guidance for personnel involved with application of advanced technology such as robotics or remote control in high-risk environments (e.g., commercial plant operations or decommissioning inside containment, fuel cycle facility operations and maintenance) (1989).
   Justification: Use of advanced technology such as robotics for operations in high-risk environments at nuclear facilities may place different demands upon the skills, knowledge, and abilities of nuclear facility personnel. These special demands need to be identified to ensure that qualified personnel are available to operate equipment involving robotic technology.

4.5.2 Research Program Description

The strategy is to develop the technical basis for implementing changes to the regulations (e.g., 10 CFR Part 55 and § 50.200) and the regulatory guides (e.g., 1.8, 1.149) that establish the qualifications, certification, and training requirements and guidance for licensed and unlicensed operators and support personnel at nuclear power plants and fuel cycle facilities. This work will be coordinated with EPRI, INPO, and appropriate national standards efforts.

The research program will provide methods, information, and data for use in determining and validating the appropriate education, training, examination, and licensing requirements for operators and support personnel at nuclear facilities. Additionally, the operational capabilities of, use of, and requirements for full-scope, part-task, and concept-type training simulators will be established.

The major research products will be:

1. a. Empirical data on nuclear power plant operator performance from training simulator experiments (1985). (Also applies to Need 2.)

   b. Validated criteria and guidance for the qualifications, training, and examining of nuclear power plant operators (1987).

   c. Methods to evaluate the effectiveness of licensee programs to qualify nuclear power plant operators (1985).

2. Assessment of current qualification and training practices for operations and support personnel at fuel cycle facilities with respect to practices of other industries (1989).

3. Qualifications and training guidance for unlicensed operators and support personnel at nuclear power plants (1989).

4. Operator training criteria and guidance for severe accident management (1987).

5. Annual report on personnel status and the education, training, and experience of the staffs of nuclear power plants.

6. Manpower and staffing models for use in regulatory analysis (1989).

7. Personnel guidance for training in operation of equipment involving robotic technology (1989).

4.6 Plant Procedures

This element provides the research that is needed to develop the technical basis for the methods and criteria used to assess and upgrade, where needed, plant operating procedures necessary for safe operation of nuclear power plants, fuel cycle facilities, waste management facilities, and processors and users of special nuclear material and byproduct material. Information from PRA and SASA (see Section 6.2) efforts will be used to help focus this research on specific areas that can or do significantly impact risk. The plant procedures include emergency operating procedures, abnormal operating procedures, normal operating procedures, and surveillance, maintenance, and testing procedures. This element does not include administrative procedures for management of these facilities.

4.6.1 Major Regulatory Needs and Their Justifications

1. A technical basis for regulatory assessments regarding the adequacy and effectiveness of nuclear power plant emergency, abnormal (e.g., single-failure, common-mode-failure, and multiple-failure accident sequences), and normal operating, surveillance, maintenance, and testing procedures and for assessing operator performance and training by using upgraded procedures, to develop regulations and regulatory guides (1985-1988).†
   Justification: This work is needed to develop methods and provide a technical basis for criteria for regulatory use in the human factor evaluation of plant procedures and specifically to assess the technical soundness and adequacy of applying current and upgraded emergency operating procedures and other applicable procedures, the techniques and formats for presenting procedures, the readability and comprehensibility of procedures, and the impact of upgraded procedures on safe operation and operator training needs. The near-term research will concentrate on LWR plants.

2. Evaluation of alternative generic techniques and formats for presenting procedures (passive versus dynamic), including hard-copy printed page, computer-based CRT systems, liquid crystal display (LCD), and other advanced technology concepts (e.g., artificial intelligence (AI) interactive computer-based systems, voice-actuated systems, robotics interface), to provide a technical basis for regulatory assessment of this aspect of procedures (1986-1989).
   Justification: This research complements Section 4.4, "Human Factors Engineering," and supports resolution of Section I.C.9 of the TMI Action Plan (NUREG-0660). It will provide a technical basis to support regulatory assessments of current and anticipated procedure presentation practices of passive and dynamic media (e.g., hard-copy printed page, computer-based CRT display, and other advanced technology concepts interfacing with AI, robotics, voice actuated) for nuclear power plant operation. The information will also be used to develop regulatory positions and standards in this area.

3. A technical basis for regulatory human factors assessment of the adequacy and effectiveness of operating, surveillance, maintenance, and testing procedures for fuel cycle facilities, waste management facilities (low- and high-level radioactive wastes), and special nuclear material and

byproduct material processors and users, to be used in developing regulatory positions, criteria, guidelines, and regulations for assessing and upgrading such procedures (1987-1989)†

Justification: This research is responsive to Section IV.C of the TMI Action Plan (NUREG-0660) to extend the lessons learned to other licensed activities and will provide the technical basis for regulatory human factors decisions and standards regarding operating, surveillance, maintenance, and testing procedures for fuel cycle facilities, including fuel conversion, fuel fabrication, handling, storage, transportation, reprocessing, waste management facilities, and special nuclear material and byproduct material processors and users. The technology developed for nuclear power plant operations will be used where appropriate for this research.

4.6.2 Research Program Description

The strategy for the research is to establish an integrated systems approach for assessing the needs and determining the adequacy and effectiveness of the operating surveillance, maintenance, and testing procedures for nuclear power plants, fuel cycle facilities, waste management facilities, and special nuclear material and byproduct material processors and users. Methods to quantitatively evaluate facility procedures will be developed and tested against a data base. These findings will provide a technical basis to establish criteria for procedure assessment and for the development of appropriate regulatory positions, regulations, and guides.

The major research products will be:

1. Handbook for staff use in evaluating plant-specific emergency, abnormal, and normal operating, surveillance, maintenance, and testing procedures for PWRs and BWRs and its validation for usefulness and effectiveness (1985-1988).

2. a. Handbook for application and assessment of alternative techniques and formats for presenting procedures and its validation (1985-1987).

   b. Determination of need for and assessment of impact of computer diagnostics and automation on procedures and regulatory requirements (1986-1987).

3. Methods for evaluating operating, surveillance, maintenance, and testing procedures for waste management facilities (e.g., low- and high-level waste) for processing, handling, transporting, storage, and monitoring (1986-1989).

4.7 Human Reliability

This research involves analysis of nuclear power plant (NPP) operations and maintenance personnel errors and their contributions to man-machine safety system failures. Human error assessment methods and data (rates/probabilities) emerging from this research will support Section II.C of the TMI Action Plan (NUREG-0660) and NRC reliability evaluation programs, including PRA and complex man-machine safety systems design and evaluation.

4.7.1 Major Regulatory Needs and Their Justifications

1. Valid human error data and methodologies and techniques for qualitative and quantitative assessment of NPP operator and maintenance personnel reliability, especially for control room personnel, to determine their contribution to risk, for use in support of PRA (1985-1988).
   Justification: Human error in the operation and maintenance of safety systems and equipment has been identified as a major contributing factor to NPP unreliability and risk. Human reliability research is therefore directed toward the development of valid, reliable human error data (rates/probabilities) and techniques for applying these data to the human reliability analysis segment of PRAs and for developing insights as to what can be done to reduce/eliminate human error.

2. Human performance data banks (human error data acquisition, storage, and retrieval) and application of assessment methodologies to establish performance criteria for assessing the adequacy of current and advanced NPP man-machine safety systems for assisting in reducing human error (1985-1987).
   Justification: Human performance is a primary basis for assessing the utility of products developed under all human factors research, e.g., staffing, training, procedures, and organization and management. Human reliability research is directed toward the development of human performance criterion measures from available human error data sets to support rigorous evaluations of products emerging from other human factor research.

3. Guidelines developed through analysis and modeling of NPP operations and maintenance functions crucial to safety, for use in identifying additional human reliability research needs and for use as regulatory design requirements for advanced man-machine safety systems (1985-1989).
   Justification: Human error (omission, commission, extraneous acts, sequential, time) is of critical importance in identifying immediate and future human reliability research needs and in developing design requirements for advanced man-machine safety systems. A long-range goal of the human reliability program is the transformation of human error data into specific research needs and design guidelines for advanced man-machine systems so as to improve both plant safety and public safety.

4.7.2 Research Program Description

The strategy for the research is to develop a technical basis for supporting complete and accurate NPP human performance reliability analysis programs (e.g., PRA). Its objective is to develop (1) baseline human error probability statistics from data obtained from operating plants, nuclear power industry training simulators, performance modeling, and expert judgment; (2) a human reliability data bank for compiling, collating, and storing human error data from all the above media; (3) performance aids, e.g., handbook, workbook, human reliability models, to assist the PRA specialist in conducting human reliability analyses of NPP safety-related events; and (4) technical data for use in conducting near- and long-term human reliability research and in developing technical design criteria for advanced man-machine safety systems for NPPs. This research will be coordinated with INPO, EPRI, utilities, user groups, other Government agencies, and foreign countries.

The major research products will be:

1. a. Computer-based model for developing human error probability data for selected maintenance functions (1985).

   b. Probabilistic risk assessment specialist aids (e.g., handbook, workbook, analytic models) for analyzing a variety of crucial normal, transient, and accident precursor sequences involving human action (1987).

   c. Computer-based model for developing human error probability data for a variety of operations and maintenance functions (1987).

2. a. Validated human reliability data bank concept (1985).

   b. Human error probability data from operating plants, industry training simulators, and expert judgment to support a wide variety of operations and maintenance reliability analyses (1985-1987).

   c. Human reliability data bank implementation plan (1986).

   d. Human reliability data bank combining varied human error data acquisition media and automated storage and retrieval techniques (1986).

3. a. Human error data and criterion measures for evaluating the effectiveness of man-machine safety systems (1987-1988).

   b. Regulatory requirements and design criteria for advanced man-machine safety systems (1989).

   c. Guidelines for identifying human reliability research needs using human error data (1987).

4.8 Emergency Preparedness

The research program in emergency preparedness provides the technical basis for NRC regulatory actions needed to improve the capability of Federal, State, and local governmental authorities and licensees to mitigate the consequences of an accident at a nuclear facility.

4.8.1 Major Regulatory Needs and Their Justifications

1. 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 ability to predict accident progress (e.g., containment failure time) (1985-1986).
   Justification: 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.

2. Identification and verification of emergency action levels (plant indicators) at which various emergencies should be declared by examining the ongoing PRA and severe accident analysis efforts, to provide a basis for revisions to the current guidance (NUREG-0818) and inspection procedures for event classification (1985-1986).
   Justification: Appendix E to 10 CFR Part 50 requires that licensees and applicants establish emergency action levels (EALs) at which emergencies are declared. The results of the ongoing research on severe accident sequence identification need to be considered in the revisions to the current guidance and criteria used to develop and evaluate EALs.

3. Improved instrumentation, sampling techniques, and procedures to measure airborne radioiodine concentrations in the field when high concentrations of radioactive noble gases are present and a technique to quickly assess radioiodine doses through the milk pathway from field measurements. In addition, studies of portable instruments to measure other radionuclides identified as important by the new accident source term work (1985-1987).
   Justification: Work will be continued to develop adequate metnods for use by NRC response teams and in support of the NRC/interagency task force for emergency monitoring in order to improve instrumentation, sampling techniques, and procedures to measure airborne radioiodine and radioiodine in milk as well as other radionuclides.

4. Criteria for evaluating the adequacy of the calibration methods and standards for instrumentation to be used during accident conditions (1986-1988).
   Justification: The licensees are required to install sufficient radiological instrumentation to enable detection and classification of accidents, to follow the course of accidents, and to make protective action decision recommendations. This instrumentation will be used during accidents to assess a mix of fission products at very high ranges. NRC needs specific inspection criteria for evaluating the calibration adequacy of this instrumentation for emergency conditions.

5. Determination of adequacy of portable monitoring instrumentation currently to be used following an accident by licensees and the NRC for the purpose of conducting dose assessments and surveys to support corrective actions (1985-1988).
Justification: A great variety of portable instruments is used by licensees and the NRC to conduct onsite and offsite monitoring during accident conditions for offsite dose assessment and onsite surveying in support of onsite response (e.g., corrective actions). A great deal of testing has been performed on portable instruments by many testing facilities. These data need to be collected and assessed in order to determine the adequacy of these instruments under accident conditions and to determine if other testing is needed.

6. Development of operator action event-tree techniques for emergency response (1985-1987).
Justification: A methodology for incorporating operator action event-tree techniques into emergency operating procedures was developed in 1983 (NUREG/CR-3177, Vols. 1 and 2) and shows promise of being a very valuable technique for aiding emergency response.

7. A technical basis for guidance on implementing emergency preparedness regulations for fuel cycle and material licensees, to be used in developing regulatory guides (1985-1987).
Justification: In FY 1985, the NRC plans to publish a final rule on emergency preparedness for fuel cycle and material facilities. Guidance will be needed on implementing this new requirement.

8. Development of NRC guidance on ground-deposited radionuclides following a nuclear power plant accident (1986-1989).
Justification: The EPA is now developing criteria for reentry into contaminated areas following accidents. It will be necessary to study the significance of the EPA criteria as well as the significance of the entry of radionuclides into the food chain.

9. Assessment of severe accidents on the local ground-water system and estimates of the consequences to the accessible environment, including the analysis of mitigative methods to interdict ground-water transport of the radionuclides (1985).
Justification: Recommendations from "TMI Lessons Learned" and the Siting Policy Task Force cited ground-water siting factors and interdictive capabilities as important factors for severe accident considerations. The relationship between source terms and consequences to the accessible environment cannot be determined without information and analysis of the ground-water pathway.

10. Validation of dispersion models for atmospheric transport of released airborne effluents over a wider range of site terrain, to be the basis for developing a regulatory guide (1986).
Justification: Currently, the available models for the atmospheric dispersion of airborne releases from nuclear facilities are unverified for some terrains. Tracer experiments have been done to validate the available models for flat and lakeshore terrain, and similar data are needed for ocean front or hilly terrain.

4.8.2 Research Program Description

The strategy is to build on existing emergency preparedness experience with research that 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.

Research in FY 1985 through FY 1989 will provide information to form an improved technical basis for revising regulations and guides, formulating inspection procedures, and assessing the adequacy of emergency preparedness.

The major research products will be:

1. Evaluation of protective action decisionmaking (1985-1986).

2. Evaluation of emergency action level identification (1985-1986).

3. Evaluation of portable instruments for measuring radioiodine and other radionuclides identified as important by new source term work (1985-1986).

4. Evaluation of adequacy of calibration methods for radiological emergency instrumentation (1986-1987).

5. Evaluation of adequacy of radiological instrumentation under accident conditions (1985-1988).

6. Operator action event-tree techniques for aiding emergency response (1985-1987).

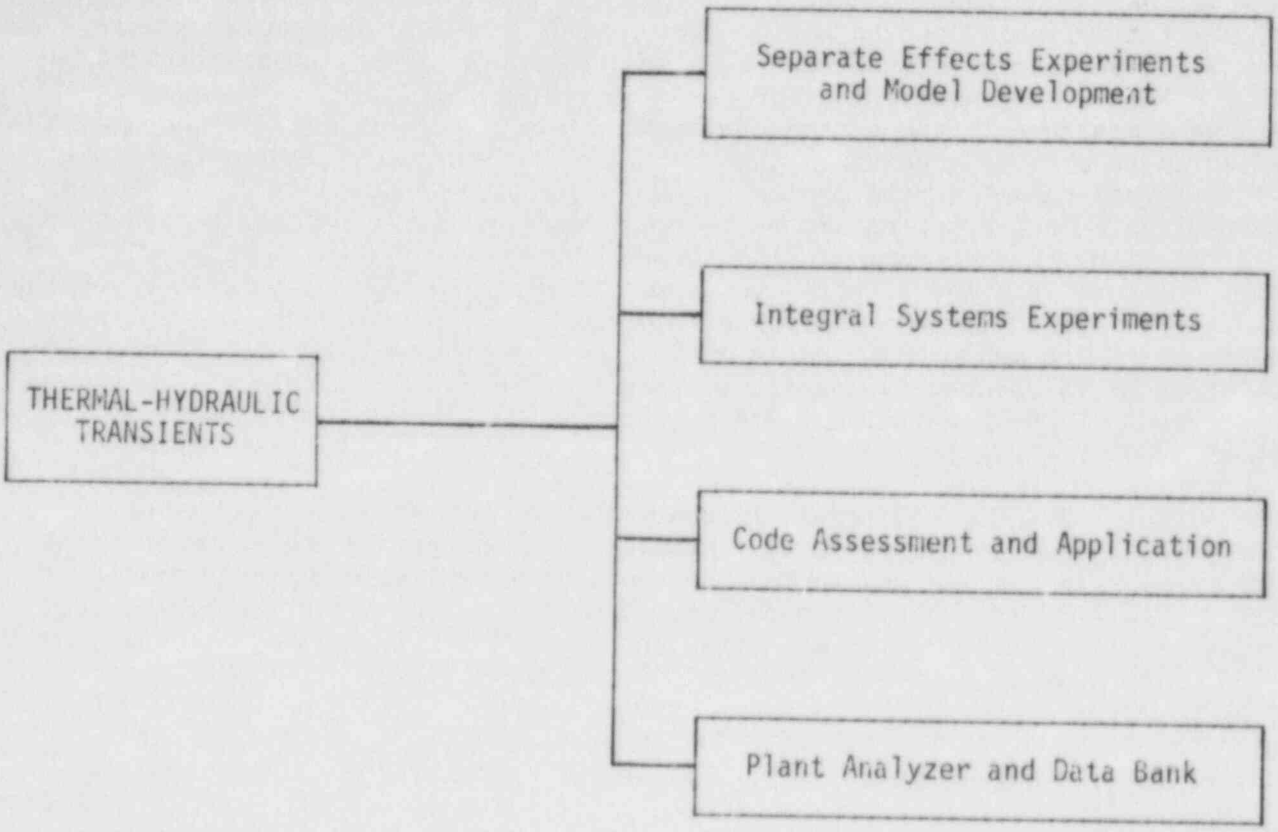
7. Technical basis for guidance on implementing emergency preparedness requirements for fuel cycle and material licensees (1986-1987).

