NUREG-0961

Long-Range Research Plan

FY 1984-FY 1988

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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FY 1984-FY 1988

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U.S. Nuclear Regulatory Commission

Previous Reports in Series

NUREG-0740, "Long Range Research Plan for FY 1983-1987," Office of Nuclear Regulatory Research, published March 1981.

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 5, 1983

RECIPIENTS OF NRC'S LONG-RANGE RESEARCH PLAN

Enclosed is a copy of the Nuclear Regulatory Commission's Long-Range Research Plan (LRRP) for fiscal years 1984 through 1988 (NUREG-0961). This is the NRC's third published 5-year research plan.

The primary purpose of the LRRP is to be the basis on which agreement on basic research directions is established, that is, to identify the principal areas or questions that require research in their resolution and to provide insights as to the level of future resource requirements.

This year's plan has been restructured so that each research element shows

1. the strategy planned to accomplish the research (including resources required),

2. the major research product expected, and

3. the fiscal year of delivery of the product.

We welcome any comments you may have on the plan. Please send your comments to Mr. Edward M. Podolak, Chief, Program and Administrative Services Branch, Office of Nuclear Regulatory Research, Washington, D.C. 20555.

Sincerely,

Robert B. Minogue

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosure:
NRC's Long-Range Research Plan
(NUREG-0961)

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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*

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. There should be increased emphasis on using research results in the regulatory process and on getting research results that are useful. Staff should not engage in research merely to postpone tackling difficult regulatory issues.

Planning Guidance

1. In view of general budgetary considerations, the agency must be prepared to carry out its research mission with fewer resources. This can be accomplished through more business-like methods; consolidation of programs with industry, other agencies and foreign countries; and the elimination of marginal programs.

2. The research resources identified in NRC's budget should be allocated to support a balanced program between supportive research for regulatory needs, 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.

3. The highest priority for NRC research efforts will be light water reactor safety.

4. An advanced reactor concepts program will be maintained to provide a technical base on which to make specific CRBR licensing decisions and other advanced reactor concepts consistent with programs adopted by the Executive Branch and the Congress.

5. NRC will develop and maintain a long-range research plan directed toward areas of importance to the licensing and inspection

*This is the section on research in NUREG-0885, Issue 2, "Policy and Planning Guidance, 1983."

processes. The research plan will be revised and updated annually and will be subjected to agency-wide and Commission review. Research undertaken by the staff will be consistent with the long-range research plan.

6. The staff should prepare a 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 also be identified. Any remaining research programs should be listed along with a brief explanation of their purpose. Resources allocated to each of these categories should also be provided. This report is to be provided to the Commission by early 1983 and annually thereafter.

7. 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.

The senior management of user offices review and endorse 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.

1

Introduction

1. 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, future research will be based on smaller-scale 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.

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 plan is broadly distributed for review and comment. It 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.

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 5). 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. Pressurized Thermal Shock (Chapter 3). The safety issue is that embrittlement resulting from continued exposure to neutrons is making certain older reactor vessels more susceptible to overcooling transients. The regulatory issues are: How serious is the problem at specific plants? What corrective actions are required and how soon? What are the acceptance criteria? What is an acceptable basis for continued operation while corrective actions are being planned and implemented?

3. Risk Analysis (Chapter 8). 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.

4. Human Factors (Chapter 9). 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.

5. <u>Plant Aging (Chapter 2)</u>. 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.

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.

Table 1 shows the proposed funding levels for the major research program areas described in the LRRP for Fiscal Years 1984-1988. Tables 2 and 3 show the crosscuts (for FY 1984 and FY 1985 respectively) between LRRP program areas and the RES budget decision units.

Finally, the Commission believes that the nuclear industry and DOE 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 LRRP is distributed to DOE and to such industry groups as Electric Power Research Institute (EPRI). This is done 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.

Table 1

Long-Range Research Plan

FUNDING LEVELS

FY 1984-FY 1988

(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
PLANT AGING	$ 23.3	$ 23.0	$ 22.8	$ 15.4	$ 14.6
PRESSURIZED THERMAL SHOCK	1.1	1.0	0.5	0.3	0.3
EQUIPMENT QUALIFICATION	5.4	5.9	6.4	5.5	4.5
SEVERE ACCIDENT	51.8	42.9	42.1	42.0	30.4
THERMAL-HYD. TRANSIENTS	37.7	31.4	25.6	23.5	20.9
ADVANCED REACTORS	9.9	9.0	9.5	9.5	9.5
RISK ANALYSIS	9.4	12.0	17.9	15.7	10.1
HUMAN FACTORS	6.2	7.3	8.2	8.3	8.3
INSTRUMENTATION AND CONTROL	6.2	6.4	6.9	6.3	6.4
EXTERNAL EVENTS	12.5	16.0	12.7	12.3	12.0
RADIATION PROTECTION AND HEALTH EFFECTS	5.3	5.2	7.2	8.0	8.6
WASTE MANAGEMENT	9.3	10.4	15.5	14.9	14.2
SAFEGUARDS	1.0	1.0	1.0	1.0	1.0
DECOMMISSIONING	0.8	1.0	1.2	1.2	1.0
MATERIAL SAFETY	2.5	3.3	6.4	6.2	5.8
TOTAL	$182.4	$175.8	$183.9	$170.1	$147.6

Table 2

CROSSCUT

Long-Range Research Plan
vs.
Budget Decision Units
FY 1984

(Dollars in Millions)

	REACTOR FAC. ENGR.	FACILITY OPERATIONS	THERMAL-HYDRAULIC TRANSIENTS	SITING AND HEALTH	RISK ANALYSIS	ACCIDENT EVALUATION	LOCA	LOFT	ADVANCED REACTORS	WASTE MGMT.	TOTAL
PLANT AGING	$22.9		$0.4								$23.3
PRESSURIZED THERMAL SHOCK			0.5		$0.5	$0.1					1.1
EQUIPMENT QUALIFICATION	5.4										5.4
SEVERE ACCIDENT	3.5				4.5	43.8					51.8
THERMAL-HYD. TRANSIENTS			26.7				$11.0				37.7
ADVANCED REACTORS									$9.9		9.9
RISK ANALYSIS					9.4						9.4
HUMAN FACTORS		$4.8			1.4						6.2
INSTRUMENTATION AND CONTROL		6.1	0.1								6.2
EXTERNAL EVENTS	6.6			$5.9							12.5
RADIATION PROTECTION AND HEALTH EFFECTS		2.6		2.7							5.3
WASTE MANAGEMENT										$9.3	9.3
SAFEGUARDS		1.0									1.0
DECOMMISSIONING	0.8										0.8
MATERIAL SAFETY	0.6				1.9						2.5
TOTAL	$39.8	$14.5	$27.7	$8.6	$17.7	$43.9	$11.0	$0	$9.9	$9.3	$182.4