8. Evaluation of ground-deposited radionuclides as they affect reentry and uptake into the food chain (1986-1989).

9. Regulatory guidance on interdictive strategies for severe-event accidents at nuclear facilities dealing with ground-water-contaminant transport and available mitigative techniques (1985).

10. Validation of atmospheric transport models for the atmospheric dispersion of airborne releases (1986).

Thermal-Hydraulic Transients



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
THERMAL-HYDRAULIC TRANSIENTS	$27.3	$30.3	$43.3	$39.5	$37.8

5. THERMAL-HYDRAULIC TRANSIENTS

This 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 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 complex plant system transients. The research related to Appendix K is nearly completed and will culminate in revisions to Appendix K during the next year. The data obtained through the separate effects research programs enables code developers to produce models that represent physical phenomena. The assembled models are then compared with the data obtained from the integral 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 bank.

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 so as to provide data to develop and assess accurate correlations of the parameters used for the prediction of these phenomena. Ongoing experiments include downcomer thermal mixing, countercurrent flow, grid spacer effects, nonequilibrium post-CHF (critical heat flux) heat transfer, PWR loop and steam generator oscillations, and low-flow heat transfer. Some of these programs are jointly funded with EPRI and industry.

5.1.1 Major Regulatory Needs and Their Justifications

1. Best estimate of the temperature of mixed emergency core cooling and primary cooling water as the mixed fluids cool the pressure vessel under

conditions that might lead to PTS (see Chapter 1), to be used to help resolve USI A-49 and to assist NRR in providing guidance to licensees on how to perform plant-specific PTS analysis (1985).
<u>Justification</u>: The possibility of PTS may affect the operating procedures of existing reactors and, therefore, is a regulatory issue that must be expeditiously addressed. There are only limited experimental data on the mixing of cold emergency core cooling water with hot reactor vessel water in the cold leg and downcomer. This information is needed to determine possible thermal stresses on the reactor vessel.

2. Improvement of flow and heat transfer models to serve as a basis for understanding and validating processes simulated in transient and LOCA systems analysis codes (1985).
<u>Justification</u>: To assess operator guidelines proposed for management of a range of accident sequences, system analysis codes, frequently using special fluid flow models, are used. The regulatory assessment of these guidelines depends on the validation process for estimates of error ranges. Since such assessments concern existing operating reactors, the work responds to a currently important regulatory issue.

3. Experimental data and analysis for revising the post-CHF heat transfer correlation and the fuel element blockage criteria of Appendix K to 10 CFR Part 50 (1985).†
<u>Justification</u>: The Congress has recommended that Appendix K be reviewed and revised (House Report 97-795, House Science and Technology Committee, Recommendation 1A) in response to the Nuclear Safety Oversight Committee and industry testimony that excess conservatisms exist in this (and other) rules. Chairman Palladino committed NRC to revise Appendix K in testimony to that committee on June 15, 1982.

4. Revision and assessment of heat transfer package in the RELAP and TRAC codes, to be used in evaluating operator guidelines (1987).
<u>Justification</u>: 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 CHF that are purposely chosen to be conservative for use in emergency core cooling system (ECCS) licensing evaluations. The conservatisms are known 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.

5. 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).
<u>Justification</u>: 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.

6. 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.

7. Scoping tests for integral behavior, to be used in evaluating test facilities to resolve scaling issues (1985).
Justification: Small test facilities are needed to support major integral facilities to investigate facility scaling and to scope unexpected behavior. In addition, the scoping test results should provide data for assessing system codes to be used as an audit tool for licensing evaluations.

8. 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.

5.1.2 Research Program Description

Ongoing experiments include grid spacer effects, nonequilibrium post-CHF heat transfer, PWR loop and steam generator flow oscillations, Babcock and Wilcox (B&W) primary loop transient simulations, and two-phase flow modeling.

The mission for performing heated grid spacer effects testing is to characterize the individual thermal-hydraulic mechanisms that tend to enhance heat transfer just downstream of a grid spacer, thus resulting in additional cooling of the reactor fuel rods. The research is to obtain droplet-size data from a 4-rod electrically heated bundle using a Westinghouse grid spacer and operating under typical reflood conditions.

Post-CHF tests will be performed in a 9-rod electrically heated bundle for the purpose of measuring and correlating nonequilibrium effects under typical flow conditions of low inlet mass fluxes and qualities. The test results will be used for code assessment and development and for verification of previously developed correlations.

PWR loop experiments at the Massachusetts Institute of Technology 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.

Building of a small-scale B&W simulation reactor loop at the University of Maryland is currently in progress 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. In addition, separate effects testing of the B&W hot leg will be 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 suppression pool into rising steam bubbles and from the lower BWR plenum into the fuel bundles and jet pumps.

A thermal fluid mixing program is being conducted at Purdue and Creare 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. This research will provide NRR with valuable support in evaluating the PTS issue.

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, will also be investigated under other programs.

The major research products will be:

1. Thermal fluid mixing and pressure vessel fluid heat transfer models (1985).

2. a. Inverted annular flow model (1985).

   b. Condensation heat transfer model (1985).

   c. Interfacial friction model (1985).

3. a. Bundle post-CHF data analysis and heat transfer model (1985).

   b. Flow blockage criteria (1985).

   c. Qualified grid heat transfer enhancement models (1985).

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

5. Validated steam generator data from MB-2 (1986).

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

7. Simplified B&W loop simulation for understanding separate effects phenomena and for B&W experimental facility formal design (1985).

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

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 FIST program in 1986 and the MIST/OTIS program in 1987. Foreign agreements and understandings provide for data exchanges with similar facilities in the FRG (PKL), Commission of the European Community (LOBI), and Japan (ROSA IV). 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. 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 steam generator tube rupture data for code validation and review of operator guidelines (1985).

2. Experimental steam line and feedwater line break data for evaluation of operator guidelines, plant recovery technique assessment, and validation of code capability (1985).

3. Data on BWR ATWS and recovery techniques for NRR use in analysis of ATWS (1985) and boron mixing data (1987).

4. Experimental data on BWR accident sequences for code verification and evaluation of operator guidelines for loss of feedwater, turbine trips, station blackout, and intermediate breaks (1985).

5. Data for the evaluation of severe transients in BWRs to evaluate severe accident capability and operator guidelines (1986).

6. Experimental tube rupture data on once-through steam generators for code validation and review of operator guidelines (1986).

7. Integral loop (2x4) test data to assess operator guidelines during natural circulation transients (1986).

8. Data on small-break LOCA (cold leg and pressurizer) with and without pump operation for code validation and review of operator guidelines (1986).

9. Feed-and-bleed data for primary coolant system to evaluate cooling effectiveness and procedures for operator guidelines (1986).

Justification: The following justifications apply to all needs listed above:

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 human factors) 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 integral 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 response during transients and the influence of operating procedures.

The Semiscale facility testing from 1984 to mid-1985 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. In addition, events with a high probability of occurrence will be studied. 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 FY 1985 and FY 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 will consist of two series of tests to be 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 will primarily investigate small-break LOCA events with completion in FY 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 FY 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.

DOE owns and operates the LOFT facility, which is under the sponsorship of the OECD LOFT project, and funding is shared by approximately 15 countries and agencies. The NRC is a member of the project based on a one-time contribution of $25 million in FY 1983. NRC has indicated its preference for tests of the type in Needs 5 through 7 of Section 5.2.1. 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 FY 1983 to FY 1985, with follow-on analyses through FY 1987.

The first phase of testing in the FIST facility, which represents a BWR-6 plant, was completed in late FY 1983. At this time, simulations of a wide variety of BWR coolant system transients were available for use in assessing the adequacy of the codes. During a period of time from late FY 1983 to early FY 1984, the data will be used to assess the codes, and the adequacy of the data base to perform this task will be evaluated. Instrumentation will be upgraded and modifications to the test program will be made based on evaluation of the first phase of testing. A second phase of testing will be conducted in 1984 and a similar assessment of code capability will be made in 1985. If a need is shown for further data (severe transients, boron mixing) to support the final versions of the codes, continued testing will be conducted in 1986.

The major research products will be:

1. Semiscale MOD-2B data used to assess steam generator tube rupture models (1984).

2. a. Semiscale data obtained in 1985 to be used to assess steam line and feedwater line break calculations (1985).

   b. Semiscale data obtained in 1985 to be used to assess small-break LOCA without HPI (1986).

   c. Semiscale data obtained in 1985 to be used to assess analysis codes for large-break LOCA in upper-head-injection (UHI) equipped plants (1986).

   d. Evaluation of Semiscale data base (1985).

3. FIST Phase II ATWS tests (1985).

4. Evaluation of BWR transients using FIST Phase II data (1985).

5. FIST Phase III test results (1986).

6. Experimental once-through-steam-generator tube rupture data from OTIS and MIST (1986).

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

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

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

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 will be completed in 1984 and remain fixed until the end of 1986.

5.3.1 Major Regulatory Needs and Their Justifications

1. Maintenance and evaluation of operational TRAC and RELAP5 codes (1985-1987). Justification: Assessments of the fixed versions (as of end of 1984) 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.

2. Information needed for the resolution of USI A-47, "Safety Implications of Control Systems," and USI A-49, "Pressurized Thermal Shock" (1985). Justification: The resolution of these issues has a high priority because of potential risk-reduction significance and use in regulatory decisionmaking for both operating plants and those under licensing review. Research is necessary to provide information and the technical basis for the needed regulatory judgments.

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 1983, 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 1985-1986 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 early 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. Code application is planned to increase over the next several years as resources are shifted from code development and independent assessment although independent assessment by foreign organizations should continue until 1989. Using the results of independent assessments, the codes will be maintained and improved.

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

The major research products will be:

1. a Development of acceptance criteria for large thermal-hydraulic analysis codes (1985).

   b. Final major versions of the advanced multidimensional two-fluid transient analysis codes: TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1 (1987).

   c. Assessment of TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BD1/MOD1 (1985) and TRAC-BF1 (1987) by DOE laboratories.

   d. Best estimates of how one BWR and one PWR respond to large-break LOCA (1985).

e. Integration of system codes to benchmark and audit risk analysis methods (1986).

f. Assessment of COBRA/TRAC by DOE laboratories (1985).

2. Technical basis for resolving USI A-47, "Safety Implications of Control Systems" (1985).

5.4 Plant Analyzer and Data Bank

This element includes computational technique improvements and the development of user-oriented capabilities to use 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 will be used directly by the licensing staff, beginning in 1984, 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 top ability for interactive operational manipulations at midpoints during long transients (1988).
   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 expected to be in use by the licensing staff as early as 1984.

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 will be provided to NRR in 1984. This office can then 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 (1985).

   b. Speeding of TRAC by vectorization on the CRAY-1 and speeding of RELAP5 by conversion to CRAY-1 (1985).

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

   d. First general user version of BWR system plant analyzer completed and demonstrated for use by NRC personnel (1987).

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

f. Adaptation of nuclear plant analyzer technology to CRAC and SCDAP (1988).

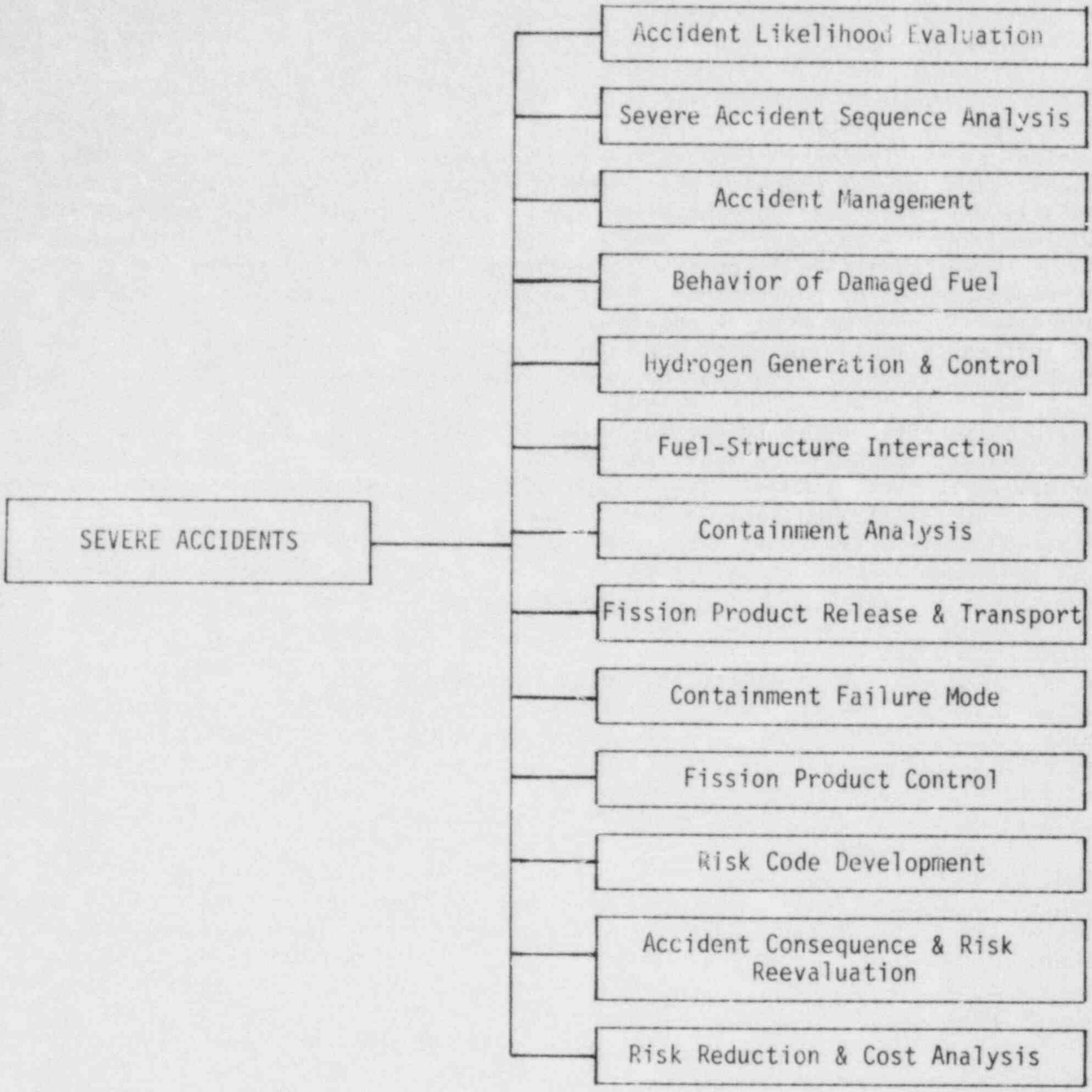
2. a. Installation of two additional NRR plant analyzer and plant data bank terminals (1985).

   b. Plant secondary systems and control systems implemented in plant data bank (1985).

   c. Use of plant analyzer and plant data bank to analyze transients in full-scale LWRs to resolve licensing and safety issues (1985-1988).

6

Severe Accidents



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
SEVERE ACCIDENTS	$50.9	$48.3	$48.5	$35.0	$30.0

6. SEVERE ACCIDENTS

This program supports 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), accident management, 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 anticipate areas of phenomenological uncertainty that may be identified as part of the current American Physical Society review of NRC source term research. 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. A major objective of this program element is to improve overall confidence that significant accident initiators and sequences have not been overlooked.

6.1.1 Major Regulatory Needs and Their Justifications

1. Identification and probabilistic evaluation of dominant accident sequences, including uncertainties that would have the potential for leading to severe core damage or core melt (1985).
   Justification: Identification and description of the principal accident sequences that could lead to core damage will provide the fundamental and essential element for defining the scope and direction of the SASA program (see Section 6.2) and its related experimental program and for the systematic analysis of the risk-reduction effectiveness of backfitting safety systems on operating plants (see Section 6.13). Identification of plant classes will also be provided to the severe accident elements and to other regulatory activities.

2. Validation of the likelihood of dominant reactor accident sequences based on recent PRA studies and LER data (1985).
   Justification: Future decisions on the incorporation of additional safety requirements into the regulations and the eventual implementation of the Safety Goal and backfitting on operating plants require that the quantification of accident likelihoods be made using the most recent data available.

3. Identification of major common-mode and interactive failure mechanisms and methods for incorporating this information into the results of PRAs (1985-1986).
   Justification: Longer-range decisions regarding the need to provide additional protection of plant safety systems against common-mode failures such as fires, sabotage, and operator error will require a comprehensive identification and quantitative assessment of the contribution to plant risk of such types of failures.

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) and to support decisions on backfitting additional safety features to existing plants (see Section 6.13). This research is also 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 (event tree) 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, the reevaluated event trees, and the event 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 6.13). The objective of the event tree 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 program will make modifications in accident likelihoods to account for (probabilistically) poorly understood events such as fires and operator error and to attempt to make the event trees more generalized than originally established without sacrificing necessary detail to differentiate among plant types. Studies of earthquakes, etc., are being developed to draw upon ongoing research programs such as the Seismic Safety Margins Research Program. 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 (1984-1986); reports on likelihood of additional accident sequences and related probabilities (1985-1986); and identification of generic plant class (1985-1986). (Also applies to Need 2.)

2. Evaluations of plant operating data, LERs, and vendor information for component and system reliability (1985). (Also applies to Need 3.)