Table 3

CROSSCUT

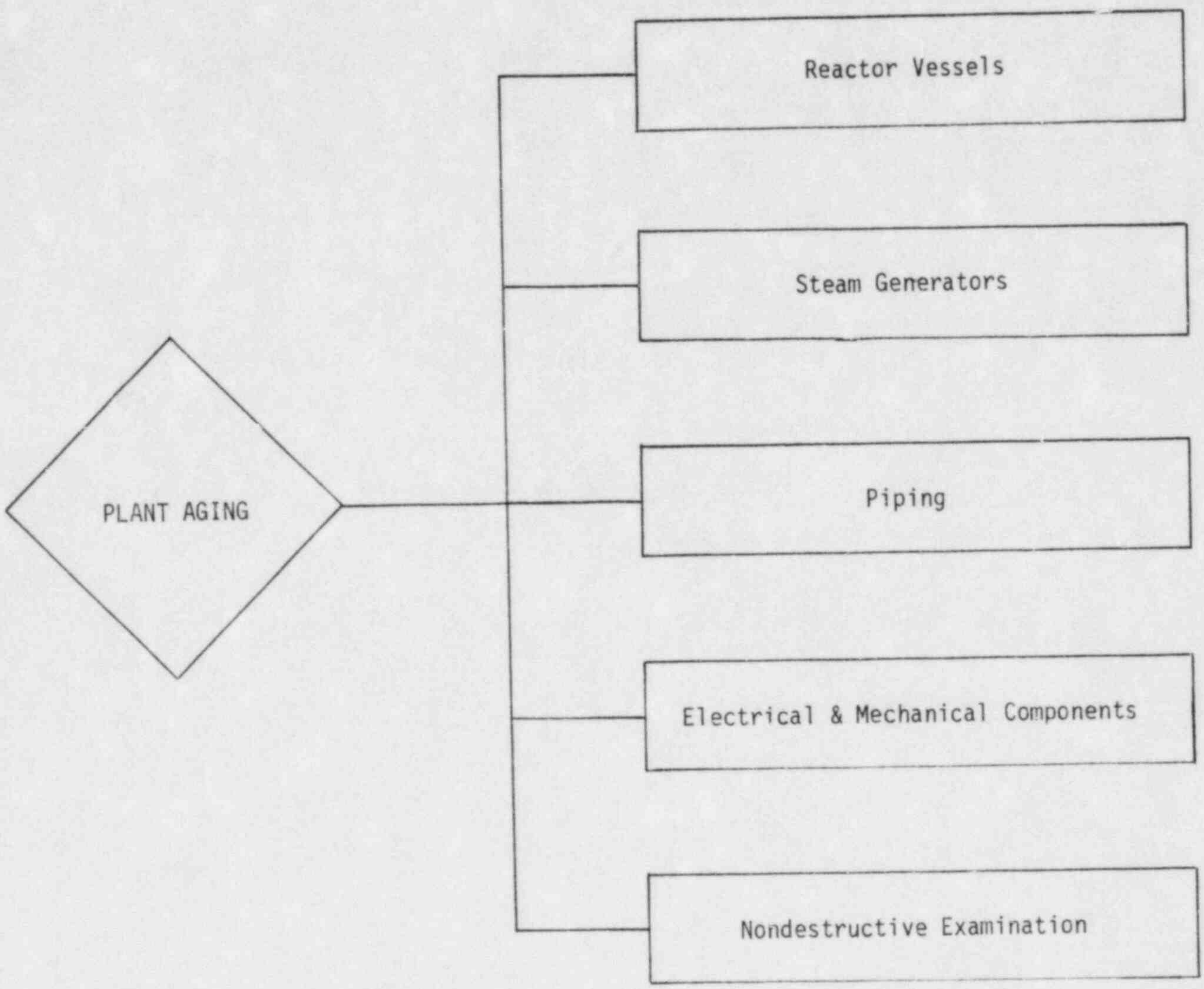
Long-Range Research Plan
vs.
Budget Decision Units
FY 1985

(Dollars in Millions)

	REACTOR FAC. ENGR.	FACILITY OPERATIONS	THERMAL-HYDRAULIC TRANSIENTS	SITING AND HEALTH	RISK ANALYSIS	ACCIDENT EVALUATION	LOCA	LOFT	ADVANCED REACTORS	WASTE MGMT.	TOTAL
PLANT AGING	$22.5		$0.5								$23.0
PRESSURIZED THERMAL SHOCK			0.5		$0.2	$0.3					1.0
EQUIPMENT QUALIFICATION	5.9										5.9
SEVERE ACCIDENT	1.5				2.0	39.4					42.9
THERMAL-HYD. TRANSIENTS			21.7				$9.7				31.4
ADVANCED REACTORS									$9.0		9.0
RISK ANALYSIS					12.0						12.0
HUMAN FACTORS		$6.8			0.5						7.3
INSTRUMENTATION AND CONTROL		6.2	0.2								6.4
EXTERNAL EVENTS	9.6			$6.4							16.0
RADIATION PROTECTION AND HEALTH EFFECTS		2.5		2.7							5.2
WASTE MANAGEMENT										$10.4	10.4
SAFEGUARDS		1.0									1.0
DECOMMISSIONING	1.0										1.0
MATERIAL SAFETY	1.3				2.0						3.3
TOTAL	$41.8	$16.5	$22.9	$9.1	$16.7	$39.7	$9.7	$0	$9.0	$10.4	$175.8

2

Plant Aging



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
PLANT AGING	$23.3	$23.0	$22.8	$15.4	$14.6

2. PLANT AGING

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. Aging and degradation have not, however, been ignored in the past. Section III, "Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code addresses some aspects of aging with detailed construction requirements, e.g., limits on material application, use of general corrosion allowances, and requirements to avoid metal fatigue. Section XI, "Rules for Inservice Inspection," of the ASME B&PV Code addresses aging and degradation with requirements for inspection, testing, repair, and replacement during the lifetime of components.

2.1 Reactor Vessels

This research applies to the structural integrity of pressure vessels especially as affected by irradiation embrittlement and growth of assumed cracks in service. Information developed here is also applicable to Section 3.1.

2.1.1 Major Regulatory Needs and Their Justifications

1. Validated methodology for fracture analysis of reactor vessels under accident conditions, to provide a basis for resolution of unresolved safety issue (USI) A-49 and the development of licensing criteria and a regulation (FY 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 resolution of USI A-49 and 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 (FY 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, nor 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 (FY 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 Charpys 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 (FY 1986).
Justification: Knowledge of 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.

2.1.2 Research Program Description

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 products will be:

- Unified elastic and elastic-plastic fracture mechanics analysis procedure for licensing evaluations of pressurized thermal shock in plants (1984).
- Large-scale verification of unified methodology by thermal shock and pressurized thermal shock experiments, PTSE-2 (1984).
- Embrittlement and annealing data for vessels being reviewed under Systematic Evaluation Program for licensing decisions (1984).
- Benchmarking methods for measuring and predicting fluence and embrittlement using reactor vessel surveillance capsules (1984).
- 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 (1984).
- Establishment of basis for an NRC position on ASTM standard on crack arrest testing specimen (1985).
- Confirmation of K_{Ic} curve for present practice steels in ASME Section XI (1985).
- Validation, by large-scale tests, of unified fracture mechanics methodology for licensing evaluation of pressurized thermal shock (1985).
- Technical basis for proposed licensing criteria and standards for in situ annealing of commercial reactor vessels (1985).
- Revision of fatigue curves for ferritic materials in ASME Section III (1986).
- Development of basis for fracture toughness requirements under conditions of thermal shock to reactor vessels (1986).
- Irradiated specimen test completed to validate ASME Section XI fracture toughness curves (1986).
- Final revisions recommended for the environmentally assisted fatigue crack growth curves in ASME Section XI (1987).

- Revisions recommended for ASME Section XI fracture toughness curves for newer higher-strength materials (1987).

- Final revision to 10 CFR Part 50 governing reactor fracture toughness requirements under both normal and accident operating conditions (1988).

2.2 Steam Generators

The research discussed below deals with corrosion, cracking, and degradation of steam generator tubing and sleeved tubing during service; integrity of tubing as degraded by the water and stress environment during normal operation and upsets over the long term as affected by decontamination and tube bundle cleaning and by other causes such as preheater vibration studies to detect loose parts in steam generators; and water chemistry control processes for primary and secondary coolant systems.

2.2.1 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 (FY 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 of degradation, cracks, etc., for requirements on tube plugging and additional inspection of tubes. If inspection "indications" translate to potential breaks, 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 (FY 1986).
   Justification: The only way to predict tube integrity is from knowledge of the signal taken from inspection. Thus, signal errors must be reduced or eliminated, and signals must be carefully correlated to exact defect size, as well as to measurements of tube burst strength.

3. Work to supplement and to evaluate industry efforts on influence of water chemistry and stress factors on corrosion and cracking (FY 1986).
   Justification: Knowledge of the relationships among water chemistry, metallurgy, and stress factors, including geometric details, is necessary for the NRC to evaluate industry positions to mitigate corrosion.

4. Long-term effect of decontamination, cleaning operations, and repairs on tube integrity, to provide a basis for approval of applications (FY 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. Because the long-term effects are not known, they must be independently studied so that informed decisions can be made on applications from utilities.

2.2.2 Research Program Description

The strategy for research in this element on steam generators is to use a retired-from-service generator as a test bed and validation tool for inspection procedures, integrity limits and margin of safety against burst and collapse, and study and removal of crud and contamination, all of which will support and validate licensing criteria for tube inspection and plugging and the water chemistry for generator operation. 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 other countries, including France, Italy, Japan, Taiwan, and Canada, have either joined or plan to join the work through financial contributions.

The approach is to study and test tubes from a retired-from-service steam generator 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. The research program also provides for evaluating the effectiveness of various methods for decontamination of steam generator entry boxes and tubes, as well as for cleaning and removal of crud, etc., from the secondary side. The longer-term safety and integrity implications of such treatments are examined to ensure that licensing of such procedures will not create future problems. Finally, studies will be performed to evaluate the influence of various constituents of cooling water composition on corrosion, cracking, and other generator performance factors.

The major research products from this element will be:

- o Validation of results by current and advanced NDE through examination of removed tubes (1984).
- o Validation of models for licensing evaluation and prediction of stress corrosion cracking (1984).
- o Engineering data on new technologies for controlling coolant chemical impurities (1984).

- o Correlation of remaining tube integrity with NDE to validate regulatory guide inservice inspection (ISI) plans and tube plugging criteria (1985).

- o Demonstration of generator cleaning and decontamination as basis for action on licensing applications (1985).

- o Implementation of licensing procedures for enhanced ISI plans and tube integrity and plugging criteria (1986).

- o Analytical techniques for evaluating licensee-proposed chemical adjustment processes (1986).

- o Water composition and corrosion data for validating licensing criteria for generator operation (1987).

- o Recommendation for licensing criteria for secondary-water composition and corrosion inhibition (1988).

2.3 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.

2.3.1 Major Regulatory Needs and Their Justifications

1. Experimentally validated analysis methodology for capacity of flawed and degraded piping during normal operation, accidents, and earthquakes and validation of the leak-before-break concept, to provide the basis for a regulatory guide and changes to the regulations and the standard review plan (FY 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. 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.

2. Independent basis for evaluating factors causing stress corrosion cracking in stainless steel piping and welds, to be used in revising the standard review plan (FY 1987).
   Justification: Decisions are regularly required on the "fixes" proposed to eliminate stress corrosion cracking in stainless steel piping. Background information is needed for independent evaluation of the fixes. In particular, procedures have been developed in the U.S. 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.

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 (FY 1986).
   Justification: Knowledge of rate of growth of cracks 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 safety of piping during normal operations or accidents.

4. Data base for evaluating toughness loss in cast austenitic stainless steel from long-time aging at reactor operating temperature, to be the basis for developing a regulatory guide and revising the standard review plan (FY 1987).
   Justification: A certain level of toughness is required to ensure safety in piping, especially to resist failure if flaws should develop in service and under accident loading. Long-term time-at-temperature can cause the metastable austenitic stainless steel to transform to lower states that are less tough. 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.

2.3.2 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 light-water-reactor (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 will be initiated in FY 1983. 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 reduce 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.

The major research products will be:

- Evaluation of pipe cracking predictive models, proposed fixes, and weld repair criteria, and development of regulatory positions and licensing criteria (1984).
- Initial findings on toughness of cast stainless steels transmitted to NRR for use in evaluating pipe cracking incidents (1984).
- Regulatory positions and licensing criteria proposed for repair welding and repair of stainless steels (1985).
- Experimental validation for elastic-plastic fracture mechanics analyses for use in developing position on leak before break (1986).
- Computerized data base on piping materials transmitted to NRR for use in licensing evaluations (1986).
- Licensing criteria proposed for establishing limits on environmental variables to control pipe cracking in LWR piping systems (1986).
- Licensing criteria proposed for prevention of toughness degradation due to aging in LWR stainless steel piping materials (1987).
- Technical basis for licensing decision on acceptance of leak before break in LWR piping systems and initiation of rulemaking on leak before break (1987).

2.4 Electrical and Mechanical Components

This research applies principally to the time-related degradation of electrical and mechanical components during service and the potential impacts of degradation upon public safety. Specifically, this research should develop methodologies to identify such potential impacts on safety, including the prevention or correction procedures, well in advance of their actual occurrence.

2.4.1 Major Regulatory Needs and Their Justifications

1. Identification of significant component/environment aging mechanisms with respect to potential risk to public safety (FY 1984).
   Justification: The initial identification of significant component/environment aging mechanisms is essential to the formulation of subsequent research programs and priorities on methodologies to prevent or mitigate the effects of such aging mechanisms on mechanical and electrical component operability. This identification is needed to focus the research efforts in areas of greatest potential impact on safety.

2. Validated methodology to predict significant adverse aging mechanisms during the installed life of components, to provide the bases for regulatory guides by use of which decisions can be made on component maintenance and replacement needed to ensure safety (FY 1986).
   Justification: It is better to anticipate aging and degradation phenomena in a timely manner and to correct, mitigate, or otherwise

eliminate any aging problems before they seriously affect public safety. Thus, there is a need for rigorous methodologies that can predict by simulation, analysis of experience, or surveillance the onset of various aging phenomena well in advance, especially in view of the progressive aging of an ever-increasing number of operating plants. Component aging effects to be predicted would include loss of operability during normal operation, loss of operability of components needed during accident or other emergency situations, summary mechanical failure due to metal fatigue, loss of fire-resistant capabilities, and loss of ability to withstand accident environment or seismic stresses.

3. Evaluation criteria to assist the staff in determining the validity of licensees' safety analyses of the potential effects of age-related degradation on component operability and to provide a basis for revising the standard review plan (FY 1987).
   Justification: NRC staff reviewers must be capable of evaluating licensees' safety analyses of potential risks resulting from age-induced component inoperability. This capability can only be in the form of rules and regulatory guides whose provisions are firmly based on research in support of their technical provisions.