3. Identification and assessment of operator contribution to severe accident initiation and mitigation (1985).

6.2 Severe Accident Sequence Analysis

Research uses the analytical assessment of plant accidents within and 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, MARCH/CORRAL, CRAC). 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 (1985-1989).
   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 the 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 (1989).
   Justification: NRR has transmitted to RES for analysis by the SASA program some current safety concerns that arose during the course of licensing reviews. 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 in this time interval by responding to concerns as they arise.

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 (1988).
   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.

5. Analytical evaluation of such unresolved safety issues (USIs) as:

   a. Safety implications of control systems interaction with protective action during accidents, A-47 (1986).
   b. Anticipated transients without scram (ATWS), A-9 (1985).

   Justification: SASA studies provided a significant part of the data generated to support the resolution of the station blackout USI. Detailed accident scenarios were studied for both BWRs and PWRs to develop the time of key events in the accident sequence and to identify operator actions that would arrest or prolong the accident. The resolution of these safety issues depends on the detailed analyses of accident sequences, as provided by SASA, to identify the specific nature of the threat and the usefulness of proposed remedies. The specific kind of support provided by the research results of this program is significant in the regulatory process.

6.2.2 Research Program Description

The strategy is to analyze dominant 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 severe accident sequences. Test programs such as the Power Burst Facility (PBF), Semiscale, Loss-of-Fluid Test (LOFT), 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. Evaluation of plant abnormal and emergency operating procedures for preventing or mitigating severe accidents in both PWRs and BWRs (1989). (Also applies to Need 2.)

2. Development of plant response characteristics to multiple failure sequences identified by risk assessment programs and severe accident research programs (1989). (Also applies to Need 1.)

3. Assessment of instrumentation requirements to provide unambiguous information to the operator during accidents or transients (1988).

4. Evaluation of fission product (FP) pathways, FP transport rates, and FP inventories in plant structures resulting from severe accidents (1988).

5. a. Assessment of potential detrimental influence of control system failures on severe accident response (1988).

   b. Assessment of effectiveness of potential procedural and plant design changes to ensure acceptability of consequences of an ATWS (1985).

6.3 Accident Management

The goal of this element is to integrate strategies combining plant systems design and operation with operator guidelines and procedures to prevent, arrest, or mitigate the consequences of potentially severe accidents. The scope of the potential integration will be based on results of SASA (see Section 6.2) and of human factor research (see Chapter 4).

6.3.1 Major Regulatory Needs and Their Justification

1. Identification of plant system designs, configurations, or operational capabilities that reduce the probability of plant damage from system failures, limit the extent of damage resulting from such failures, or mitigate the consequences of severe damage when failures occur, this information to be used to support severe accident rulemaking and to evaluate safety systems in new plants (1985).†

2. Improvements in the amount, form, and content of information presented to the operator through instrumentation to assist decisionmaking as accident precursors occur and during severe accident transients, for use in regulatory evaluation of operating procedures and risk analysis (1986).

3. Criteria, procedures, and data that will minimize the effects of human error in design, operation, and maintenance, for incorporation into the severe accident rulemaking (1985).†

4. Improved procedures and methods (e.g., hydrogen control) for ensuring containment integrity in the event of core meltdown and pressure vessel rupture, thereby limiting the consequences of severe damage to the public, to be the basis for amendments to the regulations (1985).†
   Justification for Above Needs: Accident management is not oriented to specific tasks but interfaces with several other elements. Hence the justifications for the various regulatory needs listed above are combined here. The accident management element is needed to integrate results of research studies on SASA with industry programs and with human factor, reliability, and phenomenological research programs of the NRC, and with licensee emergency plans approved by IE. This integration will achieve related benefits from research efforts for improving operational requirements for operator action during severe accidents.

6.3.2 Research Program Description

The research plan is to reevaluate accident likelihood and to analyze severe accident sequences so as to identify changes in plant design or administrative controls that can reduce accident probability. These changes will be based on research described in other elements of this program as well as on research sponsored by industry. The principal experimental research supporting this element will be that on the behavior of damaged fuel, containment integrity, and hydrogen generation and control.

Of equal significance is research on human factors where data on reactor operator response characteristics during simulated accidents and the effects of computerized plant control displays on operator performance will be developed.

The major research products will be:

1. a. Development of operator training requirements for severe accident management (1985). (Also applies to Needs 2, 3, and 4.)

   b. Identification of operator error rates (1985). (Also applies to Need 3.)

   c. Establishment of man-machine interaction requirements for accident management (1985). (Also applies to Need 2.)

2. See RES Products 1.a and 1.c.

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

4. a. Development of operator procedures for recovery from potentially severe accidents (1986).

   b. Development of operating procedures for ensuring containment integrity during severe accidents (1986).

6.4 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, 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.

6.4.1 Major Regulatory Needs and Their Justifications

1. Determination of the actual hydrogen release from the core and recommendations for appropriate regulatory action (1985).
   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 establishing severe accident policy (1985).
   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 (1985).
   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.

6.4.2 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 scope of the currently planned program depends on obtaining foreign government contributions in the form of financial support and/or conduct of research that NRC would otherwise have to fund.

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 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 development 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 will be completed in FY 1985.

The major research products will be:

1. a. Report on results of PBF SFD tests under core-recovery conditions (1985).

   b. Report on NRU SFD full-length verification tests (1985). (Also applies to Need 2.)

2. a. Report on analysis of PBF tests (1985).

   b. Initial report (1985) and final report (1987) on independent examination of selected TMI-2 core samples.

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

3. a. Report on initial ACRR debris formation experiments (1985).

   b. Report on ACRR debris formation experiments (1987).

   c. Report on ACRR quench-debris experiments (1986).

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

   e. Results of ACRR melt progression separate-effects experiments (1988).

   f. Assessment of MELPROG with results of ACRR melt progression experiments (1989).

4. a. Assessment of SCDAP-Mod 2 with PBF and ACRR results (1985).

b. Assessment of final version of SCDAP with PBF, NRU, and ACRR results (1986).

c. Report on sensitivity studies using MELCOR/SCDAP/MIMAS/MELPROG (1988).

6.5 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 or could cause the failure of safety-related equipment in 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.

6.5.1 Major Regulatory Needs and Their Justifications

1. Data from all areas of hydrogen research such as generation, ignition conditions, and mixing to support the proposed rule on interim requirements related to hydrogen control (1985).†
   Justification: The interim rule on hydrogen requires that analysis be performed to assess various accident scenarios using best-estimate and risk codes.

2. An assessment of the survivability of safety equipment during a hydrogen burn for the "Licensing Requirements for Pending Construction Permit and Manufacturing License Application" (CP/ML Rule) (1985).†
   Justification: Both the rules and licensing requirements call for assessment of the survivability of safety-related equipment. The hydrogen burn survival research program will provide supporting and confirmatory data.

3. Technical data and information on hydrogen generation and control to help formulate the Commission policy on hydrogen regulations not covered by the existing rule (1985-1986).
   Justification: This program will be assessing the threat posed by hydrogen from core-melt accidents more severe than those currently covered by the proposed and final hydrogen rules. This information will be used in PRA to determine if additional control and mitigation requirements are cost effective.

4. As with Need 1 above, a wide spectrum of hydrogen research information to resolve USI A-48, "Hydrogen Control Measures and Effects of H_2 Burning on Safety Equipment" (1985).
   Justification: Appropriate regulatory actions are being considered for resolution of USI A-48 relating to hydrogen issues in BWR Mark III and ice condenser containments. This program will provide technical support for that task.

5. Specific data on hydrogen combustion phenomena for ice condenser and Mark III containments (1985-1986).

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.

6.5.2 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 and pressure/temperature transients on the equipment.

The hydrogen behavior program is developing multicompartment combustion and improved detonation models to predict containment pressure and containment 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. Mitigation options include deliberate ignition of lean mixtures of hydrogen, catalytic igniters, and oxygen depletion and pre- and post-accident inerting. This program includes studying the effects of sprays and aerosols on igniter performance under accident conditions. A deliberate flaring technique in conjunction with high-point vents and modification of containment atmosphere are also being assessed.

Assuming that the hydrogen burn is not mitigated, the consequences of the failure of important safety equipment is being assessed. Equipment tests in actual hydrogen burning environments are being conducted. The program is in two phases: (1) to provide information rapidly for near-term licensing decisions and (2) to develop reliable methods of predicting the response and survivability of equipment.

The major research products will be:

1. a. Analysis of three to five specific plants for degraded core and severe accidents (1985). (Also applies to Needs 3, 4, and 5.)

   b. High-point vent mitigation (1985). (Also applies to Needs 3 and 5.)

   c. Report on the hydrogen burn survivability of safety-related equipment in ice condenser and Mark III containments (1985). (Also applies to Needs 2, 4, and 5.)

d. Improved hydrogen combustion mitigation systems (1985-1986). (Also applies to Needs 3 and 5.)

2. See RES Product 1.c.

3. a. Assessment of effects of aerosols on hydrogen control system and the effects of hydrogen burning on source term (1985-1986).

   b. Preliminary hydrogen diffusion model and flame acceleration model (1985). (Also applies to Need 5.)

4. See RES Products 1.a and 1.c.

5. See RES Products 1.a-d and 3.a and b.

6.6 Fuel-Structure Interaction

Experimental research described in this element will obtain data on the consequences of high-temperature core fuel debris interaction with structures below the reactor vessel 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 vessel cavity concrete basemat, (2) water present at the time of fuel debris escape or subsequently introduced to the cavity by the safety injection system, and (3) mitigating structures or devices.

6.6.1 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 (1985).

2. Noncondensable gas and aerosol release for analysis of containment performance during severe accidents (1985).

3. Rapid steam generation with potential for containment failure (1985).

   Justification for 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 vessel cavity. This applies to the three needs listed above. The sources for the containment challenge are not adequately known, and more research is needed to evaluate (1 and 2) effects of the interaction between hot fuel debris and concrete basemat materials; (3) effects involving rapid steam generation; and (1-3) the quantification of parameters used to provide a basis for verifying analytical models used in severe accident assessment and accident management planning.

6.6.2 Research Program Description

The plan of research is to conduct small- and large-scale scoping and phenomenological tests of core-melt/concrete interactions and core-melt/coolant interactions in order to quantify models for gaseous and aerosol sources used in the CORCON and CONTAIN codes. In addition, the effects of in-vessel and ex-vessel introduction of 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 and fuel-melt/retention materials interaction models in CORCON to be obtained from large-scale test facilities (1985).

2. a. CORCON and CONTAIN code verification (1985).

   b. Experimental data for modeling the interaction of hot solidified melts with concrete and long-term cooling characteristics of solidified core melts (1985).

   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 (LMF) (1985). (Also applies to Need 3.)

3. See RES Product 2.c.

6.7 Containment Analysis

This element will provide an analytical tool for the assessment of the challenge to the containment system from postulated severe accidents. The types of challenges produced by the postulated accidents consist of overpressure from steam generation, 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 findings will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.7.1 Major Regulatory Need and Its Justification

A comprehensive, flexible computer code to adequately assess severe accident challenges to containment integrity; the CONTAIN code is under development to meet this need (FY 1985-1989). 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 such 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 a better basis for analyzing the risk. Decisions on future regulatory actions on dealing with severe accidents will require a state-of-the-art capability for analyzing containment performance.

6.7.2 Research Program Description

This research will undertake to develop a computer code capable of simultaneously assessing the many-faceted challenges to containment during severe accidents. Incorporation into a single code structure of phenomenological models for the many simultaneous processes occurring within the containment structure during severe accidents 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 core-melt/concrete interaction tests, high-pressure melt ejection tests, hydrogen burning tests, and rapid steam generation tests, from which aerosol source term information will also be generated.

The major research products will be:

1. a. CONTAIN code verification (1985).

   b. Integration interface between CONTAIN code and a code such as TRAP-MELT for fission product and energy source data (1985).

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

   d. CONTAIN code maintenance and minor improvements (1985-1989).

6.8 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 within the reactor coolant system, and on attenuation within the containment vessel.

6.8.1 Major Regulatory Needs and Their Justifications

1. Determination of kinetics and quantities of fission product release and in-vessel attenuation in a correct in-core thermal-chemical environment, for use in regulatory decisions concerning conditions beyond the design basis (1985).
   Justification: Direct measurement of the fission product release from prototypic fuel under well-characterized primary system conditions will allow reliable application of the Marviken experiments to full scale, using the TRAP-MELT code. These measurements will also be used to

benchmark out-of-pile separate effects tests on fission product aerosol release from fuel under severe accident conditions. Given such information, the NRC will be able to answer more precisely the question of fission product attenuation in the primary system external to the core and to predict the amount, timing, and chemical form of the fission products that enter the containment.

2. Improvements in PRA consequence calculational methods (1985).
   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.

3. Siting policy (1985).
   Justification: Nuclear power plant siting requirements have the potential of alleviating most 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 siting requirements. The Commission has recognized the need of reassessing the source term prior to issuing new siting regulations (Policy and Planning Guidance, NUREG-0885).

6.8.2 Research Program Description

The strategy of the fission product release and transport program is to develop models and an experimental data base for rectifying the predicted release, deposition, and transport behavior of radionuclides under severe LWR accident conditions and for use as a basis for improved engineered safety features (ESFs) (see Section 6.10). Computer models are being developed 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 also being developed to assess the transport 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 ESFs in mitigating fission products under severe accident (fission product and aerosol) loadings and environmental conditions are discussed in Section 6.10.

Laboratory-scale separate effects experiments are under way to provide data for model development in a number of areas, including fission product release from fuel, fission product chemistry, and fission product interactions with prototypic surface materials.

Larger-scale multiple effects experiments are being conducted to assess the validity of the computer models. These multiple effects tests include the in-pile fission product release experiments being carried out in the PBF reactor and the large-scale fission product and aerosol transport tests to be conducted at the Marviken facility in Sweden.

Using the models developed within this element, periodic analyses will be conducted to develop and refine best-estimate, release-from-plant, radionuclide source terms for severe LWR accident sequences. The first such source term assessment was completed in 1983.

The major research products will be:

1. a. Data report for benchmarking fission product release models from out-of-pile tests up to 2900K (1985).

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

2. a. Improved version of TRAP-MELT fission product and aerosol transport code (1985). (Also applies to Need 3.)

   b. Documented computer code model for fission product release during core-melt interaction with concrete (1985). (Also applies to Need 3.)

   c. Data report from experiments with fission product chemistry in the gas or aqueous phases to confirm or improve fission product behavior model (1985).

   d. Large-scale (Marviken) fission product and aerosol transport test package to be used to assess TRAP-MELT code (1985).

   e. Report of TRAP-MELT sensitivity study (1985).

   f. Severe accident source term uncertainty report (1985). (Also applies to Need 3.)

3. See RES Products 2.a, 2.b, and 2.f.

6.9 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 will 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 6.12. The findings of the containment failure mode research program will be reviewed for usefulness in improving the response capability of the NRC Operations Center.

6.9.1 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 (1986).

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 estimates made, on behalf of licensees, 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 implementation 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 (1988).
   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.

6.9.2 Research Program Description

The research effort will focus on four areas:

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

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

3. Experiments on leakage performance of electrical penetrations in high-temperature environments.

4. Experiments on the performance of containment valves.

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 6.5, 6.6, and 6.7). In addition, there will be interactions with the risk code development element (see Chapter 4). There will also be interaction with other United States programs. Contributions to this program from EPRI are anticipated in the way of analytical predictions of capacity to be compared against test results.

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 1984. Reinforced concrete models will be tested in 1985.

The planning of tests with dynamic, unsymmetric pressure loads will begin in 1984. Based on results from the hydrogen generation and control program and results from the static pressure test series, dynamic pressure experiments for steel and concrete containment models will be designed. These experiments will be performed in 1985-1987.

Plans 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 will be performed in 1983 and 1984. Experiments on models of major penetrations will run from 1983 through 1986.

The major research products will be:

1. a. Comparison of estimated steel containment capacities with experimental static pressure results (1985).

   b. Results of experiments on valve performance and electrical penetrations (1985).

   c. Results of initial test series on models of major penetrations (1985).

   d. Comparison of predicted capacities for prestressed and reinforced concrete containments with experiments under static pressure (1986).

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).

6.10 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.

6.10.1 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.

6.10.2 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. Continuation of code (SPARC) improvement on the performance and scrubbing efficiency for suppression pools (1985) and verification based on experimental data to be provided by EPRI (1986).

   b. Code (ICEDF) verification for evaluation of PWR ice-condenser effectiveness (1985-1986).

   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; 1985). 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 (1985-1987).

   e. Report and input for code development on hydrogen burning and its effect on aerosol concentrations and selected ESF-system performance (1985).

   f. Evaluation of existing PWR/BWR ESF-system design (1985).

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

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

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

6.11 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 6.12 and 6.13. 1985 will be a transition year for MELCOR activities. Code development will continue, and initial applications will be conducted. Risk code development beyond 1985 should be substantially reduced following Commission decisionmaking on severe accidents. MELCOR activities beyond 1985 will concentrate on applications to regulatory issues.

6.11.1 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 (1985).
Justification for Above Need: 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 6.12 and 6.13 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 6.12 and 6.13. 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.

6.11.2 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:

- o Use of early version of MELCOR for Commission decisionmaking (1985); MELCOR code documented for users such as the Severe Accident Research Program (SARP) and the Risk Methodology Integration and Evaluation Program (RMIEP) (1986).

6.12 Accident Consequence and Risk Reevaluation

In this section, the risk codes discussed in Section 6.11 are being applied in concert with products of other sections (e.g., 6.1, 6.9) 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.

6.12.1 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 (1985-1987).
Justification: As noted in Section 6.11, Item II.B.8 of the TMI Action Plan discusses undertaking a rulemaking process related to the need to consider severe accidents in the regulatory process. The consequence and risk reevaluations described in this section provide a baseline level of risk from which risk-reduction analyses (see Section 6.13) needed for this rulemaking 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. Thus, this work directly supports resolution of the severe accident rulemaking.