2.4.2 Research Program Description

The strategy for this research is to develop methodologies to predict (and prevent or mitigate) in a timely manner the onset of significant component aging phenomena that can adversely affect public health and safety. Ongoing research on accelerated aging simulation techniques will support a limited portion of this element.

This is a new research program, the initial phase of which will culminate in the completion of a scoping study 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.

Following completion of the scoping study in FY 1983, specific research programs will be conducted during FY 1983-1986. Based on preliminary results of the scoping study, including information generated during the workshop on nuclear power plant aging held in August 1982, it appears that feasibility research in the following areas will be initially beneficial:

1. Use of failure data to identify component aging trends and potential failure mechanisms (e.g., metal fatigue).

2. Development of more sophisticated surveillance techniques to predict potential failures of components to operate.

3. Development of systematic preventive maintenance procedures.

4. Use of data collected at older plants (lead plants) to predict the onset of component failures at the newer plants.

The major research products will be:

- o Initial scoping study (1983).
- o Initial feasibility studies in the recommended areas of research (e.g., the feasibility of using licensee event reports (LERs) and related data to predict the onset of component failures) (1983-1984).
- o Research in those areas deemed feasible (1986-1987).
- o Validated licensing criteria on acceptable methodologies to predict and prevent or mitigate significant component age-related failures (1987-1988).

2.5 Nondestructive Examination

This research applies to the validation of reliable, reproducible nondestructive examination (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.

2.5.1 Major Regulatory Needs and Their Justifications

1. Documentation of 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 and provide the basis for revising regulatory guides and for recommendations for updating Section XI of the ASME Code (FY 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 (FY 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.

2.5.2 Research Program Description

The strategy for research in NDE 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 (Plate Inspection Steering Committee) 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. This approach has already yielded a series of recommendations for changes to improve the code procedures for ultrasonic testing. The round-robin approach 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 described in Section 2.2. 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. 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.

The major research products from this element will be:

- Criteria for new licensing position on use of acoustic emission (AE) for leak detection in hydrotests (1984).

- Recommendations for improvement of ASME B&PV Code, Section XI requirements for ultrasonic inspection of vessel plate and forging, for electromagnetic methods for through-weld and stainless steel inspection, and for multifrequency eddy current testing of steam generator tubes (1984).

- Code acceptance of continuous AE monitoring for cracks and validation leak monitoring by AE for licensing use where conventional methods cannot be used (1985).

- 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).

- Code acceptance for unified inspection requirements for vessels and piping, for the improved SAFT-UT method for flaw evaluation and detection, and for continuous AE leak monitoring (1986).

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

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

- Code acceptance of recommendations for improved inservice and continuous monitoring inspections (1988).

3

Pressurized Thermal Shock

3

PRESSURIZED THERMAL SHOCK

FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
PRESSURIZED THERMAL SHOCK	$1.1	$1.0	$0.5	$0.3	$0.3

3. PRESSURIZED THERMAL SHOCK

The safety issue here is that aging is making certain older PWR vessels more susceptible to overcooling transients. Irradiation embrittlement raises the temperature below which a reactor vessel behaves in a brittle fashion. Ongoing research and surveillance results establish (1) the neutron fluence and energy spectrum received by reactor vessels and (2) the rise in nil-ductility-transition temperature due to neutron fluence and spectrum and many other parameters such as copper impurities in the welds. Taking into account the uncertainties in these results, the staff calculates (conservatively) that some of the older vessels with high copper impurities in welds now have nil-ductility-transition temperatures above 200°F, and several are projected to rise above 270°F during the life of the plant. Also, the staff notes that reactors have experienced overcooling transients to temperatures below 270°F. A probabilistic analysis is needed to better understand the severity of this safety issue and how the severity is increasing with plant operating life. The staff estimates that for a reactor vessel with axial welds whose nil-ductility-transition temperature is conservatively determined to be 270°F (none is that high yet), the likelihood of vessel failure is less than 10^{-5} per reactor-year. Some improvements in operating procedures, operator training, and flux reductions may be appropriate now. However, the low estimated frequency of vessel failure allows time for research to reduce uncertainties and provide results for staff use so that regulatory requirements for long-term resolution of this complex unresolved safety issue (USI) A-49 may be carefully determined.

Additional possible corrective measures that the industry and the staff are considering to keep the likelihood of vessel failure low include improving instrumentation and control systems to reduce the frequency of overcooling transients, heating the emergency core coolant and emergency feedwater, changing fuel-loading schemes to reduce neutron flux at the vessel, in situ annealing of the reactor vessel, and possible reactor shutdown before full design plant life.

The regulatory issues are: How serious is the problem at specific plants? What corrective actions are required and how soon? What are acceptance criteria? What is an acceptable basis for continued operation while any corrective actions are being planned and implemented?

To resolve these issues, the staff is developing a two-phase regulatory program. The near-term resolution of USI A-49, now undergoing Commission review, would establish a screening level for nil-ductility-transition temperature. Three years before a plant is conservatively calculated to reach the screening limit, the licensee would report to NRC a plant-specific evaluation of pressurized thermal shock (PTS) overcooling event sequences, vessel material properties, fracture mechanics analysis, flux reduction program, inservice inspection, plant modifications, operating and training improvements, in situ annealing, and the basis for continued operation. In each of these areas, the definition of what will be acceptable to NRC remains to be developed in the second (i.e., longer-term) phase of USI A-49 resolution.

To help the staff determine these licensing requirements and to provide a basis for possible future rulemaking, research is planned in the areas of metallurgy, fracture mechanics, and inservice inspection (see Chapter 2) and in the areas of thermal-hydraulic analysis and risk analysis described below.

3.1 Major Regulatory Needs and Their Justifications

1. Development of TRAC and/or RELAP-5 models of representative plants of Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse design, including secondary and control systems pertinent to the behavior of overcooling transients, for use in licensing audit calculations of PTS (FY 1983-1984); calculation of the temperature and pressure of reactor coolant in the downcomer during selected overcooling transients to be used in the probabilistic analysis described below (FY 1983); followup transient analyses to determine the effects of new information such as results from other research on human factors and safety implications of control systems (FY 1984-1988).
   Justification: Previous NRC analyses have bounded overcooling transients. Best-estimate calculations are needed to estimate the most likely sequences. Therefore, the pertinent parts of the secondary and control systems must be modeled and their effects calculated for use in the probabilistic analyses described. Secondary and control systems have not previously been modeled on TRAC and RELAP-5. The results of these transient calculations will provide input to the probabilistic analysis of vessel integrity described in Need 3 below.

2. Preliminary correlation of the degree of coolant mixing in the downcomer during small-break LOCAs for near-term resolution of USI A-49 (FY 1982); detailed correlations and analyses as input to the probabilistic analysis described below (FY 1983) and for longer-term resolution of USI A-49 (FY 1983-1984).
   Justification: The degree of coolant mixing is an important uncertainty in assessing the severity of PTS. EPRI has conducted a series of experiments to model the mixing of cold emergency core coolant (ECC) with the warmer coolant in the cold leg and downcomer, and NRC is cooperating with EPRI to extend these experiments to include the heating effects of the vessel wall. The results will serve as input both to the probabilistic risk analysis and to establishing acceptable methods for licensee PTS analyses.

3. Estimation of the likelihood of reactor vessel failure due to PTS; identification of important sequences, uncertainties, operator actions, and control features; comparison of risk-reduction effectiveness of alternative corrective measures. The results will support the longer-term resolution of USI A-49 and licensing decisions (FY 1983-1984).
   Justification: This research integrates research results on fracture mechanics, metallurgy, and thermal hydraulics in order to develop plant-specific methods for probabilistic assessment of the severity of PTS and to provide trial applications of these methods. This information will support the long-term resolution of USI A-49 by helping the staff to determine, for those plants that reach the screening limits, what corrective measures are required, what are NRC's acceptance criteria, and what are acceptable bases for continued operation.

4. Estimation of the risk to the public, i.e., expected consequences, of PTS (FY 1984).
   Justification: This information will form part of the basis for possible rulemaking.

5. Metallurgical research to improve fracture mechanics methodology, to improve the data base on fracture toughness and crack arrest toughness of irradiated weld metal and clad pressure vessel steel, to improve neutron dosimetry and vessel surveillance, and to improve and assess methods for periodic inservice inspection of vessels. (Regulatory products and target dates are provided in Chapter 2, "Plant Aging.")
   Justification: Results of this research on materials behavior will be used directly in developing a long-term resolution for PTS and, in addition, the data and methods will be used in probabilistic risk analysis research described below. This materials research, the major research effort on PTS, is described in Chapter 2 and is not repeated here.

3.2 Research Program Description

The first step of the plan for this project is for the study team (Oak Ridge National Laboratory (ORNL), Idaho National Engineering Laboratory (INEL), and Los Alamos National Laboratory (LANL)) to obtain from selected utilities plant data pertinent to overcooling transient behavior and initiating events. ORNL then performs an event-tree analysis to delineate event sequences that could lead to overcooling and estimates the frequency of occurrence of these sequences. Some of the important sequences are selected for detailed thermal-hydraulic analysis by INEL and/or LANL.

Thermal-hydraulic computer models are being developed to analyze PTS transients. The H.B. Robinson (Westinghouse) plant is being modeled at INEL with the RELAP-5 code. The Calvert Cliffs 1 (CE) plant is being modeled at LANL with the TRAC code. Both INEL and LANL are developing models for the Oconee 1 (B&W) plant to cross-check the computed results. ORNL is assisting in modeling control and secondary systems. To assess these TRAC and RELAP-5 code models, which include secondary and control systems not previously modeled, the codes will calculate plant behavior during an actual plant transient (e.g., a turbine trip). Also, Brookhaven National Laboratory is independently auditing the models.

For those scenarios requiring multidimensional downcomer calculations, LANL will provide the analyses. In addition, this program is providing simplified analyses for local downcomer temperatures expected to occur under stagnant or near-stagnant conditions. These simplified analyses are based on test data from the 1/5-scale CREARE test facility funded by EPRI.

The results of the RELAP-5, TRAC, and mixing calculations will be best-estimate downcomer temperature and pressure versus time for each of the selected event sequences. ORNL will use the temperature and pressure as input to calculate the conditional probability of vessel failure if the transient occurs. (This conditional probability of vessel failure increases with plant operating life because of the irradiation embrittlement of the vessel.) For each of the many sequences not analyzed with RELAP-5 or TRAC, ORNL will estimate the coolant

temperature and pressure and then estimate the corresponding conditional probability of vessel failure. These conditional probabilities of vessel failure will be multiplied by the expected frequency of occurrence of the transients and summed to estimate the likelihood of vessel failure. Since the conditional probability of vessel failure depends on vessel operating life, the frequency of vessel failure will be presented as a function of operating life. Dominant event sequences will be apparent in these results. Sensitivity analysis will identify important operator and control actions, uncertainties, and effectiveness of corrective actions such as warming ECC water or increasing the toughness of the vessel through annealing.

The status of the project is as follows: Following a preliminary survey of available information in the summer of 1981, the probabilistic analysis of PTS at Oconee was begun in October 1981, and this analysis will be completed with a draft report in mid-1983. In the summer of 1982, the owners of Calvert Cliffs and H. B. Robinson agreed to participate in this study, and these analyses will be completed with draft reports in 1984.

An in-house study will extend the ORNL results to estimate consequences and public risk due to PTS. This consequence analysis will use calculations from the Severe Accident Sequence Analysis research (Section 5.2) to help understand what vessel failure modes would lead to core melt. This limited in-house analysis will also evaluate the likelihood that a missile may be generated as the result of vessel failure and that this missile could cause early failure of the containment. The results will be a generic estimate of (1) the likelihood of core melt given a vessel failure and (2) the likelihood of a large release and the risk to the public given a core melt.

This estimate of expected consequences can then be combined with the plant-specific likelihood of vessel failure to estimate the likelihood of core melt or the risk to the public. Similarly, the results can be used to show the risk-reduction effectiveness of proposed corrective measures. These analyses of these specific plants may be extended by NRR to assess classes of plants. For example, important control features or operator action identified for one plant may be evaluated at other similar plants.

Whereas this research study will estimate the risk-reduction effectiveness of possible corrective measures, other efforts (principally in industry) will assess the feasibility and cost. Thus, this research will support the benefit portion of future cost/benefit studies regarding possible licensing requirements and eventual rulemaking.

The major research products will be:

- Probabilistic analysis of likelihood of vessel failure due to PTS at representative plants designed by B&W (Oconee 1), CE (Calvert Cliffs 1), and Westinghouse (H. B. Robinson 2) (1983-1984).

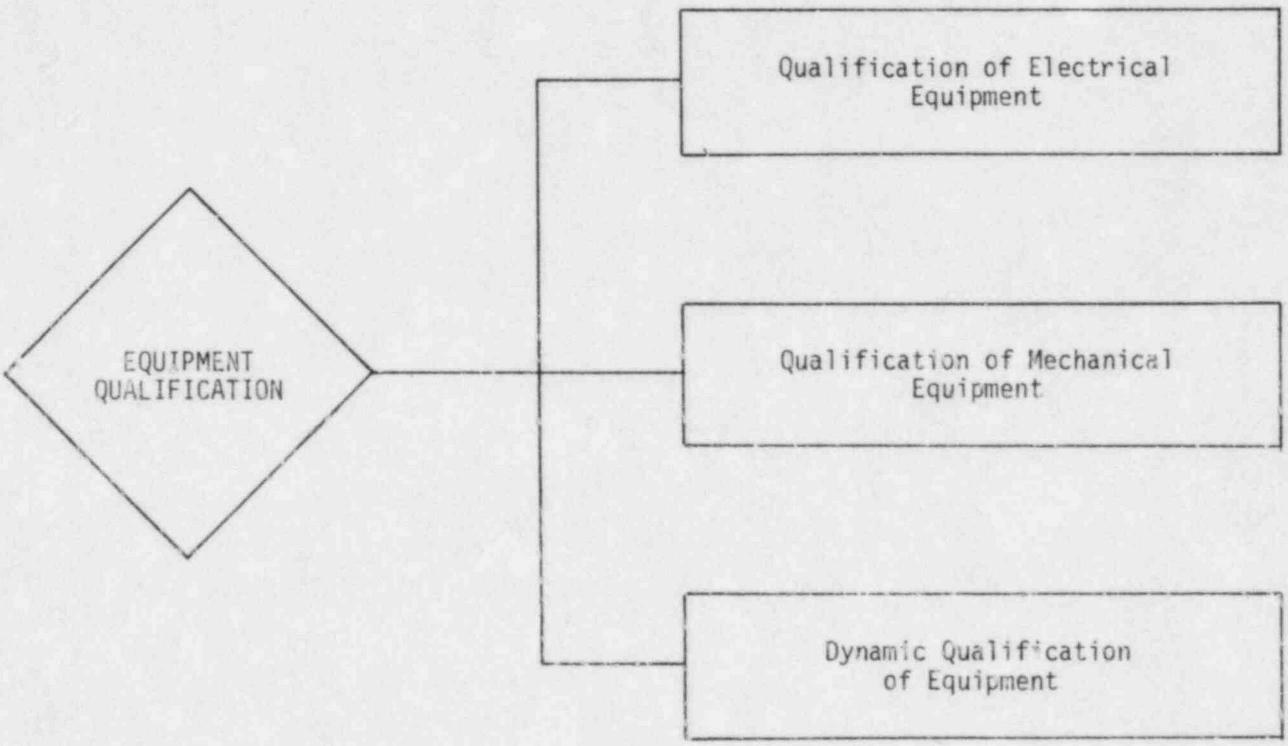
- Estimate of the consequences and risk to the public as a result of PTS (1983-1984).