6.12.2 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 6.11) 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

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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.

studies. Further, as these consequence analyses of specific sequences are 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 (1985-1987), e.g., determination of likelihoods and consequences of early containment failures for surrogate plants (1985).

6.13 Risk Reduction and Cost Analysis

In this work, methods are being developed and analyses performed for the systematic evaluation of the costs and benefits of alternative concepts for reactor design and operation. Value/impact criteria are to be used to determine the cost effectiveness of current or proposed regulatory requirements.

6.13.1 Major Regulatory Need and Its Justification

Identification of those possible plant modifications that offer the most cost-effective means of reducing risk for the major LWR design types (1985). Justification: The severe accident rulemaking discussed in Item II.B.8 of the TMI Action Plan is intended for determining the need to consider severe accidents in the regulatory process. The identification of cost-effective means of reducing risk in LWRs is important to this rulemaking. The work described in this section will identify such cost-effective features.

6.13.2 Research Program Description

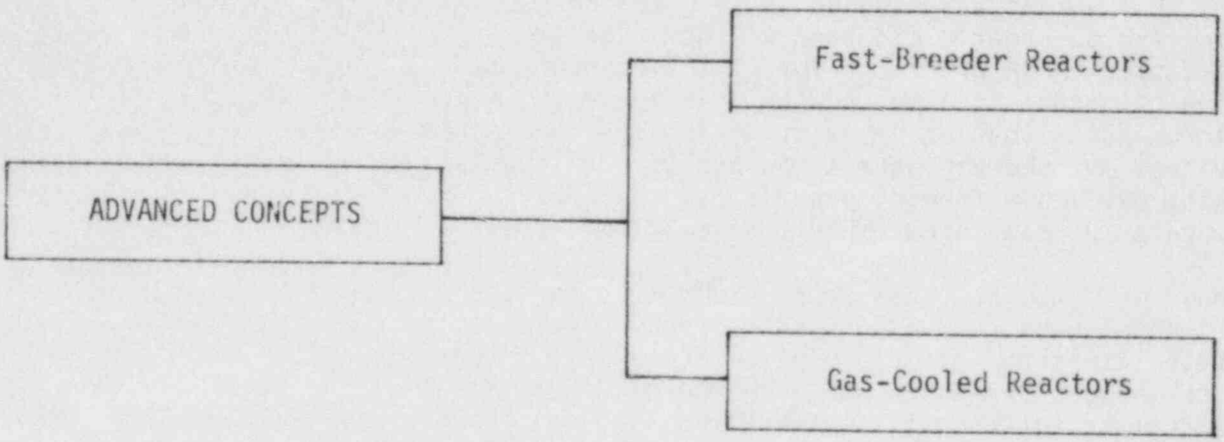
In this element, analyses of the risk-reduction potential and costs associated with a spectrum of possible plant modifications for severe accident prevention and mitigation are to be performed. Included in these possible modifications are, for example, filtered-vent containment systems, alternative shutdown heat removal systems, and better testing and maintenance procedures. The objective of such analyses is to identify those modifications that appear to present cost-effective risk reduction. Since such results will vary with the specific plant design being considered, analyses are to be performed for all major design types. Criteria need to be developed to judge when there are large enough costs or benefits so that meaningful decisions can be made. When decisions cannot be made, the areas of greatest uncertainty need to be identified and additional work performed to reduce the uncertainties.

The major research product will be:

- o Revised risk reduction and cost estimates using updated information (1985-1987).

Advanced Concepts

7



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
ADVANCED CONCEPTS	$5.1	$5.1	$5.1	$5.1	$5.1

7. ADVANCED CONCEPTS

7.1 Fast-Breeder Reactors

The Office of Nuclear Regulatory Research, as a result of the termination of the Clinch River Breeder Reactor (CRBR) project, has conducted a thorough review of its breeder reactor program support and has initiated the orderly closeout or redirection of all contracts directly supporting the CRBR licensing activity. The total research program for liquid-metal-cooled fast-breeder reactor (LMFBR) activities amounted to $7.0 million for FY 1984 when the decision not to continue the CRBR project was made in late November 1983. This amount was reduced to $5.0 million for FY 1984, which would provide for an orderly closeout of CRBR specific projects and for a transition to a reduced LMFBR program. The FY 1985 amount of $3.5 million will maintain a cadre of technical experts with the necessary specialized skills to allow continued participation in existing foreign agreements and remaining DOE activities and thereby permit the NRC to stay abreast of the latest technology by using available foreign experimental results and DOE information to assess and update NRC-developed LMFBR analytical methods.

A number of computer codes (e.g., COMMIX, SIMMER, SSC, CONTAIN) and a body of experimental data (e.g., in-pile accident energetics experiments, molten-core/concrete interactions) are in place to support LMFBR licensing and are being used to support LWR licensing. The COMMIX code is being used throughout the LWR safety calculational spectrum, and the experimental data are being put into the severe accident research data base.

The major research products will be:

1. COMMIX and SSC validated against Phoenix and SNR 300 (1985).

2. a. Data/models on transition-phase fuel removal processes (1985).

   b. Report on ACRR experiments on propagation of an explosive fuel-coolant interaction (FCI) in a molten fuel-sodium mixture and on potential for FCI augmentation of CDA energetics (1985).

   c. Improved version of SIMMER code for assessing accident energetics in large reactors (1989).

3. Data/models for assessing long-term ex-vessel debris coolability in sodium (1987).

4. a. Fuel aerosol simulant test (FAST) completed (1985).

   b. Source term modeling completed (1985).

7.2 Gas-Cooled Reactors

This element provides for a Ft. St. Vrain research report and for regulatory research on advanced high-temperature gas-cooled reactors (HTGRs). Included is the development of a comprehensive research plan to consider the recent DOE and industry interest in the modular pebble bed HTGR and in a smaller, 1140 MW(t) version of the General Atomic prismatic HTGR. Among the topics to be considered in the research plan are research and standards activities on general design criteria, siting criteria, development of the siting source term, assessment of basic standards, recommended changes to the standard format and content of safety analysis reports for HTGRs, modifications of certain appropriate codes and regulatory guides, and the development of a base of physical data, computer codes, and design and engineering information so that the technical bases for licensing HTGRs is clear. The research program addresses these subjects in detail and specifically aims to identify and develop or verify the chemical, metallurgical, structural, and system performance data and methods necessary to allow the NRC to assess the level of protection to the public health and safety from operation of a gas-cooled reactor facility.

7.2.1 Major Regulatory Need and Its Justification

Capability to deal promptly with any regulatory issues that may be necessary for the Ft. St. Vrain plant and to deal with NRC's regulatory responsibilities for any new commercial activity that may arise in the HTGR field.
Justification: The research program on the HTGR reflects the thinking that NRC research supports the regulatory program and is keeping pace with the DOE HTGR development program. The objectives of the research are to prepare NRC for licensing the next HTGR plant by resolving, to the extent possible, issues that affect the process; to assist NRR in developing safety criteria for HTGRs; to provide whatever guides and standards may be appropriate on a timely basis; and to extend the scope of NRC rules to include the HTGR. Specific objectives are to identify and develop or verify the chemical, metallurgical, structural, and system performance data and methods necessary to allow the NRC to adequately assess the licensability of the HTGRs.

7.2.2 Research Program Description

The near-term strategy for research in this element includes early completion of the research needs identified to RES by NRR in FY 1982 in support of the operating Ft. St. Vrain reactor. Many of the research tasks addressing these issues are of generic nature and fit within the long-range goals for HTGR safety research. These goals are being reassessed in FY 1983-1984 through redevelopment of the long-range plan to be consistent with the latest generation HTGR commercial concepts and the latest state of the art in HTGR safety. This research plan redevelopment effort employs PRA techniques to refocus priorities and to ensure consideration of otherwise hidden issues and addresses safety concerns revealed while evaluating the generic HTGR severe accident source terms. Careful attention is given to industry and DOE research on reliability and design of systems and equipment that affect the safety of plant response to accident conditions. NRC research seeks to cooperate with both DOE and foreign programs to maximize the effectiveness for NRC of safety research carried out both in the United States and abroad.

Specific technical and safety issues for new generation HTGRs will be more accurately identified when an initial lead-plant design is provided to the NRC. In the intervening period, the most pressing early technical issues are the development of NRC's site suitability criteria and the determination of appropriate safety criteria for HTGR-specific systems such as the reactor vessel and its thermal barrier. Clearly, new priorities will still require focus on fission product transport and deposition behavior in the reactor materials and system, better understandings of the structural material performance limitations, and better capability of evaluating system transient performance under accident conditions. Development of criteria for and behavior of the structural graphite core support system will continue to receive particular attention.

Plans will be implemented for cooperation with industry in standards development such as ASTM standards and ASME Code Sections III and XI as related to HTGR technology and for inservice inspection requirements for thermal barrier, prestressed concrete reactor vessel (PCRV) closure design, and containment requirements. The HTGR Safety/Licensing Handbook, initiated in FY 1982-1983, containing HTGR-specific guides, standards, data, and analytical techniques will be continually upgraded during the period.

The major research products to be identified in the plan are likely to include:

1. a. Evaluation and testing of graphite failure criteria and failure mechanism model developed in 1984 (1985).

   b. Handbook (2nd edition) on HTGR safety and licensing, to include a summary description of the Ft. St. Vrain design and operating history (1985).

   c. New Chapter 15 for HTGR edition of the standard format and content of safety analysis reports for nuclear power plants (1985).

   d. Completion of background data for site suitability criteria (1985).

   e. In-core testing of convective flow mixing and natural convection phenomena for code verification (1985).

   f. Completion of thermal barrier and liner cooling system requirements (1986).

   g. Advanced analysis code for HTGR fission product plateout and liftoff to supersede SUVIUS code (1986).

   h. Compendium of material property and strength data for nuclear-grade HTGR graphites, including test results on irradiation-induced creep and dimensional changes in isotropic graphite (1986).

   i. Adaptation of CHAP and ORECA codes to predict severe accident radiological releases (1987).

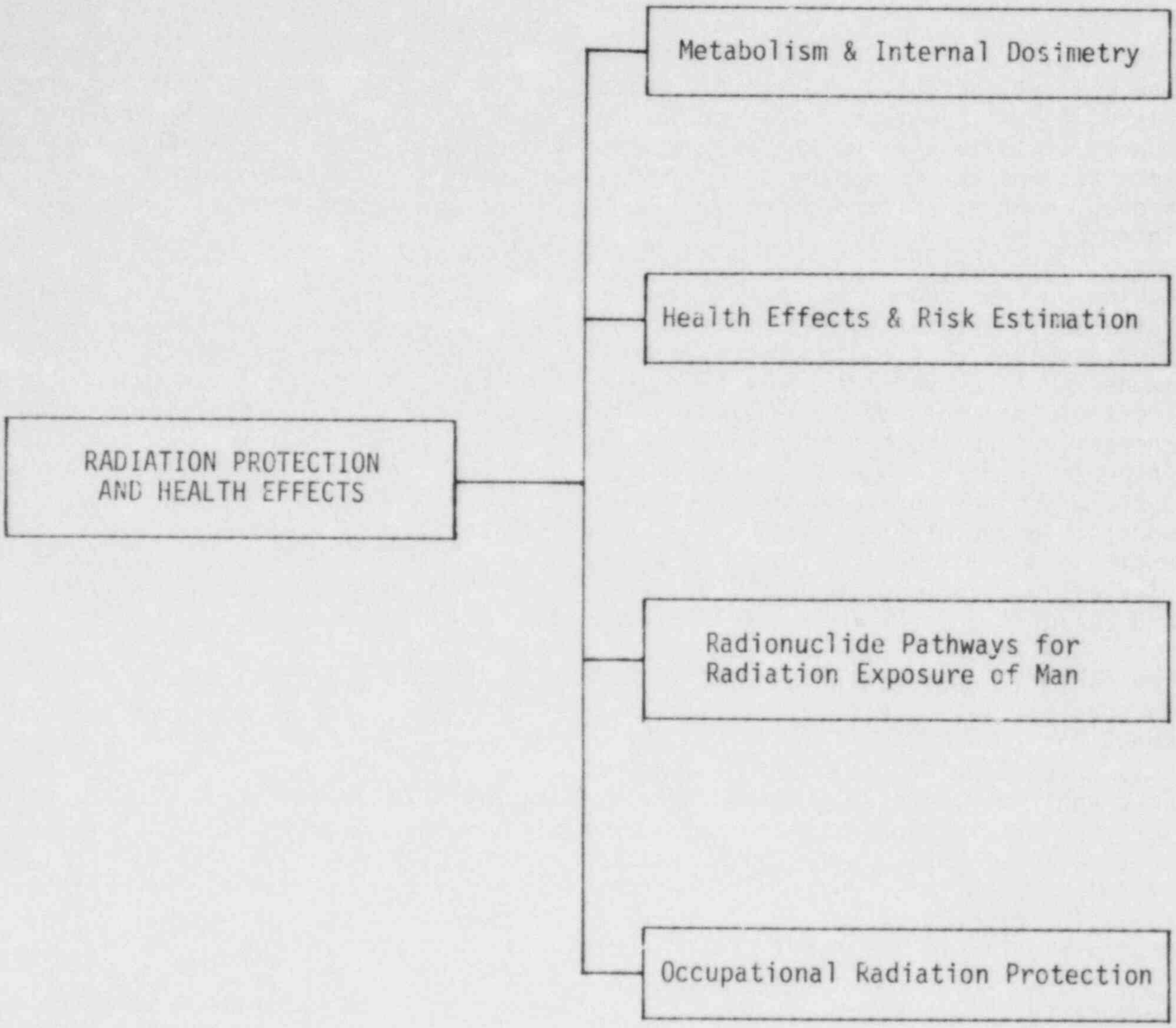
   j. NRC/RES cooperation in development of ASME Code revisions and code cases for evaluation of PCRV structure, penetration closures, and liner cooling (1985-1988).

   k. Advanced computer code for analysis of HTGR containment system response during depressurization events (1988).

If a licensing application is received, consideration would be given to programs for equipment qualification and fire protection, to long-term metals programs, to larger-scale testing programs for code validation, and to in-depth evaluations of specific safety issues pertaining to the reactor design and sites identified.

Radiation Protection and Health Effects

8



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
RADIATION PROTECTION AND HEALTH EFFECTS	$2.8	$3.6	$2.8	$2.8	$2.8

8. 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 at or above unacceptable levels and are as low as is reasonably achievable (ALARA), taking into account the technologies available, the dollar costs of improvements in reduction of risks, and other socioeconomic considerations that serve the public interest.

Achieving that goal requires the availability of adequate technical and scientific data bases to permit valid identification and measurement of sources of radiation exposure that can produce health damage, defining the relationship between exposure and consequent health effects, and determining and defending acceptable limits to control radiation exposure in the work place and in the general environment. Insufficient information to reduce significant uncertainties results in public health protection and safety regulatory policies that can be either overprotective and therefore uneconomical or underprotective and therefore socially unacceptable. Significant uncertainties remain in some areas of radionuclide metabolism and internal dosimetry, dose-effect relationships and risk estimation, environmental pathways for public exposure, and radiological and dosimetric measurements in the work place.

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

To enhance the required coordination and exchange of information on research needs and programs, the NRC participates with these and other Federal agencies in the Interagency Radiation Research Committee established by the Secretary of Health, Education and Welfare (now Health and Human Services) upon instruction of the President by memorandum dated February 21, 1980. In addition, agency-to-agency communication and exchanges (e.g., workshops and symposia) on specific scientific issues are used to recognize the potential application of research to regulatory decisions.

Discussed in this chapter are metabolism and internal dosimetry, health effects and risk estimation, radionuclide pathways for radiation exposure of man, and occupational radiation protection.

8.1 Metabolism and Internal Dosimetry

The radiation dose to the human body following the introduction of radionuclides through inhalation, ingestion, or absorption through skin and wounds will depend on the physiologic and metabolic processes that determine the distribution and retention of radionuclides in the various body organs and on the nature (i.e., the type and energy) of the radiation emitted from the internally deposited radionuclides.

Results from such investigations of these nuclides provide the basic data found in national and international radiation protection publications. These data serve as references for Commission licensing decisions and regulations, particularly for the maximum permissible concentrations in air and water in Appendix B to 10 CFR Part 20, "Standards for Protection Against Radiation." They also serve as the means by which estimates of dose are made from bioassay data. Such bioassay data are vital to determining doses for an individual worker and thus compliance with NRC standards.

8.1.1 Major Regulatory Needs and Their Justifications

1. Reduction of the uncertainties in data on metabolic behavior of materials in the front end of the fuel cycle, to be the basis for regulatory guides and amendments to 10 CFR Part 20 (1987).†
   Justification: Materials generated during the production of reactor fuel that contain naturally occurring uranium, thorium, and, in some cases, their decay products expose workers to internal radiation doses, especially at mills and fuel fabrication plants. Unlike the data base for 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 heavy metals are not well understood. The amounts 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 and changes during the milling and processing of ores and the fabrication of fuel. Reducing these uncertainties will improve the dose and risk assessments and therefore the standards and other forms of 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: Although extensively investigated, uncertainties remain concerning intake parameters for some transuranic elements. For example, 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 not to behave the same as the laboratory materials used most often in research and upon which standards for protection are based. Current uncertainties must be resolved in order to verify the validity of present radiation protection standards for transuranics, particularly for application to the management of nuclear waste and the potential 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-26, 1977) in 10 CFR Part 20 (1989).†
   Justification: The ICRP-26 dose limitation system based on health risk necessitates practical calculational methods for summing internal and

external doses. For those cases where simplified approximate methods are not adequate, such as using specific metabolic data acquired for an individual, improved computer codes that allow flexibility in parameter values, including age and sex, need development.