- Followup calculations to determine effects of new information, i.e., longer-range results from research in other areas such as human factors and safety implications of control systems (1985-1988).

4

Equipment Qualification

4



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
EQUIPMENT QUALIFICATION	$5.4	$5.9	$6.4	$5.5	$4.5

4. 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, and accelerated aging techniques. 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. As priorities and schedules change through management review, some details of the equipment qualification research program may require modification.

4.1 Qualification of Electrical Equipment (Environmental)

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. Qualification methods for LOCA, main steam line break, hydrogen burn, and other design basis accident conditions will be considered. The research will include a limited study of severe accident conditions.

4.1.1 Major Regulatory Needs and Their Justifications

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

   Justification: A rule on the environmental qualification of electrical equipment important to safety for nuclear power plants is being developed in order to provide the nuclear industry with the methods and criteria that the Commission considers acceptable for meeting its regulatory requirements. Detailed procedures defining the steps required to meet this rule are to be issued as regulatory guides. Relevant national standards are evaluated by the NRC and if determined to be suitable are endorsed by regulatory guides, generally with modifications considered acceptable to the NRC. It is essential that the provisions of the regulatory guides have firm technical bases.

   Electrical equipment plays a significant role in safety-related nuclear systems. Thus, the satisfactory functioning of such equipment during and following an accident is essential to the protection of the public. Hence, the rules and regulatory guides on equipment qualification are expected to make a significant contribution to risk reduction by ensuring that electrical equipment is adequately tested.

2. Assessment of methods and procedures for qualification testing to determine whether electrical equipment will function under simulated design basis accident conditions, to be the basis for validating the acceptability of national standards (FY 1988).

Justification: The Institute of Electrical and Electronics Engineers (IEEE) standards define only in general terms the steps required to qualify electrical equipment. It has been found that to implement the standards greater definition is needed. Industry is performing tests to qualify their nuclear equipment to meet the general terms of the standards but performs only limited research on determining their adequacy and validity. Research is being performed by the NRC to provide a basis for assessing the adequacy and safety margins inherent in the testing methods.

3. Criteria and methods for accelerated 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 (FY 1988).
   Justification: For most electrical equipment, the margin to the failure threshold under accident conditions decreases as the equipment approaches the end of its qualified life. Thus, the qualification testing of equipment for survival and functionability during and following an accident normally requires testing the equipment in its worst time-related degraded condition. Many items of nuclear equipment are expected to have a 40-year life. A method for accelerating the aging process is 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 duplicated by the prescribed accelerated aging procedure in national standards and guides. The industry, in order to complete equipment qualification in a short period of time, will frequently use a high radiation dose rate in aging. However, increased aging degradation to cable and gasket polymers by using low dose rates has been observed in NRC aging research. The national standards provide guidance on only the limiting upper allowable dose rate but are otherwise silent. NRC research is further assessing this and other aspects of accelerated aging.

4. 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 (FY 1987).
   Justification: Some material such as polymers (elastomers) used in gaskets, seals, 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.

4.1.2 Research Program Description

The research strategy for this element is to develop basic information on physical processes involved in the qualification testing of electrical equipment and to translate this information into qualification testing methodologies. Qualification research methodology tests will be performed to verify the methodologies and determine if synergistic effects or thermal interactions between parts occur in the tests. The research results will be reflected in revisions to the NRC rules, regulatory guides, and national standards for qualifying equipment. A data base of information on the behavior of critical materials (polymers, integrated circuits) and failure modes of electrical equipment will also be developed to assist the licensing staff in evaluating the safety of nuclear plant equipment. This research will rely on data from the fission product behavior research and will provide input to the seismic and dynamic qualification of equipment research.

The research program will evaluate accident qualification testing methods and will address issues pertinent to simulation such as synergism, dose rate effects, superheated steam, humidity, thermal shock, and oxygen depletion. The program will establish the accident radiation signature for qualification testing, will evaluate the adequacy of radiation simulators, and will establish calculational and testing methodologies. Research to evaluate the aging of electrical equipment will be performed and accelerated aging methodologies developed. Methods for qualifying materials and components will be evaluated by testing typical equipment. In addition, the research will include some work on assessing the qualification of electrical equipment in mild environments (e.g., aging before seismic tests) and also an evaluation of electrical penetration fragility limits during severe accidents.

The research required to develop criteria, guides, and standards for electrical equipment qualification testing will be completed in 1987. Completing the revisions to the standards, guides, and rules is expected to extend into 1989.

The major research products from this element will be:

- o Evaluation of effects of aging, radiation dose rate, synergism, and steam exposure to polymers (1985).
- o Determination of fragility limits of electrical penetrations under severe accident conditions (1984).
- o Establishment of normal and accident radiation signatures for equipment qualification based on current fission product source term research and evaluation of the adequacy of radiation simulators (1985).
- o Assessment of the testing methods and sequences for electrical equipment qualification given by IEEE-323 and -344 and by Regulatory Guides 1.89 and 1.100, and revision of guides as appropriate (1986).
- o Evaluation of aging mechanisms for electrical equipment and establishment of methodologies for accelerated aging (1984).

- o Assessment of methodologies given in the standards for the qualification of electrical cables and penetrations (1984), motors and electrical valves (1985), and postaccident monitoring equipment (1986).

- o Validated accelerated aging and accident simulation methods by examining and testing components removed from nuclear plants (1987).

4.2 Qualification of Mechanical Equipment (Environmental)

This research will provide the technical basis for developing environmental requirements for functional qualification of active 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 4.3. Included is research on extrapolation, characterization of environments, load sequencing, load combinations, and qualification by testing and/or analysis.

4.2.1 Major Regulatory Needs and Their Justifications

1. Determination, based on risk-reduction potential, of the specific mechanical equipment that should be qualified when subjected to environmental loads in new and operating plants and those under construction, to be the basis for amending the regulations (FY 1984).
   Justification: All mechanical equipment does not have the same impact on safety. This research will determine, based on risk-reduction potential, what environmental equipment qualification requirements should be codified.

2. Determination and characterization of those environments affecting the ability of the equipment to perform its safety function, to provide criteria for assessing qualification programs submitted by applicants and licensees (FY 1988).
   Justification: Since mechanical equipment will be subjected to many different environments, it is necessary to determine which environments may affect the safety function of the equipment and the uncertainty of the magnitude or level of the environment to be simulated during qualification testing. The research will also identify areas where the uncertainties in defining the environmental parameters may be beneficially reduced.