8.1.2 Research Program Description

The strategy for this research effort is to rely on the data generated by the more extensive programs supported by DOE to the maximum extent possible and to fund only those research projects with objectives that are specific to NRC's regulatory needs and not otherwise addressed.

The NRC research program will develop metabolic models or provide metabolic distribution and retention values for materials found in the front end of the fuel cycle 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 also be obtained by external counting of persons exposed and retaining 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. a. Metabolic model for inhaled mixed oxides (1985).

   b. Metabolic model for inhaled yellowcake (1987).

2. Values for the gastrointestinal absorption factor for plutonium (1984), neptunium (1985), and other actinides (1986-1987).

3. a. Assessment of internal dosimetry code (1988).

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

8.2 Health Effects and Risk Estimation

Establishing appropriate radiation protection measures depends on the 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 based on 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 that will increase the confidence in health risk assessment and thereby improve the scientific bases for rules and licensing decisions, including those underlying safety goals.

8.2.1 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 (1988).
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 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 assessments and risk-reducing ALARA programs that depend on tradeoffs between individual and collective doses and cost of controls.

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. In addition, the biological effects of radon decay products at levels encountered in the vicinity of tailings piles are being estimated from data gathered on risk to uranium miners, the applicability of which to members of the general public is very questionable, particularly in regard to the risk of radiation-induced lung cancer for a nonsmoking population. 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 tailings 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.

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.

8.2.2 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 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 1984). Others are applying such procedures using dose-response relationships and risk coefficients derived from data on exposed populations (e.g., Hiroshima and Nagasaki survivors, U.S. uranium miners). Since the results of these investigations might have an impact on NRC's radiation protection standards, it is necessary to know the strengths and weaknesses of the methodologies applied.

A statistical reevaluation of the effects of exposure to radon decay products on lung cancer risk in nonsmoking uranium miners will begin in FY 1984. An epidemiological study of inhaled radon decay products and associated lung cancer risk in radium dial painters will also be initiated in FY 1984. These studies 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 products exposure estimates for nonsmoking populations rather than on total population statistics.

To aid in assessing risk associated with radon emanation at uranium mill sites, measurements of radon and radon daughters and determinations of daughter equilibrium values and of the unattached daughter fractions will be performed in and around mills, including mill tailings sites.

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

To improve the ability to predict the consequences of 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.

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. Dosimetric data for assessment of risk from radon and its daughters (1986).

c. Evaluation of magnitude of non-radon-related lung cancer risks in uranium miners (1986).

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

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).

8.3 Radionuclide Pathways for Radiation Exposure of Man

8.3.1 Major Regulatory Need and Its Justification

Updating, validation, and maintenance of procedures and models for predictive assessment of individual and population exposure for use in impact assessment, licensing, and standard setting (1989).
Justification: In order to predict the radiological impact of NRC-licensed operations, it is usually necessary to determine the movement of released radionuclides to man. Models and regulatory guides must be revised and updated to incorporate current data.

8.3.2 Research Program Description

The strategy is to rely primarily on compilation of data from the scientific literature supplemented where necessary by field and short-term laboratory studies. Specific information obtained will be (1) the updating of parameters used for liquid effluent radiological impact assessments for both freshwater and marine ecosystems, (2) the development of models to account for leaching of deposited radionuclides from the ground surface to make current models more realistic, and (3) the adaptation of models for uranium mill dose assessments for possible use in non-arid ecosystems. Eastern mills might differ significantly from western mills in the resuspension and deposition of particulates onto the ground and vegetation and leaching of tailings piles.

The major research products will be:

1. a. An improved data base of radionuclide concentration factors in freshwater and marine ecosystems (1986).

b. Estimates of radionuclide loss from surface soil due to leaching by rainwater (1985).

c. Estimates of the amount of radionuclide removal from ground surfaces by wind in non-arid areas (1986).

8.4 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.

8.4.1 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 as well as monitoring applications of existing dose-reduction technology so as to provide the basis for revising regulations or developing regulatory guides as warranted (1986-1988).†
   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 are needed to arrest this trend. Close, critical observance of industry and DOE research programs by the NRC is necessary to obtain assurance that these programs are adequately comprehensive and effective and to obtain the results from these programs 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 (1985-1989).
   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.

8.4.2 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. During FY 1983 and FY 1984, the funding level for such projects exceeds \$33.5M.* The NRC staff, supported as 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

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

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, inplant observations and measurements will be conducted at nuclear power plants undergoing decontamination. This will 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, handling, shipping, and disposal can be affected. Because of chelating agents, stability of the solidified material after disposal is also in question. For these reasons, the NRC needs to establish a position on the solidification processes used. Under a contract, laboratory tests on solidification processes and methods of degrading the chelating agents are now being performed. This program will be completed in FY 1985.

Health physics requirements at reprocessing plants and advanced reactors will be identified to facilitate future license reviews. A value/impact study will be conducted to evaluate a regulation that would require licensees to perform quantitative fit-testing for each worker using a respirator.

The major research products will be:

1. a. Evaluation of processes for dealing with chelating agents and for producing acceptable decontamination waste disposal forms (1985).

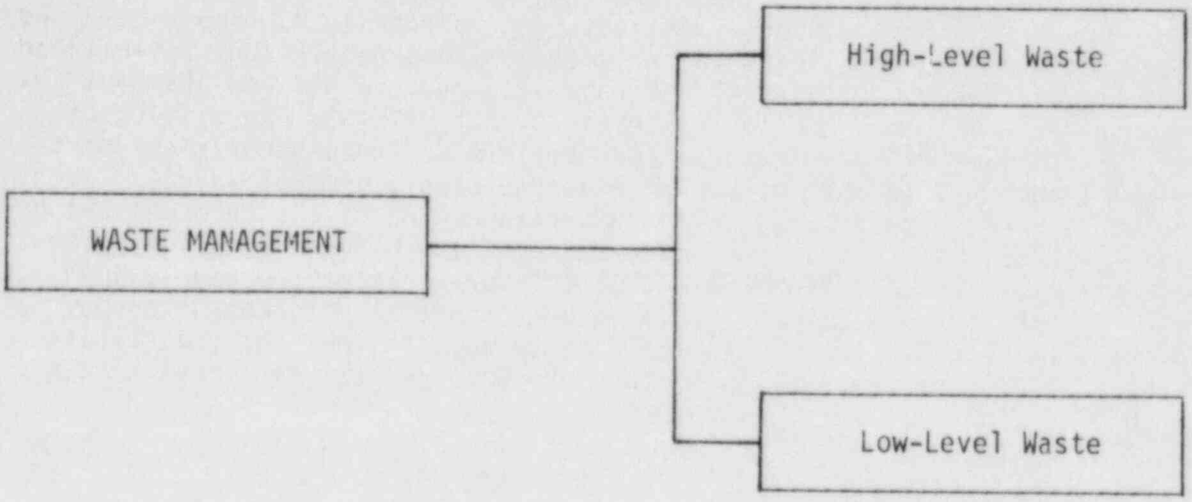
   b. Assessment of decontamination projects, including waste handling, at selected nuclear power plants with regard to overall effectiveness and safety (1985).

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

   b. Technical information for developing standard review plans to be used in licensing reviews for advanced reactors (1989).

Waste Management

9



FUND LEVELS
(Dollars in Millions)

	FY 1985	FY 1986	FY 1987	FY 1988	FY 1989
WASTE MANAGEMENT	$9.4	$10.8	$13.0	$14.0	$14.0

9. 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.

9.1 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 1983. 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 waste emplacement operations and postclosure. 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.

9.1.1 Major Regulatory Needs and Their Justifications

1. Capability to evaluate DOE's safety analysis reports (SARs) to assess compliance with 10 CFR Part 60 and 40 CFR Part 191, 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.

2. 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. 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).
<u>Justification</u>: 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.

9.1.2 <u>Research Program Description</u>

The strategy of the HLW research program is to identify and develop an understanding of basic phenomena and processes of HLW geologic disposal. This identification and understanding form the technical basis 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., cannister corrosion, 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); response of the engineered barrier system to changes in the repository environment (e.g., climatic changes, resaturation, wet-dry cycling); and models used to predict overall system performance. The HLW research program also addresses monitoring methods and instrumentation reliability.

The major research products will be:

1. a. Identification and description of the relationship between quality control in container manufacturing and expected long-term container performance (1985).

   b. Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes (1986).

   c. Report on field validation studies of radionuclide transport model (1988).

   d. Parameters and processes important to evaluation of effectiveness of borehole plugging and sealing (1984) and shaft sealing techniques (1987).

   e. Assessment of techniques for determining ground-water flow rates (1985).

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

3. a. Final report on effects of oxidation state on radionuclide mobility (1985).

   b. Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes (1986).

   c. Report on field validation studies of radionuclide transport model (1988).

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

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

9.2 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. States are either writing compacts to establish new shallow-land burial sites that will serve regional LLW disposal needs or are planning in-State sites. 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 mobility of radionuclides, and the associated uncertainties that enter into any demonstration that release criteria are met.

9.2.1 Major Regulatory Needs and Their Justifications

1. Methods and procedures for ensuring that LLW is properly classified (1986).
   Justification: Research is needed to improve basic understanding of the methods and abilities of existing analytical procedures to detect and quantify the activity of specific nuclides within particular waste streams and also to improve direct survey techniques for determining whether waste packages are in compliance with NRC requirements on classification.

2. 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.

3. Capability to assess facility engineering for land disposal of LLW for determining compliance with § 61.51 of 10 CFR Part 61 (1986).
   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.

4. 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 (1987).
   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.

5. 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.

6. 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 does not apply to disposal methods other than near surface. Such other methods may include aboveground or engineered facility disposal, deep-well disposal, and mined cavities. Research is needed (1) to determine the feasibility of alternative methods of LLW disposal, (2) to guide applicants on acceptable methods of demonstrating the suitability of alternatives, and (3) to develop the tools for staff evaluation of the suitability of alternatives.

9.2.2 Research Program Description

The strategy for shallow-land-burial research is to use field data and laboratory experiments to understand the phenomena that determine the performance of LLW shallow-land-burial 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 important uncertainties.

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 will reduce 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. Effectiveness and reliability of methods for monitoring releases of radioactivity to the unrestricted environment are important elements being tested. Information is being developed to provide technical bases for establishing exempt levels of radionuclides in wastes below which regulatory action would not be necessary. An assessment of existing information, input from the research cited above, and studies of existing facilities will be used to develop guidelines for closure of LLW disposal facilities and for long-term control.

The major research products will be:

1. a. Evaluation and characterization of waste streams and volume-reduction techniques to aid in establishing clear criteria for classification of LLW (1987).

   b. Evaluation of capabilities and strategies for surveying radionuclide content of waste packages (1988).

2. a. Evaluation of long-term performance of wastes and containers produced through currently available processing and containment technologies (1985).

   b. Evaluation of radionuclide containment characteristics of and criteria for volume-reduced wastes (1986).

3. Assessment of methods to ensure trench cap stability (1984).

4. Description of characteristics of operational and postclosure monitoring programs to assess performance and provide early warning of incipient failure (1987).

5. a. Assessment of interaction of radionuclides with soils to predict LLW disposal facility performance (1985).

   b. Description of coupled geochemical/hydrological phenomena relevant to migration of radionuclides from shallow-land burial facilities (1985).

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

   d. Assessment of chelating agents on ground-water migration (1985).

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

   b. Assessment of nonradiologic hazardous chemicals that are contained in or accompany LLW (1987).

Listing of Unresolved Safety Issues and TMI Action Items

A

Appendix A

LISTING OF UNRESOLVED SAFETY ISSUES AND TMI ACTION PLAN ITEMS

Some of the research described in the Long-Range Research Plan (LRRP) is being, or will be, conducted to contribute to the resolution of Unresolved Safety Issues (USIs) or of TMI Action Plan items. Listed below are those USIs and TMI Action Plan items referenced in the LRRP as well as the specific sections of the plan where the references may be found.

USI	Subject	LRRP Section
A-9	Anticipated Transients Without Scram	6.2
A-14	Nondestructive Examination	1.6
A-17	System Interactions (The research involves human aspects of systems interactions.)	4.4
A-46	Seismic Qualification of Equipment in Operating Plants	2.3
A-47	Safety Implications of Control Systems	5.3 6.2
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	6.5
A-49	Pressurized Thermal Shock	1.1 4.4 5.1 5.3

TMI Action Plan Item No.		
I.A.4.2	Long-Term Training Simulator Upgrade	4.5
I.C.9	Long-Term Plan for Upgrading Procedures	4.6
II.B.8	Rulemaking on Degraded-Core Accidents	4.2 6.11 6.12 6.13
II.C	Reliability Engineering and Risk Assessment	4.7
IV.C	Extending Lessons Learned to Licensed Activities Other Than Power Reactors	4.5 4.6

Setting Priorities for Research Program

B

Appendix B

SETTING PRIORITIES FOR RESEARCH PROGRAM

This appendix describes the method and its implementation currently being used by the Office of Nuclear Regulatory Research in setting priorities for research. The results of this year's efforts as well as plans for extending the effort to the element level are also discussed.

The method employed to generate priority ranks, the Analytic Hierarchy Process (AHP), was developed by T. L. Saaty. Reference 1 provides a good overview of this method, whereas the theory and many examples are provided in References 2, 3, and 4. Reference 5 is a report that describes the expansion of this method to setting priorities for research. The technique enables pairwise measures of relative importance (by how much is A more important than B?) to be translated into a single ranking scheme for many entities with a quantitative weight factor expressing the magnitude of the difference in rank for entities in the set. The result gives a comparative value only. However, the technique has been shown to translate assessments of pairwise relative values into an objective -- or at least repeatable -- common rank for the whole set of entities to be ranked in a wide variety of applications.

The AHP is a method of breaking down a complex ranking task into its component parts, arranging these components into a hierarchic order, assigning numerical values to each based on the relative importance of each component, and synthesizing these values to determine the priority of each.

The AHP was selected as the procedure for setting research priorities because (1) it is systematic and logical, (2) it is auditable, specifically identifying the factors and rationale entering into the priority setting, (3) the approach accommodates both hard (quantitative) and soft (qualitative) factors, and (4) it is analytical and produces numerical rankings as well as qualitative rankings.

The AHP builds upon the systems approach to problem solving. The structure of the system (the office research program) is organized into a hierarchy that begins with an overall objective (an integrated research program), then criteria to judge the research, research areas, and the research programs. An important assumption of the AHP is that the entities identified can be grouped into mutually exclusive sets. The entities of one level influence those of the level above and are influenced by those of the level below.

Figure B-1 shows the hierarchy that was used in setting the priorities. The first level has the focus or overall objective of an integrated research program. The next level consists of the criteria used to set the priorities for the research areas, and the third level contains the research areas.

The criteria used in the first phase were risk/uncertainty relevance and regulatory significance. Risk/uncertainty relevance is indicated by the degree to which the research will contribute to risk-reduction potential and the degree to which the uncertainties can be completely identified, their magnitudes estimated, and the uncertainties reduced. <u>Regulatory significance</u> is indicated by the scope of the issue being addressed by the research (generic vs. specific), the urgency and importance of the need identified by the user, the degree of outside concern (public, Congress, industry), and the relevance and timeliness of the research results with respect to resolution of the regulatory issue.

Ten research areas were identified. These areas correspond to the chapter headings of the FY 1985-1989 Long-Range Research Plan (one exception: Reactor Risk and Human Factors are herein listed separately while these two areas are combined in the LRRP in Chapter 4, "Reactor Operations and Risk"). Obviously, a thorough understanding of the content and utility of the program area is essential. The areas are (1) operating reactor inspection, maintenance, and repair, (2) equipment qualification, (3) seismic research, (4) risk analysis, (5) human factors, (6) thermal-hydraulic transients, (7) severe accidents, (8) advanced concepts, (9) radiation protection and health effects, and (10) waste management.

The next step is to determine the priorities of the entities within a level. To do this, a matrix is formed with the rows and columns labeled by the entities within the level. For example, for the entities associated with the Level 2 research areas denoted by A_i, the matrix would have the form:

$$\begin{array}{c c}
 & \begin{array}{ccccc} A_1 & A_2 & \cdot\ \cdot\ \cdot & A_{10} \end{array} \\
\begin{array}{c} A_1 \\ A_2 \\ \cdot \\ \cdot \\ \cdot \\ A_{10} \end{array} &
\begin{bmatrix}
1 & a_{12} & \cdot\ \cdot\ \cdot & a_{1,10} \\
a_{12}^{-1} & 1 & \cdot\ \cdot\ \cdot & a_{2,10} \\
\cdot & \cdot & \cdot & \cdot \\
\cdot & \cdot & \cdot & \cdot \\
\cdot & \cdot & \cdot & \cdot \\
a_{1,10}^{-1} & a_{2,10}^{-1} & \cdot\ \cdot\ \cdot & 1
\end{bmatrix}
\end{array}$$

The entities are then compared in a pairwise fashion, and a rank is inserted into the (A_i, A_j) position of the matrix as follows:

If A_i and A_j are equally important, insert 1
If A_i is weakly more important than A_j, insert 3
If A_i is strongly more important than A_j, insert 5
If A_i is demonstrably more important than A_j, insert 7
If A_i is absolutely more important than A_j, insert 9

By convention, a 1 is inserted into all diagonal positions of the matrix. The reciprocal of the number in the (A_i, A_j) position is inserted into the (A_j, A_i) position. Note that the comparison is always of a row entity, A_i, against a column entity, A_j. If A_j is preferred to A_i, the reciprocal is placed in the (A_i, A_j) position and the integer value into the (A_j, A_i) position of the matrix. After the $k(k - 1)/2$ pairwise comparisons (45 for each criterion) have been made, the "normalized" eigenvector corresponding to the maximum eigenvalue yields the weights that we desire. The method also provides a check on the consistency of the rankings.