3. Evaluation of proposed methods of qualifying equipment for normal operation and postulated accident environments 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 (FY 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 methods must be performed by the NRC.

4. 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 (FY 1987).
Justification: Prudent acceptance criteria for mechanical equipment qualification methods must be developed. These acceptance criteria must account for uncertainties in defining the environmental loads and the qualification methods, yet must provide a measure of reduction in risk over mechanical equipment qualified to other criteria.

4.2.2 Research Program Description

The approach for this research is to first identify thoroughly the component's environments that are anticipated during normal operations and postulated accidents (including design bases events) to determine if all environments are currently being considered. Following this effort, those environments that are significant in establishing that an equipment is qualified 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, 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 like probabilistic risk will mature to the stage that they will be discrete to the subsystem level, possibly to the major component level, and will characterize not only the accident environmental loads but also the normal operational environmental loads.

The industry safety and relief valve testing program was required by the NRC Action Plan developed as a result of the TMI-2 accident. The program is being monitored to verify that the PWR pressurizer safety and relief valves and associated block valves and piping and the BWR safety/relief valves and associated piping will adequately perform for all fluid conditions expected under operating, transient, and accident conditions. It also involves review of data developed from the tests to provide methods for analyzing these valves and associated piping to confirm the adequacy of existing and future installations. The project started in FY 1980 and will be completed in FY 1985.

The major research products will be:

- Identification of range of environmental loads to be evaluated (1984).
- Identification of significant loads or combinations of loads (1985).
- Evaluation of criteria to extrapolate test results from one size component to another (1986).
- Standard review plan to include criteria for environmental qualification by test (1987).
- Codification of acceptable methodology for environmental qualification of mechanical equipment (1988).

4.3 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 active 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, sequencing and combinations of loads, margins, uncertainties, and qualification by testing and analysis.

4.3.1 Major Regulatory Needs and Their Justifications

1. Determination, based on risk-reduction potential, of the specific mechanical and electrical equipment that should be qualified when subjected to dynamic (including seismic) loads in new and operating plants and those under construction, to be the basis for amending the regulations (FY 1984).
   Justification: All mechanical and electrical equipment does not have the same impact on safety. This research will determine, based on risk-reduction potential, what dynamic qualification requirements should be codified.

2. Determination and characterization of those dynamic loads affecting the ability of the equipment to perform its safety function, to be the basis for licensing decisions and for assessing qualification programs submitted by applicants and licensees (FY 1985).
   Justification: Since mechanical and electrical equipment will be subjected to many different dynamic (including seismic) loads during the life of a plant, it is necessary to determine what characteristics of the dynamic loads may affect the safety function of the equipment and the uncertainty of the magnitude or level of the dynamic loads to be simulated during qualification testing. The research will also identify areas where the uncertainties in defining the dynamic parameters may be beneficially reduced.

3. Evaluation of proposed methods of qualifying equipment for normal operating and postulated accident dynamic loads 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 (FY 1986).
   Justification: Evaluation of methods of mechanical and electrical equipment qualification for dynamic (including seismic) 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 methods must be performed by the NRC.

4. 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 (FY 1987).
   Justification: Prudent acceptance criteria for mechanical and electrical equipment qualification methods for dynamic (including seismic) 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 criteria.

4.3.2 Research Program Description

The approach for this research is to first identify thoroughly the component's dynamic and seismic loads that are anticipated during normal operations and postulated accidents (including design bases events) to determine if all vibratory loads are currently being considered. Following this effort, those loads that are significant in establishing that an equipment is qualified will be studied to determine if the assumptions currently used concerning the vibratory loads are correct and to determine if there are any synergistic effects when those loads are combined and when they are combined with the environmental load defined in Section 4.2.

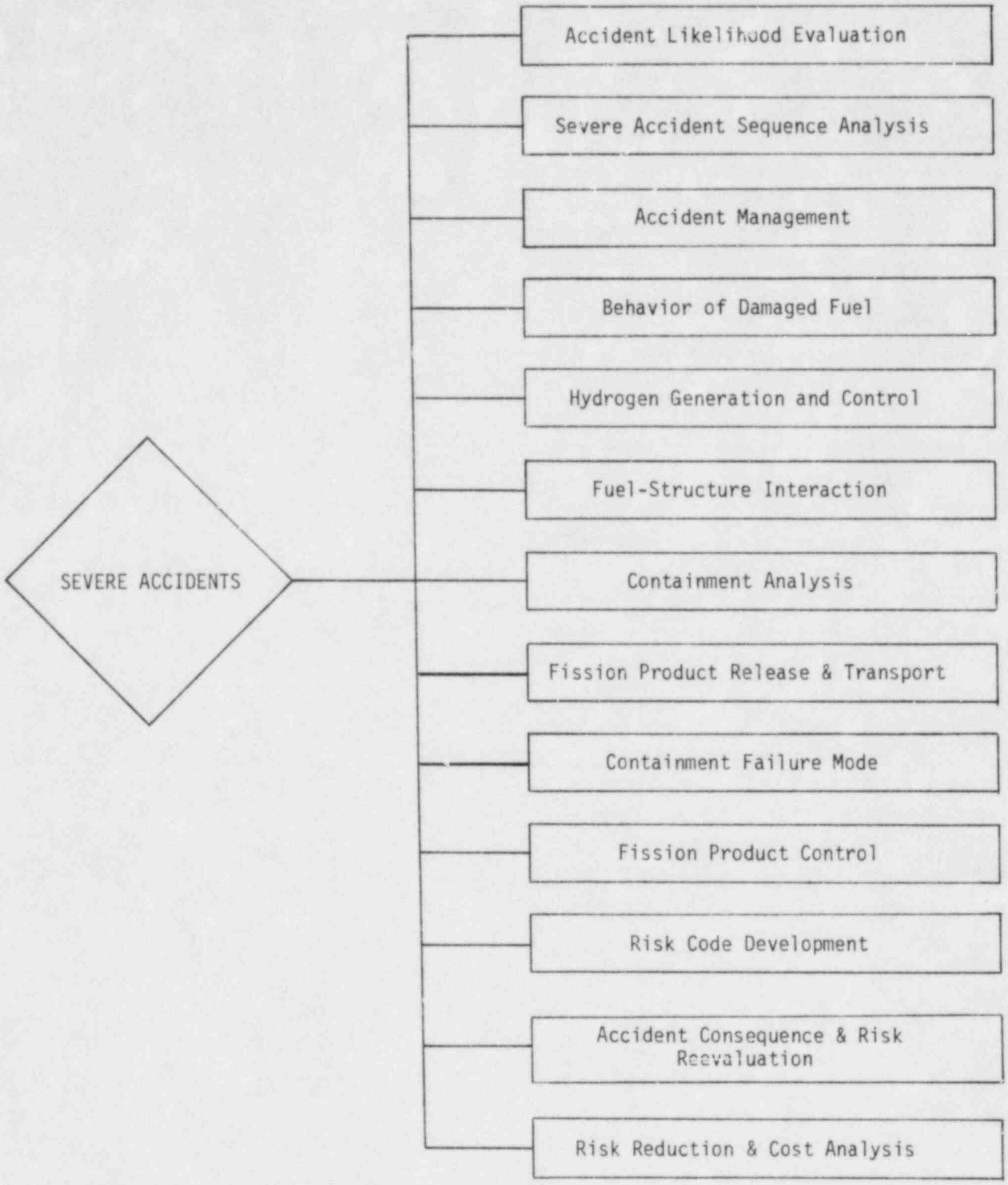
This effort will then be combined with the results of other studies, e.g., probabilistic risk, to determine which components, based on potential risk reduction, should be subjected to codified dynamic qualification requirements. It is anticipated that techniques like probabilistic risk will mature to the stage that they will be discrete to the subsystem level, possibly to the major component level, and will characterize not only the accident dynamic loads but also the normal operational dynamic loads.