The concept of comparing entities has the advantage of focusing on only two objects at a time and their relationship to each other. This method produces more information than simply ranking the objects individually since each object is methodically compared with every other.

For problems where there is no scale to validate the result, the process of pairwise comparisons can prove to be an asset because, in a sense, it is simpler in each of its steps than comparing all objects collectively.

To facilitate the prioritization activity, a worksheet was developed. A portion of the worksheet used is shown in Figure B-2. Each RES Division Director was asked to fill in the worksheet using the numerical rankings in Table B-1. The responses from each Division Director were then tabulated, and the geometric mean was calculated for each comparison. These rankings were analyzed using a computer program that calculated weights of the research programs within each criterion and the overall weights. These weights, summarized in Table B-2, give the relative ranking of each research area with respect to the criteria of risk/uncertainty relevance and regulatory significance.

The next step in the process of setting priorities is to prioritize the research elements within each research area. This will be done by the cognizant RES Division Directors and Branch Chiefs. The same criteria, as well as research impact, will be used as were used in setting priorities for the research areas. To be able to set priorities for the research elements according to the criteria, information is needed as a basis for the prioritization. Table B-3 identifies the sources of information to be used. This information may come from user offices, may result from PRAs, etc.

The same pairwise comparison process will then be used to develop the weights for each element within each area. The overall weights will be determined by multiplying the weight of an element by the area weight. These final weights may then be used to determine the overall priorities for the research elements for use in helping in budget preparation and in determining future research directions.

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Table B-1

SCALE FOR PAIRWISE COMPARISONS

Qualitative Comparison	Quantitative Value
Equally Important	1
Weakly More Important	3
Strongly More Important	5
Demonstrably More Important	7
Absolutely More Important	9

Table B-2

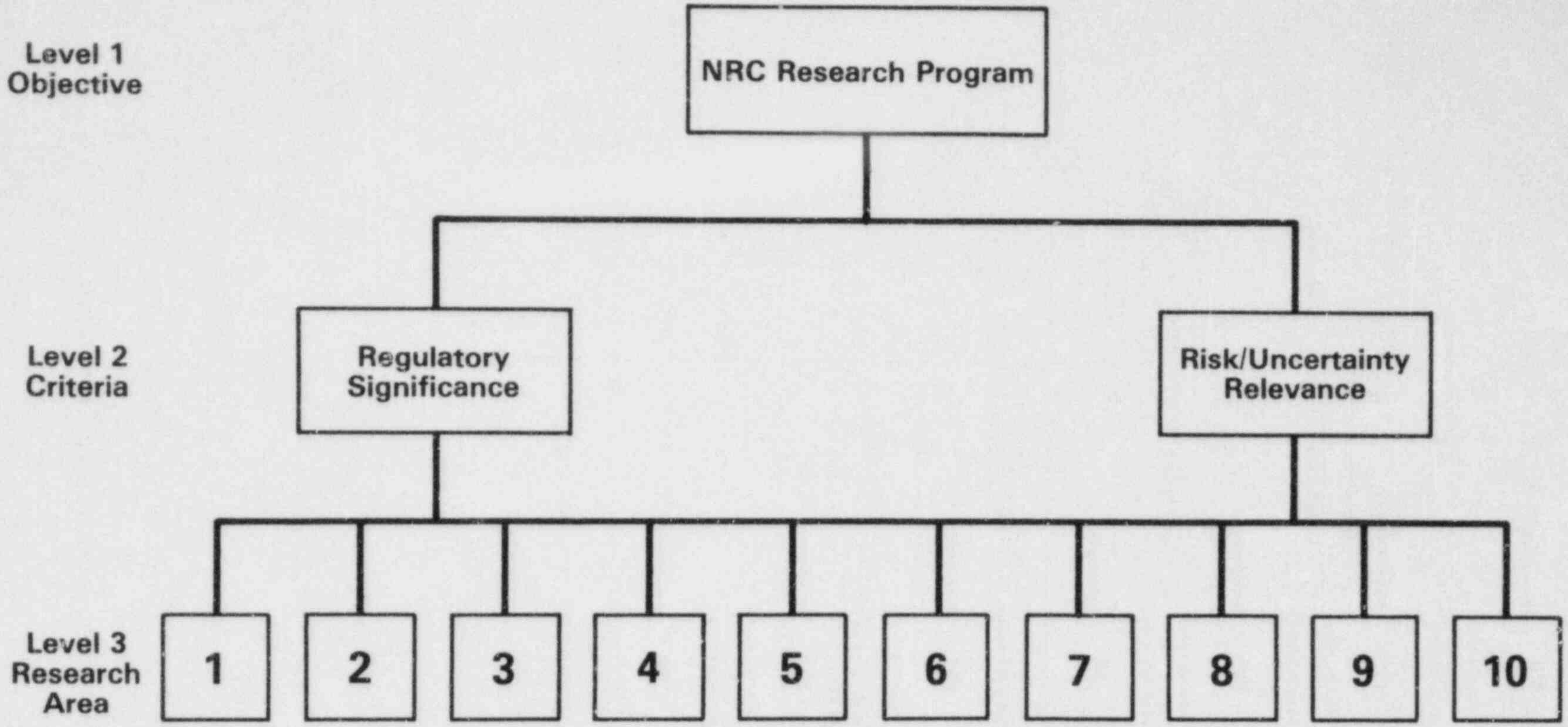
RESULTS OF SETTING OF PRIORITIES FOR RESEARCH AREAS

Research Area	Regulatory Significance		Risk Relevance		Composite	
	Weight	Rank	Weight	Rank	Weight	Rank
1. Operating Reactor Inspection, Maintenance, and Repair	8.49	6	10.11	4	9.30	4
2. Equipment Qualification	9.25	4	8.09	5	8.67	5
3. Severe Accidents	29.49	1	27.53	1	28.51	1
4. Thermal-Hydraulic Transients	5.49	8	7.44	7	6.46	8
5. Advanced Concepts	2.65	10	2.11	10	2.38	10
6. Risk Analysis	12.37	3	18.09	2	15.23	2
7. Human Factors	8.56	5	13.07	3	10.82	3
8. Seismic Research	7.76	7	7.74	6	7.76	7
9. Radiation Protection and Health Effects	3.21	9	2.37	9	2.79	9
10. Waste Management	12.71	2	3.44	8	8.08	6

Table B-3

INFORMATION SOURCES

Criterion	Information Sources
Regulatory Significance	Rankings of unresolved safety issues; rankings of regulatory needs; priorities set by management
Risk Relevance	Sizes of uncertainties and their impacts on overall risk uncertainties; PRA results on risk and uncertainty contributors and importances
Research Impact	Assessments of the potential reduction in uncertainties that can result because of the research; assessments of the likelihood of resolving the issue or the feasibility of the research



1. Operating Reactor Inspection, Maintenance, and Repair
2. Equipment Qualification
3. Severe Accidents
4. Thermal-Hydraulic Transients
5. Advanced Concepts
6. Risk Analysis
7. Human Factors
8. Seismic Research
9. Radiation Protection and Health Effects
10. Waste Management

Figure B-1 Research Hierarchy

Figure B-2

WORKSHEET

Criterion for Comparison: ______________________

Pair No.	Area A	Area B	Check ( ) the More Important Areas: A	B	Numerical Value of Comparative Importance*
1.	Operating Reactor Inspection, Maintenance, and Repair	Equipment Qualification			
2.		Severe Accidents			
3.		Thermal-Hydraulic Transients			
4.		Advanced Concepts			
5.		Risk Analysis			
6.		Human Factors			
7.		Seismic Research			
8.		Radiation Protection and Health Effects			
9.		Waste Management			

(Areas to be Compared (A vs. B); Compare the two areas A and B for the above criterion)

* See Table B-1 for a list of numerical values of comparative importance.

Research Program Outline

C

Appendix C

RESEARCH PROGRAM OUTLINE

Research Program

Research Elements

Typical Products

Chapter 1
Operating Reactor Inspection, Maintenance, 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.
- Metallurgy and dosimetry results from simulated vessel wall and void box experiments for pressurized thermal shock and embrittlement studies.
- 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, including the effects on the integrity of tubing of such factors as water and stress environment during normal operation and upset conditions, decontamination, and tube bundle cleaning.

- 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 deals with the effects on the structural integrity of piping of the water, stress, and temperature environment during service, including stress corrosion cracking, fatigue and cyclic crack growth, and toughness loss due to long-term aging at temperature.

- 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 the significance of aging in the capability to withstand seismic and dynamic stresses.
- Practical and cost-effective techniques for monitoring equipment for service wear effects.
- Criteria for evaluating surveillance, maintenance, and replacement programs for selected components.
- Information on technology, safety, and costs from the actual decommissioning of nuclear facilities.

1.5 Surveillance and Diagnostic Techniques. This research element provides research needed to evaluate surveillance and diagnostic techniques to help prevent undesirable plant transients or accidents and to help avoid damage to equipment important to safety in reactor systems.

- Methodology for arriving at optimum test frequencies for engineered safety feature actuation systems and reactor trip systems.
- Surveillance and diagnostic techniques for early detection and diagnosis of power plant anomalies.

1.6 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 study 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.

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 penetrations, cables, and motors.
- Guidelines for the design, manufacture, installation, operation, testing, and maintenance of pressure transducers and solenoid- and motor-operated control valves.
- Data on fire damage thresholds of safety equipment.
- Results of a set of benchmark tests yielding fire 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.
- Identification of significant environmental parameters.

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 equipment response, failure modes, and fragilities to determine safety margins available in existing equipment.
- Evaluation of safety margins available in existing mechanical and electrical equipment subjected to the Safe Shutdown Earthquake.
- Acceptable methodology for dynamic qualification of mechanical equipment.
- Criteria for extrapolating test results from one size component to another.

Research Program

Typical Products

Chapter 3
Seismic Research

This research program will study earthquakes, which can be the most severe of the natural hazards faced by nuclear power plants. In order to assess seismic risks and to establish appropriate regulatory equipments, it is necessary to determine the seismic hazard (earthquake magnitude and occurrence intervals) for a power plant site and to predict the response of the site and the facility to earthquakes within the range of magnitude appropriate to the site. Current estimates of seismic risk not only contain large uncertainties that stem primarily from a lack of earthquake records, failure experience, and fragility test data, but also include significant difficulties associated with analytically modeling such phenomena as soil-structure interaction and response of piping systems to seismic motions.

- Seismographic network data, geophysical data for determining crustal structure in areas of suspicious geologic structures, and data from the in situ stress measurement program in the Northeastern United States.
- Techniques for calculating site-specific response spectra.
- The changes in floor response spectra and damping for input motion resulting in linear and nonlinear behavior.
- Validation of the DESRA soil-failure code and information on the use and limitations of soil-structure-interaction computer codes; determination of the seismic risk contribution from dam and embankment failure.
- Risk-based recommendations for decoupling SSE and LOCA loads in design criteria for primary piping.
- Fragility data from equipment testing.
- Risk comparison of results from analyses of stiff and flexible piping.
- Comparison of failure conditions assumed using old methods with those that can be derived from current knowledge on seismic failure experience and fragility data.
- The effect of the new seismic design methods on seismic design.
- Quantification of the inherent seismic margins and associated uncertainties in the older designs.
- Estimates of the probability of a seismic event that would cause core melt and evaluation of associated uncertainties.

Research Program

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. Past efforts in this area have identified the man-machine interactions as an area of significant uncertainty and therefore a large contributor to risk. As a result, major emphasis in this program is on identification of opportunities for human error and of ways to improve plant reliability. 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.

Research Elements / Typical Products

4.1 Risk Assessment Methods Development. 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.
- Integrated methods for identifying and displaying quantitative uncertainties.

4.2 Methods Development for Risk Reduction. 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.

- Assessment of costs and risk-reduction potential of alternative safety features applicable to specific classes of LWRs.
- An evaluation of the feasibility of using PRA to improve reliability of existing plant systems.
- An evaluation of the likelihood of pressurized thermal shock causing a through-wall crack in a reactor pressure vessel.
- The predicted response of spent fuel shipping containers to the forces generated in severe accidents.

4.3 Methods Development for Acceptable Risk Level Maintenance. This research element will develop 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 and to provide the technical basis for future Commission actions relative to operating plants.

- Recommendations for a reliability assurance program.
- Summary of the results of demonstration and recommendations for reliability assurance program.
- Information for PRA analyses to aid in developing and setting of priorities for NRC inspection activities.

4.4 Human Factors Engineering. The research element will provide the technical bases needed by NRC to evaluate the man-machine relationships at information and control stations and in control rooms, to assess and recommend human factors standards and guidelines for new or improved designs affecting operating or maintenance personnel, and to establish criteria for regulatory applications of human factors engineering.

- Comprehensive data base reflecting operator and crew behaviors in plant evolutions and accident sequences.
- Effects of function allocation and automation on operator motivation, vigilance, and attitudes.
- Guidelines for control room and display, control, and communication systems.
- Effects of severe stress such as that due to seismic and other extreme events on operations personnel.
- Criteria for setting priorities for alarms, including the suppression of lower-priority alarms.

4.5 Licensee Personnel Qualifications. This research element will assess and develop or confirm the technical basis for establishing and evaluating the qualifications needed by licensee personnel to safely operate and maintain a nuclear facility and reduce human-related risk.

- Empirical data on the performance of nuclear power plant operators from training simulator experiments.
- Validated criteria for the qualifications, training, and examining of nuclear power plant operators.
- Methods to evaluate the effectiveness of licensee programs for qualifying nuclear power plant operators.
- Assessment of qualification and training for operations and support personnel at fuel cycle facilities.
- Basis for qualifications and training for unlicensed operators and support personnel at nuclear power plants.
- Operator training for severe accident management.

4.6 Plant Procedures. This research element will develop the technical basis for the methods and criteria used to assess and upgrade, where needed, plant operating procedures for nuclear power plants, fuel cycle facilities, waste management facilities, and processors and users of special nuclear material and byproduct materials.

- Methods for evaluating plant-specific emergency, abnormal, and normal operating procedures and surveillance, maintenance, and testing procedures for LWRs.
- Methods and criteria for assessing plant procedures and their presentation.
- The impact of computer diagnostics and automation on procedures and regulatory requirements.
- Methods for evaluating operating, surveillance, maintenance, and testing procedures for waste management facilities.

4.7 Human Reliability. This research element involves analysis of errors by nuclear power plant operations and maintenance personnel and their contributions to man-machine safety system failures.

- Aids for performing probabilistic risk assessments of crucial normal, transient, and accident precursor sequences involving human action.
- Computer-based model for developing human error probability data for operations and maintenance functions.
- A data bank combining human error data from various media with automated storage and retrieval techniques.

4.8 Emergency Preparedness. This research element will provide the technical basis for NRC regulatory actions needed to improve the capability of Federal, State, and local governmental authorities and licensees to mitigate the consequences of an accident at a nuclear facility.

- Evaluation of protective action decisionmaking.
- Evaluation of emergency action level identification.
- Operator action event-tree techniques for aiding emergency response.
- Basis for guidance on implementing emergency preparedness requirements for fuel cycle and material licensees.

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.

5.1 Separate Effects Experiments and Model Development. 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, including degraded core cooling.

- Thermal fluid mixing and pressure vessel fluid heat transfer models.
- Condensation heat transfer model.
- Flow blockage criteria.
- Simplified simulation of B&W loop for understanding separate effects phenomena and for formal design of the B&W experimental facility.

5.2 Integral Systems Experiments. 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 MOD-2B data used to assess steam generator tube rupture models.
- Semiscale data to assess calculations of steam line and feedwater line breaks.
- Integral natural circulation data from MIST.

5.3 Code Assessment and Application. 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.

- Final major versions of the advanced multidimensional two-fluid transient analysis codes TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1.
- Assessment of the codes TRAC-PF1/MOD1, RELAP5/MOD2, TRAC-BD1/MOD1, and TRAC-BF1 by DOE laboratories.
- Integration of system codes to benchmark and audit risk analysis methods.

5.4 Plant Analyzer and Data Bank. This research element involves improvements in computational techniques and the development of user-oriented features for using the codes in the form of an automated plant analyzer with plant-specific output displays. It also involves the acquisition and manipulation of plant data needed to develop input specifications for plant-specific analyses.

- General user versions of PWR system plant analyzer and BWR system plant analyzer completed and demonstrated.
- Installation of interface between plant analyzer and plant data bank.
- Plant secondary systems and control systems implemented in plant data bank.
- Use of plant analyzer and plant data bank to analyze transients in full-scale LWRs.

Research Program

Research Elements

Typical Products

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.

Accident Analysis

6.1 Accident Likelihood Evaluation
6.2 Severe Accident Sequence Analysis
6.3 Accident Management

- Evaluations of plant operating data, licensee event reports, and vendor data for component and system reliability.
- Evaluation of plant abnormal and emergency operating procedures for preventing or mitigating severe accidents.
- Plant response to multiple failure sequences identified by risk assessment programs and severe accident research.
- Evaluation of fission product pathways, transport rates and inventories in plant structures after severe accidents.
- Effectiveness of procedural and plant design changes to ensure acceptability of the consequences of an ATWS.
- Operator training for severe accident management.
- Procedures for recovery from potentially severe accidents.
- Operating procedures for ensuring containment integrity during severe accidents.

Fuel & Fission Products

6.4 Behavior of Damaged Fuel
6.6 Fuel Structure Interaction
6.8 Fission Product Release & Transport
6.10 Fission Product Control

- Report on NRU SFD full-length verification tests.
- Report on analysis of power burst facility tests.
- Report on examination of selected TMI-2 core samples.
- Assessment of the MELPROG and SCDAP-Mod 2 codes with power burst facility and annular core research reactor results.
- Verification of the CORCON and CONTAIN codes.
- Experimental data for modeling the interaction of hot solidified melts with concrete and the long-term cooling characteristics of solidified core melts.
- Data for benchmarking fission product release models from out-of-pile tests up to 2900 K.
- Data for benchmarking models for release of fission products from in-pile experiments.
- Improved version of TRAP-MELT fission product and aerosol transport code.
- Large-scale (Marviken) fission product and aerosol transport test package to be used to assess the TRAP-MELT code.
- A validated code for evaluating the PWR ice condenser.
- A validated code for evaluating the performance and effectiveness of PWR-BWR containment sprays under severe accident conditions.
- Input data for developing a code on hydrogen burning and its effect on aerosol concentrations and the performance of selected engineered safety feature systems.
- Evaluations of existing engineered safety feature systems.

Containment

6.7 Containment Analysis
6.9 Containment Failure Mode
6.5 Hydrogen Generation and Control

- Verification of the CONTAIN code.
- Integration interface between CONTAIN and a code such as TRAP-MELT for fission product and energy source data.
- Integration interface between CONTAIN and a fission product dispersion code for computing offsite doses.
- Results of experiments on the performance of containment valves and electrical penetrations.
- Results of tests on models of major penetrations.
- Comparisons of predicted capacities of containments under static and dynamic pressure loads with experimental results.
- Results of tests of containment models under simulated seismic loading.
- Analysis of plants for degraded core and severe accidents.
- Report on the hydrogen burn survivability of safety-related equipment in ice condenser and Mark III containments.
- Improved hydrogen mitigation systems.
- Preliminary hydrogen diffusion and flame acceleration models.

Risk

6.11 Risk Code Development
6.12 Accident Consequences & Risk Reevaluation
6.13 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.
- Improved consequence and risk evaluations of important accident sequences.
- Revised estimates of risk reduction and cost using updated information.

Research Program

Research Elements

Typical Products

Chapter 7
Advanced Concepts

This research program incorporates the safety research necessary to support NRC regulatory activities for fast-breeder reactors and gas-cooled reactors.

7.1 Fast-Breeder Reactors. This research element will keep the NRC abreast of the latest technology in fast breeder reactors by using foreign experimental results to assess and update NRC analytical methods.

- The COMMIX and SSC Codes validated against Phoenix and SNR 300.
- Results of experiments on propagation of an explosive fuel-coolant interaction in a molten fuel sodium mixture.

7.2 Gas-Cooled Reactors. This research element involves research and standards activities on general design criteria, siting criteria, and siting source term; the assessment of standards, codes, and regulatory guides; and the development of the physical data, computer codes, and design and engineering information needed for licensing high-temperature gas-cooled reactors.

- Background data for site suitability criteria.
- Requirements for the thermal barrier and liner cooling system.
- An advanced analysis code for HTGR fission product plateout and liftoff to supersede the SUVIUS code.

Research Program	Research Elements	Typical Products

Research Program

Chapter 8
Radiation Protection and Health Effects

This research program will provide technical and scientific information necessary to evaluate the health risks arising from various modes and degrees of radiation exposure and the effectiveness of various strategies and means for radiation protection and control in the work place and the general environment.

Research Elements

8.1 Metabolism and Internal Dosimetry. This research element will develop methods for calculating internal doses based on improved data on the metabolic behavior of transuranics and of materials from the front end of the fuel cycle.

- Metabolic models for inhaled mixed oxides and yellowcake.
- Values for the gastrointestinal absorption factor for plutonium, neptunium, and other actinides.
- Assessment of internal dosimetry code.
- Compilation of age- and sex-specific dose conversion factors.

8.2 Health Effects and Risk Estimation. This research element will provide basic data that will increase the confidence in health risk assessments for types of radiation and levels of exposure that might be encountered in the work place, in the environment, and following potential major accidents.

- Revised (or reaffirmed) values for neutron quality factor.
- Improved estimate of dose response function for radon daughter exposure/lung cancer risk in the general public, including nonsmoking populations.
- Verification and improvement of models for early mortality resulting from radionuclide inhalation.
- Models for morbidity resulting from radionuclide inhalation.

8.3 Radionuclide Pathways for Radiation Exposure of Man. This research element will update parameters used for assessments of the radiological impact of liquid effluents for both freshwater and marine ecosystems, develop models to account for leaching of deposited radionuclides from the ground surface, and adapt models for uranium mill dose assessments for possible use in non-arid ecosystems.

- An improved data base of radionuclide concentration factors in freshwater and marine ecosystems.
- Estimates of radionuclide loss from surface soil due to leaching by rainwater.
- Estimates of the amount of radionuclides removed by wind from ground surfaces in non-arid areas.

8.4 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.

- Assessment of decontamination projects, including waste handling, at selected nuclear power plants with regard to overall effectiveness and safety.

Research Program	Research Elements	Typical Products

Chapter 9
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.

9.1 High-Level Waste. This research element will identify and develop an understanding of basic phenomena and processes involved in the geologic disposal of high-level waste, including performance of the waste packages and engineered systems, geochemical interactions, effects of excavation and emplaced wastes on geological stability and isolation, and response of the engineered barrier system to changes in the repository environment.

- Identification and description of important parameters and functional relationships for hydrological and geochemical models for assessing radionuclide transport processes.
- Assessment of techniques for determining ground-water flow rates.
- Identification and description of methods for predicting long-term performance of waste packages (including waste form, container, and overpack).
- Parameters and physical relationships important to methods for evaluating long-term performance of backfill systems.
- The applicability of hydrothermal and geothermal data and analysis to waste isolation performance assessments.

9.2 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.

- Characterization of waste streams and volume-reduction techniques for use in classification of LLW.
- Long-term performance of wastes and containers produced by available technologies.
- Characteristics of operational and postclosure monitoring programs to assess performance and provide early warning of incipient failure.
- Assessment of the interaction of radionuclides with soils to predict the performance of LLW disposal facilities.
- Assessment of alternatives to shallow-land burial of LLW.

Research Utilization Report

D

Appendix D

RESEARCH UTILIZATION REPORT

INTRODUCTION

The Commission's 1984 Policy and Planning Guidance (NUREG-0885, Issue 3) calls for the Office of Nuclear Regulatory Research to:

> "... provide to the Commission an annual report which lists regulations likely to be substantively modified or substantiated by the research program. Target dates for review of these regulations and the completion of changes to them should be specified. The particular research programs that relate to each of these regulations should be identified. Any remaining research programs should be listed along with a brief explanation of their purpose. Resources allocated to the latter category should also be provided."

This appendix has been prepared to supplement the Long-Range Research Plan (LRRP) in fulfilling these requirements.

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 rulemaking 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 nine chapters, which correspond to the nine chapters of the LRRP. These contain listings of the regulatory products* of the research program completed in 1983 and targeted for completion in 1984. A tenth 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 1985 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 D-1 summarizes the changes in regulations that may result from the research described in the LRRP, i.e., research conducted in 1985 and beyond.

Table D-2 shows the funding and staffing levels at the section level of each chapter for the base year 1985.

*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. Four fracture mechanics computer codes, all applicable to licensing needs for analyzing structural integrity of reactor pressure vessels subjected to overcooling transients, cold overpressurization events, and other unanticipated transients (1983).

2. Verification that a small, finite surface flaw in a reactor vessel's inner surface will extend into a long shallow flaw before it extends in depth when subjected to a thermal shock, thus justifying part of presently used analytic model for evaluating the consequences of pressurized thermal shock (PTS) (1983) and validating presently used NRC methodology for predicting the effects on the structural integrity of reactor vessels subjected to a PTS scenario (1984).

3. New data on importance of the sensitivity of residual or alloying elements of pressure vessel steels to irradiation damage, to be used for revising Regulatory Guide 1.99 (1984).

4. 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 (1983, 1984).

5. Validation of benchmark dosimetry methodologies and data base used to predict neutron fluence and radiation damage in reactor pressure vessel steels (1983, 1984).

1.2 Steam Generators

1. Data provided to NRR to validate current licensing positions on:

   a. Leak rates from cracked steam generator tubes (1983).

   b. Burst pressures of corrosion cracked tubes (1984).

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

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).

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

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

4. 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 (1984).

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

   a. Improved ductile fracture mechanics analysis techniques for evaluation of cracked piping (1983).

   b. Ductile fracture properties for piping materials (1983, 1984).

   c. Pipe fracture tests to validate fracture mechanics analyses (1984).

   d. Evaluations of short-term and long-term fixes for stress corrosion cracking in BWR piping for revision of NUREG-0313 (1983, 1984).

   e. Data on susceptibility of piping materials to stress corrosion cracking and rate of crack growth in spectrum of BWR water chemistries (1983, 1984).

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

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

1.4 Electrical and Mechanical Components

1. Nuclear Plant Aging Research (NPAR)

   This relatively new research project has been initiated to investigate the effects of aging on plant safety. The two major issues to be addressed are:

   a. What aging and service wear effects are likely to affect plant safety?

   b. What methods of inspection and surveillance will be effective in detecting significant aging effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented?

2. 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).

3. Decommissioning

a. Nine technical reports containing data bases for evaluating contents of decommissioning plans for nuclear reactors (1983).

b. Technical report containing data base for rulemaking on post-accident cleanup insurance requirements for nuclear reactors (1983).

c. Proposed amendments to 10 CFR Parts 30, 40, 50, 51, 70, and 72 on decommissioning criteria for nuclear facilities (1984).†

d. Two technical reports containing data bases for evaluating contents of decommissioning plans for nuclear reactors and fuel cycle facilities (1984).

1.5 <u>Surveillance and Diagnostic Techniques</u>

This program evaluates surveillance and diagnostic techniques to help prevent undesirable plant transients or accidents and to help avoid damage to equipment important to safety in reactor systems.

1.6 <u>Nondestructive Examination</u>

1. Technical data and positions provided for IE Bulletin 83-02 and inservice inspection of BWR stainless steel piping (1983).

2. Technical report providing evaluation of techniques and recommendations for inspection of near-surface flaws in pressure vessels of interest to PTS evaluations (1983).

3. Data transmitted to NRR on NDE for sizing of IGSCC and fatigue cracks for use in safety analyses (1983, 1984).

4. Recommendations and test results transmitted to NRR on improved eddy current testing of steam generator tubing (1984).

2. EQUIPMENT QUALIFICATION

2.1 Qualification of Electrical Equipment for Harsh Environments

1. Regulation (§ 50.49 of 10 CFR Part 50) on environmental qualification of electric equipment important to safety to nuclear power plants (1983).†

2. Revision to Regulatory Guide 1.89 on environmental qualification of electric equipment (1984).

3. 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).

4. Fire Protection Research Program

   a. Fire test on 20-foot separation of redundant safety-related electrical cable trains to evaluate the fire protection provisions of Appendix R to 10 CFR Part 50 (1983).

   b. Evaluation of various fire suppression methods in extinguishing electrical fires, to be used in licensing decisions (1984).

   c. Assessment of the vulnerability of control room equipment to credible fires, to be used in licensing decisions (1984).

5. Regulatory guide on criteria for programmable digital computer systems of nuclear power plants: draft guide (1983), active guide (1984).

6. Draft regulatory guide on quality assurance of computer-based protection system software (1984).

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).

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 (1984).

3. SEISMIC RESEARCH

1. Technical report on the development of power spectral densities for site-specific seismic loads for use in partial resolution of USI A-40 and in revising Section 3.7 of SRP (1983).

2. Reliability Analysis of Structures (RAS) computer code for use in GESSAR licensing actions (1983).

3. Technical report on masonry wall failure analysis in nuclear power plants for use by NRR to evaluate utility responses to IE Bulletin 80-11 on masonry wall design (1983).

4. Technical report containing seismographic network data for use in licensing, PRAs, rulemaking decisions, and engineering research projects (1984).

5. Technical report on soil failure and soil-structure interaction (1984).

4. REACTOR OPERATIONS AND RISK

4.1 Risk Assessment Methods Development

1. Source term rulemaking (1984).†

2. Rulemaking on USIs such as Station Blackout (1983), PTS (1984), and ATWS (1983-1984). On the proposed PTS rule (§ 50.61 of 10 CFR Part 50), PRA research provided support to NRR in its development and helped the staff conclude that the proposed rule does not need to include special provisions for Babcock and Wilcox reactors.†

3. Rulemaking on hydrogen control (1984).†

4. Rulemaking on emergency planning (1984).†

4.2 Methods Development for Risk Reduction

1. Source term rule and associated regulatory implementation (1984) based on:

   a. Up-to-date evaluations of likelihood of severe accidents (1983, 1984).

   b. Up-to-date evaluations of consequences of severe accidents (1983, 1984).

   c. Integration of accident likelihoods and consequences into state-of-knowledge predictions (1983, 1984).

   d. Evaluation of cost-effectiveness of various means to reduce risk (1983, 1984).

2. 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).

4.3 Methods Development for Acceptable Risk Level Maintenance

The purpose of this research program is to support the development of methods for ensuring that the level of risk associated with a licensed facility is maintained at an acceptable level over the lifetime of the plant.

4.4 Human Factors Engineering

1. Proposed rule to add Human Factors Criterion to Appendix A, "General Design Criteria," to 10 CFR Part 50 (1984).†

2. Four technical reports on crew task analysis and man-machine system design concepts, to be used in reviewing human factors in control rooms of nuclear power plants (1983).

4.5 Licensee Personnel Qualifications

1. Two technical reports based on the systems approach to training, to be used in developing a review process for licensee personnel qualifications and training (1983).

2. Proposed Revision 2 to Regulatory Guide 1.8 on personnel selection and training (1984).

4.6 Plant Procedures

1. Technical report containing guidelines for emergency procedures for use in establishing a review and evaluation process for emergency operating procedures (1983).

4.7 Human Reliability

1. 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 (1983, 1984).

2. Final rule on fitness for duty of personnel with unescorted access to protected areas (1984).†

4.8 Emergency Preparedness

1. Proposed amendments to 10 CFR Parts 30, 40, 70, and 72 regarding emergency preparedness for fuel cycle and byproduct material licensees (1984).†

5. THERMAL-HYDRAULIC TRANSIENTS

5.1 Separate Effects Experiments and Model Development

1. Results from the multidimensional cooling of BWR channels in the Slab Core Test Facility in 1983 corroborated by the BWR TRAC code for use in helping NRR to establish the margin of conservatism in simpler 1-D analyses (1983).

2. 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 (1983, 1984).

3. Computer model of downcomer and cold-leg fluid mixing for use in evaluating pressurized thermal shock (1984).

4. Thermal-Hydraulic Systems Analysis handbook completed for use by AEOD and IE (1984).

5. 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).

6. Completion of modifications to the CITADEL code for use by NRR in performing iodine transport analyses (1984).

5.2 Integral Systems Experiments

1. LOFT tests have shown that rewet is a realistic phenomenon that will occur under the right conditions. Based on this information, one of the proposed revisions to Appendix K is to allow the calculation of rewet, using models justified by experimental data. The LOFT program has also shown that the heat transfer during refill and reflood is better than that allowed by Appendix K; this area of Appendix K is also proposed for revision (1983).

2. The steam generator tube rupture tests performed in the Semiscale facility provide data that are being used for code assessment and to assist NRC in verifying the adequacy of current licensing assumptions used in tube rupture recovery procedures (1983, 1984).

3. 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 (1983, 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 are being used to assess the capability of NRC's computer codes, primarily BWR TRAC (1983, 1984).

5. Assessment of advanced computer codes using LOFT thermal-hydraulic test data to provide confidence in analyzing similar events in full-scale LWRs (1984).

6. 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).

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 (1983, 1984).

2. Completion of TRAC-BD1 (1983). This version of BWR TRAC represents a nearly completed code that will be NRC's major tool for calculating most BWR transients (1983, 1984).

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).

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 (1983, 1984).

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 (1983, 1984).

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 (1983, 1984).

6.2 Severe Accident Sequence Analysis

1. Through the SASA studies, increased understanding of the capability of existing PWRs to remove decay heat using feed and bleed following loss of all secondary cooling to contribute to partial resolution of USI A-45 (1983, 1984).

6.3 Accident Management

1. Investigation of the effects of an operator's bringing liquid level to the top of active fuel for the Browns Ferry main steamline isolation valve closure. These studies have shown that liquid level control along with pressure control can reduce the power level to 3 percent without using the standby liquid control system (1983).

6.4 Behavior of Damaged Fuel

1. Results from the first two severe fuel damage experiments in PBF indicate that most physical models in severe accident fuel behavior codes correctly predict the observed phenomena. These results provide a data base for improved models of fission product release incorporated in the SCDAP code used in the source term reassessment (1983, 1984).

2. Completion of analysis of risk-dominant phenomenological uncertainties in severe LWR accidents (1984).

3. Results from damaged core coolability experiments in ACRR and analysis will provide a basis for determining the limits on accident recovery by core reflooding in severe accidents for use in severe accident policy decisions (1984).

4. Analysis of PBF tests of fuel that has reached clad destruction temperature but has not yet reached fuel-melting temperature, to contribute to the technical basis for policy decisions regarding severe accidents (1984).

5. An analysis of high-temperature out-of-pile fuel clad oxidation experiments to contribute to the technical basis for policy decisions regarding severe accidents (1984).

6. An analysis of applicability of LMFBR debris coolability models to LWR conditions to provide a basis for determining the limits on accident recovery by core reflooding in severe accidents for use in severe accident policy decisions (1984).

6.5 Hydrogen Generation and Control

1. A research program was conducted to assess the amount of hydrogen or other combustible gases that might be released during or after design basis LOCA from the corrosion of galvanized or other types of coatings typically employed in the containments of nuclear power plants. This will provide the NRC with an assessment of the amount of combustible gas that might be released during or after a design basis LOCA and the ability to determine if the release rates are in agreement with the ANSI 56.1 standards on hydrogen control in containment (1983).

2. A preliminary hydrogen transport code with a combustion model, to be used for analysis of degraded core and severe accidents for Sequoyah and Grand Gulf licensing reviews (1983).

3. Analysis of hydrogen explosion potential of three to five specific plants with various containment types. This work is directed toward resolving the issue of local and global explosions during severe accidents (1984).

4. 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 currently under Commission review (1984).

5. Data base for resolving licensing condition at Sequoyah concerning operability of Tayco igniter in water spray environment (1984).

6. 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).

7. 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).†

6.6 Fuel-Structure Interaction

1. Intermediate-scale tests designed to investigate coarse mixing have resulted in surface events of sufficient energy to expel melt from the water, as well as prevent some melt from entering the water. These tests will provide a better understanding of the probability and consequences of direct failure of containment from in-vessel steam explosions (1983).

2. A research information letter will be issued on basemat penetration rates to be used for severe accident risk analysis (1984).

3. A research information letter will be issued on heat generation and release to be used for risk analysis of containment performance during severe accidents (1984).

6.7 Containment Analysis

1. Publication of CONTAIN computer code, an integrated systems code used to analyze the thermal and physical loads expected under severe accident conditions. This code will provide the licensing staff with improved capability for assessing the adequacy of reactor containment systems with respect to potential threat from severe accidents (1983).

6.8 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. A data report for benchmarking fission product release models from irradiated LWR rods up to 2400K, the temperature at which all of the fuel cladding will be melted or oxidized (1983).

2. A data report on core-melt aerosol release models from large (10 kg) out-of-pile fuel bundle experiments (1983).

3. The mechanistic FASTGRASS computer code for fission product release from fuel (1983).

4. A data report on the behavior of prototypic aerosols in a condensing steam atmosphere for testing computer models in the aerosol code (1983).

5. Best-estimate severe accident source terms revised for use by the Commission to reassess the source term currently used (1984).

6.9 Containment Failure Mode

1. Draft general revision of Appendix J to 10 CFR Part 50 (1984).†

6.10 Fission Product Control

Technical reports dealing with:

1. Studies of fission product scrubbing within ice compartments (1983).

2. An investigation of fission product chemical forms (1984).

3. Data base assessment and suggested experimental program for fission product removal in ESF systems (1984).

4. Background information for predicting effectiveness of ESF system fission product retention (1984).

6.11 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) (1983, 1984).

2. Improved MARCH and MATADOR codes to support future site-specific evaluations of emergency planning needs and more general siting evaluations (1984).

6.12 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 (1983, 1984).

6.13 Risk Reduction and Cost Analysis

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 (1984).†

7. ADVANCED CONCEPTS

7.1 Fast-Breeder Reactors

1. Analyses and data provided for Chapters 4, 5, and 15 of the safety evaluation report for CRBR review (1983).

2. Results of a fuel aerosol simulant test (FAST) to be used as part of data base for LMFBR fission product source term (1984).

7.2 Gas-Cooled Reactors

1. Report detailing graphite failure criteria and failure mechanism models to support licensing evaluation of the Ft. St. Vrain reactor (1984).

2. First edition of the HTGR Licensing and Safety Evaluation Handbook (1984).

8. RADIATION PROTECTION AND HEALTH EFFECTS

8.1 Metabolism and Internal Dosimetry

1. Values for the gastrointestinal absorption factor for plutonium, to be used in revising 10 CFR Part 20 (1984).

8.2 Health Effects and Risk Estimation

1. Statistical models for hazard evaluation studies and revised neutron quality factors, to be used in revising 10 CFR Part 20 (1984).

2. Dosimetric data for assessing radon risk and estimating radon cancer risk in the public, in response to Commission request for uniform health risk estimates for use in regulations, regulatory guides, policy statements, and news releases (1984).

3. Improved models for mortality resulting from radionuclide inhalation (1984).

8.3 Radionuclide Pathways for Radiation Exposure of Man

This research project was not funded during the time period covered in this report.

8.4 Occupational Radiation Protection

1. External Dose Control

   a. Two technical reports and a proposed rule on accreditation of personnel dosimetry processors (1983, 1984).†

   b. Six technical reports on neutron measurements and revision of Regulatory Guide 8.14 on personnel neutron dosimeters (1983, 1984).

   c. Two technical reports on gamma measurements (1983).

   d. Two technical reports and draft regulatory guide on beta measurements (1984).

2. Internal Dose Control

   a. Two technical reports and two rule changes on respiratory protection (1984).†

   b. Three technical reports on bioassay techniques; laboratory accreditation performance standard and rule (1984).†

c. Six technical reports on programmatic aspects of bioassay; draft regulatory guide on applications of bioassay for tritium and Revision 1 to Regulatory Guide 8.22 on bioassay at uranium mills (1983, 1984).

d. One technical report on air sampling and Regulatory Guide 8.30 on health physics surveys in uranium mills (1983, 1984).

3. Four technical reports/manuals on training (1984).

4. Six technical reports on ALARA (1983, 1984).

9. WASTE MANAGEMENT

9.1 High-Level Waste

1. Proposed amendments to 10 CFR Part 60 related to unsaturated zone (1984).†

2. Technical reports containing:

   a. Model for assessing degradation of borosilicate glass waste forms based on surface kinetics (1984).

   b. Assessment of techniques for determining ground-water flow rate (1984).

   c. State of the art in HLW geochemistry research (1984).

   d. Definition of relationship between gas conductivity and geometry of natural fracture for unsaturated zone characterization (1984).

   e. Technical considerations of HLW disposal in unsaturated zone (1984).

3. Technical document to support licensing position on preliminary assessment of radionuclide transport as vapor through unsaturated fractured rock (1984).

9.2 Low-Level Waste

1. Technical reports containing assessment of:

   a. Methods to ensure trench cap stability (1984).

   b. Source term for migration (1984).

   c. Effects of bioavailability of reactor LLW resulting from chemical changes in wastes during transport through soil (1984).

   d. Field sampling designs and compositing schemes for cost-effective detection of spills and migration at contaminated sites (1984).

   e. Radionuclide migration through soils (1984).

   f. Unsaturated zone hydrology and transport of tritium through plant transportation stream at Maxey Flats (1984).

   g. Chemical characteristics of migrating radionuclides at commercial shallow land burial sites (1984).

   h. New York Nuclear Service Center in West Valley, New York, both geologically and hydrologically (1984).

10. 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.

10.1 Safeguards

1. Revisions to eight regulatory guides dealing with measurement methods for material control and accounting (1983, 1984).

2. Ten technical reports on plant physical protection (1983, 1984).

3. Two technical reports on material control and accounting (1983, 1984).

10.2 Meteorology

1. Three technical reports dealing with tornado climatology and hazard probability assessments (1983, 1984).

10.3 Hydrology

1. A technical report describing ground-water mitigative methods and interdictive strategies for severe accidents at nuclear power plants (1984).

10.4 Material Safety

1. Data on sabotage source terms in spent fuel supplied to licensing staff to be used as basis for reducing safeguard requirements for spent fuel shipments (1983).

Table D-1

SUMMARY SHEET
LRRP INDICATIONS OF REGULATION CHANGES
AS RESULT OF RESEARCH CONDUCTED IN 1985 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	1986	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	1986	2-8
2.3	Dynamic Qualification of Equipment - evaluation of proposed methods of qualifying equipment for design basis events	1986	2-10
3.0	Seismic Research - data concerning seismic zones in Eastern U.S.	1988	3-1
3.0	Seismic Research - information base for developing site-specific spectra	1987	3-1

LRRP Section	Subject of Research	Targeted Date	LRRP Page
3.0	Seismic Research - methods for handling uncertainties in assessing potential risk from seismic hazards	1989	3-1
4.2	Methods Development for Risk Reduction - determining adequacy of current standards for certain types of radioactive shipments to prevent potential high-consequence transport accidents	1985	4-5
4.6	Plant Procedures - technical basis for decisions on adequacy and effectiveness of nuclear power plant emergency, abnormal, and normal operating, surveillance, maintenance, and testing procedures	1988	4-13
4.6	Plant Procedures - technical basis for determining adequacy and effectiveness of operating, surveillance, maintenance, and testing procedures for fuel cycle facilities, waste management facilities, and SNM and byproduct material processors and users	1989	4-13
5.1	Separate Effects Experiments and Model Development - experimental data and analysis for revising post-CHF heat transfer correlation and fuel element blockage criteria (Appendix K)	1985	5-2
6.3	Accident Management - identification of plant system designs, configurations, or operational capabilities to reduce probability of plant damage from system failures, limit extent of damage, or mitigate consequences of severe damage	1985	6-5
6.3	Accident Management - criteria, procedures, and data to minimize effects of human error in design, operation, and maintenance	1985	6-5
6.3	Accident Management - improved procedures and methods for ensuring containment integrity in event of core meltdown and pressure vessel rupture	1985	6-5

LRRP Section	Subject of Research	Targeted Date	LRRP Page
6.5	Hydrogen Generation and Control - data from all areas of hydrogen research such as generation, ignition conditions, and mixing	1985	6-9
6.5	Hydrogen Generation and Control - assessment of survivability of safety equipment during hydrogen burn	1985	6-9
8.1	Metabolism and Internal Dosimetry - reduction of uncertainties in data on metabolic behavior of materials in front end of fuel cycle	1987	8-2
8.1	Metabolism and Internal Dosimetry - reduction of uncertainties in data on metabolic behavior of transuranic elements	1987	8-2
8.1	Metabolism and Internal Dosimetry - validation of methods for calculating internal doses used in implementing ICRP recommendations	1989	8-2
8.4	Occupational Radiation Protection - monitoring and evaluating research being conducted by nuclear power industry and DOE on ALARA engineering technology and monitoring applications of existing dose-reduction technology	1988	8-7
9.2	Low-Level Waste - capability to assess alternatives to shallow-land burial of LLW	1988	9-5

Table D-2

FUNDING AND STAFFING FOR BASE YEAR FY 1985
(DOLLARS IN MILLIONS)

		FY 1985	STAFF*
1.	OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR....	24.4	38
	1.1 Reactor Vessels....................................	9.9	
	1.2 Steam Generators...................................	1.4	
	1.3 Piping...	6.6	
	1.4 Electrical and Mechanical Components...............	3.1	
	1.5 Surveillance and Diagnostic Techniques.............	0.6	
	1.6 Nondestructive Examination.........................	2.8	
2.	EQUIPMENT QUALIFICATION................................	5.6	18
	2.1 Qualification of Electrical Equipment for Harsh Environments...	4.1	
	2.2 Qualification of Mechanical Equipment (Environmental)......................................	1.5	
	2.3 Dynamic Qualification of Equipment.................	0	
3.	SEISMIC RESEARCH.......................................	10.0	12
4.	REACTOR OPERATIONS AND RISK............................	15.5	60
	4.1 Risk Assessment Methods Development................	7.0	
	4.2 Methods Development for Risk Reduction.............	1.0	
	4.3 Methods Development for Acceptable Risk Level Maintenance..	2.1	
	4.4 Human Factors Engineering..........................	1.2	
	4.5 Licensee Personnel Qualifications..................	1.9	
	4.6 Plant Procedures...................................	0.4	
	4.7 Human Reliability..................................	1.3	
	4.8 Emergency Preparedness.............................	0.6	
5.	THERMAL-HYDRAULIC TRANSIENTS...........................	27.3	25
	5.1 Separate Effects Experiments and Model Development..	7.3	
	5.2 Integral Systems Experiments.......................	12.0	
	5.3 Code Assessment and Application....................	5.8	
	5.4 Plant Analyzer and Data Bank.......................	2.2	

*Full Time Equivalents

Table D-2 (Continued)

		FY 1985	STAFF*
6.	SEVERE ACCIDENTS	50.9	30
	6.1 Accident Likelihood Evaluation	1.7	
	6.2 Severe Accident Sequence Analysis	4.6	
	6.3 Accident Management	0	
	6.4 Behavior of Damaged Fuel	20.2	
	6.5 Hydrogen Generation and Control	3.2	
	6.6 Fuel-Structure Interaction	2.8	
	6.7 Containment Analysis	2.3	
	6.8 Fission Product Release and Transport	5.7	
	6.9 Containment Failure Mode	6.8	
	6.10 Fission Product Control	1.1	
	6.11 Risk Code Development	0.7	
	6.12 Accident Consequence and Risk Reevaluation	0.8	
	6.13 Risk Reduction and Cost Analysis	1.0	
7.	ADVANCED CONCEPTS	5.1	11
	7.1 Fast-Breeder Reactors	3.5	
	7.2 Gas-Cooled Reactors	1.6	
8.	RADIATION PROTECTION AND HEALTH EFFECTS	2.8	23
	8.1 Metabolism and Internal Dosimetry	0.7	
	8.2 Health Effects and Risk Estimation	1.5	
	8.3 Radionuclide Pathways for Radiation Exposure of Man.	0	
	8.4 Occupational Radiation Protection	0.6	
9.	WASTE MANAGEMENT	9.4	27
	9.1 High-Level Waste	6.2	
	9.2 Low-Level Waste	3.2	
	TOTAL	$151.0	244

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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
AI	Artificial intelligence
ALARA	As low as is reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated transient without scram
BOP	Balance of plant
B&PV	Boiler and pressure vessel
B&W	Babcock and Wilcox
BWR	Boiling water reactor
CEA	Commissariat à l'Energie Atomique, France
CHAP	HTGR system transient analysis code
CHF	Critical heat flux
COBRA	Computer program for coolant boiling in rod arrays
COMMIX	Three-dimensional, transient, 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
CP	Construction permit
CRAC	Code to calculate consequences of reactor accidents
CRAY	Type of computer
CRBR	Clinch River Breeder Reactor
CRT	Cathode ray tube
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
EAL	Emergency action level
ECC	Emergency core cooling
ECCS	Emergency core cooling system
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
ESFAS	Engineered-safety-feature actuation system
FAST	Fuel Aerosol Simulant Test
FASTGRASS	Code to model fission product release from fuel
FCI	Fuel-coolant interaction
FEMA	Federal Emergency Management Agency
FIST	Full Integral Simulation Test
FITS	Fully Instrumented Test Series
FP	Fission product
FRG	Federal Republic of Germany

GRS	Gesellschaft für Reaktorsicherheit (Society for Reactor Safety)
HDR	Heissdampfreaktor (a decommissioned steam reactor in West Germany where reactor safety experiments are performed)
HLW	High-level waste
HPI	High-pressure injection
HTGR	High-temperature gas-cooled reactor
HVAC	Heating, ventilating, and air conditioning
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
INPO	Institute of Nuclear Power Operations
IREP	Interim Reliability Evaluation Program
ISAP	Integrated Safety Assessment Program
ISI	Inservice inspection
IST	Integral System Test
JAERI	Japanese Atomic Energy Research Institute

LCD	Liquid crystal display
LER	Licensee event report
LET	Linear energy transfer
LICA	Low-Intensity Cobalt Array (a facility at Sandia used for studies on radiation and thermal aging)
LLW	Low-level waste
LMF	Large Melt Facility
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
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
MIMAS	Multifield Integrated Meltdown Analysis System (code)
MIST	Multiloop Integral System Test
ML	Manufacturing license
NDE	Nondestructive examination
NIH	National Institutes of Health

NPAR	Nuclear plant aging research
NPP	Nuclear power plant
NPRDS	Nuclear Plant Reliability Data System
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
ORECA	HTGR system transient analysis code
OTIS	Once-Through Integral System
PBF	Power Burst Facility
PCRV	Prestressed concrete reactor vessel
PISC	Program for Inspection of Steel Components
PKL	Small-scale integrated test facility in West Germany
PPG	Policy and Planning Guidance
PRA	Probabilistic risk assessment
PTS	Pressurized thermal shock
PVC	Polyvinyl chloride
PWR	Pressurized water reactor
RAP	Reliability Assurance Program
RAS	Reliability Analysis of Structures (computer code)
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
RTS	Reactor trip system
SAFT-UT	Synthetic aperture focusing technique for ultrasonic testing
SAR	Safety analysis report
SARP	Severe Accident Research Program
SASA	Severe Accident Sequence Analysis
SCC	Stress corrosion cracking
SCDAP	Severe Core Damage Analysis Package
SEISIM	Code to estimate probability of failure of structures and components and seismic risk
SFD	Severe fuel damage
SIMMER	Code to analyze course of core-melt accidents in LMFBRs
SIMQUAKE	Series of experiments to measure soil-structure interaction effects (EPRI)
SMACS	Code to estimate structural and component seismic responses and uncertainties
SPARC	Suppression Pool Aerosol Removal Code
SPDS	Safety Parameter Display System
SQUG	Seismic Qualification Utilities Group
SSC	Super Systems Code to analyze system transients in LMFBRs
SSE	Safe shutdown earthquake
SSMRP	Seismic Safety Margin Research Program
SUVIUS	Code to solve behavior of fission gases in primary coolant of gas-cooled reactors
TMI	Three Mile Island
TRAC	Code to 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
USI	Unresolved safety issue
VGES	Variable Geometry Experimental System

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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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2 Equipment Qualification

3 Seismic Research

4 Reactor Operations and Risk

5 Thermal-Hydraulic Transients

6 Severe Accidents

7 Advanced Concepts

8 Radiation Protection and Health Effects

9 Waste Management

A Listing of Unresolved Safety Issues and TMI Action Items

B Setting Priorities for Research Program

C Research Program Outline

D Research Utilization Report

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