The major research products will be:

- o Identification of range of dynamic loads to be evaluated (1984).
- o Identification of significant loads or combinations of loads (1985).
- o Evaluation of criteria to extrapolate test results from one size component to another (1986).
- o Standard review plan to include criteria for dynamic qualification by test (1987).
- o Codification of acceptable methodology for dynamic qualification of mechanical equipment (1988).

5

Severe Accident



<u>FUNDING LEVELS</u>
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
SEVERE ACCIDENTS	\$51.8	\$42.9	\$42.1	\$42.0	\$30.4

5. 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.

5.1 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 studies (Interim Reliability Assessment Program (IREP), Indian Point, Zion, Limerick), 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.

5.1.1 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 (FY 1983).
   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 5.2) and its related experimental program and for the systematic analysis of the risk-reduction effectiveness of backfitting safety systems on operating plants (see Section 5.13).

2. Validation of the likelihood of dominant reactor accident sequences based on recent probabilistic risk assessment (PRA) studies (e.g., IREP) and LER data (FY 1983).
   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 (FY 1983).
   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.

5.1.2 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 5.2) and to support decisions on backfitting additional safety features to existing plants (see Section 5.13).

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), the IREP, 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 5.2 and 5.13). 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, sabotage, 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 sabotage, earthquakes, etc., are being developed to draw upon ongoing research programs such as the Seismic Safety Margins Research Program (see Section 14.5). 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:

- Evaluations of plant operating data, LERs, and vendor information for component and system reliability (1984).

- Identification and review of accident sequences, including accident precursors (1983-1984); reports on likelihood of additional accident sequences and related probabilities (1985).

5.2 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.

5.2.1 Major Regulatory Needs and Their Justifications

1. Analysis of severe accident scenarios for specific types of plant designs, to be used in licensing reviews (FY 1985).
   Justification: A number of postulated high-risk sequences leading to possible severe accidents are being identified by risk assessments of the IREP and the RSSMAP. Detailed analysis of these high-risk sequences is needed to determine appropriate operator actions and any need for special instrumentation. Specifically recommended operational techniques for managing accident recovery and algorithms to be used by the operator to prevent, diagnose, and respond properly to accidents are to be provided as a basis for appropriate regulatory actions.

2. Resolution of licensing and safety concerns expressed by NRR in its review of operator guidelines (FY 1983).
   Justification: NRR has transmitted to RES for analysis by the SASA program some 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.

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 (FY 1984).
   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 reevaluation 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 (FY 1984).
   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 (FY 1984).
   b. Shutdown decay heat removal requirements, A-45 (FY 1983).
   c. Anticipated transients without scram (ATWS), A-9 (FY 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. Studies are now in progress to identify alternative techniques such as feed and bleed for shutdown decay heat removal.

   The resolution of all 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.

5.2.2 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:

- Evaluation of plant abnormal and emergency operating procedures for preventing or mitigating severe accidents in both PWRs and BWRs (1983).
- Assessment of potential detrimental influence of control system failures on severe accident response (1984).
- Assessment of effectiveness of potential procedural and plant design changes to ensure acceptability of consequences of an ATWS (1984).

- Evaluation of fission product (FP) pathways, FP transport rates, and FP inventories in plant structures resulting from severe accidents (1985).

- Assessment of instrumentation requirements to provide unambiguous information to the operator during accidents or transients (1985).

- Development of plant response characteristics to multiple failure sequences identified by risk assessment programs and severe accident research programs (1985).

5.3 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 5.2) and from human factors research (see Chapter 9).

5.3.1 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 (FY 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 (FY 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 (FY 1984).

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 (FY 1984).
   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 factors, reliability, and phenomenological research programs of the NRC. This integration will achieve related benefits from research efforts for improving operational requirements for operator action during severe accidents.

5.3.2 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 and on 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 principal research products will be:

- Identification of operator error rates (1985).
- Establishment of man-machine interaction requirements for accident management (1985).
- Development of operator procedures for recovery from potentially severe accidents (1986).
- Assessment of containment system response to severe accident environments (1986).
- Development of operating procedures for ensuring containment integrity during severe accidents (1986).

5.4 Behavior of Damaged Fuel

This element describes research to determine the general behavior of damaged fuel in the 500°F to 5000°F temperature range, the fission product and 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.

5.4.1 Major Regulatory Needs and Their Justifications

1. Determination of kinetics and quantities of fission product release and attenuation in a correct in-core thermal-chemical environment, for use in regulatory decisions concerning conditions beyond the design basis (FY 1984-1985).
   Justification: Direct measurement of the fission product release from prototypic fuel under well-characterized primary system conditions will allow reliable extrapolation of the proposed Marviken programs 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 Marviken program and the TRAP-MELT code will be able to answer 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 product entering the containment.

2. Determination of the actual hydrogen release from core materials and recommendations for appropriate regulatory response (FY 1984).
Justification: The rate of hydrogen generation during a degraded-cooling accident is dependent on the details of the fuel and coolant behavior in the sequence. Evaluation of the adequacy of mitigation and control features requires knowledge of the time and quantity of hydrogen release.

3. Determination of the general behavior of severely damaged fuel in the 500°F to 5000°F temperature range, for use in establishing severe accident policy (FY 1984).
Justification: Consideration of imposing additional regulatory requirements is dependent on a substantial reduction of uncertainty in the estimated likelihood of core melt following a degraded-cooling accident. To achieve this reduction, more detailed knowledge is needed on how the core behaves under degraded cooling.

4. Determination of the coolability limits and cooling requirements of damaged cores at various stages of degradation, to be used by the NRR staff in reviewing proposed accident recovery procedures (FY 1983-1984).
Justification: 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. Currently there exists a limited data base and some sophisticated analytical models for the coolability of LMFBR (liquid-metal fast-breeder reactor) core debris in sodium. Verification data are needed on the applicability of these models to LWR severe accident conditions. Such verification data are to be furnished by the LWR debris coolability experiments and analysis in the severe fuel damage (SFD) research program.

5.4.2 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, and the coolability of the damaged core by reflooding. Key interfaces with other elements include risk assessment, SASA, 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, conduct of research that NRC would otherwise have to fund, and provision of test hardware. Without these significant outside contributions, the PBF and other NRC-sponsored in-pile experimental work will be ending in 1984. The Federal Republic of Germany (FRG) is conducting a substantial out-of-pile research program on SFD, the results of which will be available to the NRC on an exchange basis. The planned program is a four-part integrated program of in-reactor and laboratory experiments and analysis.

The first part will consist of multi-effect in-pile tests in the PBF at INEL and also possibly in the NRU in Canada to provide scoping data on governing phenomena and on multirod interactive effects. The second part will consist of separate-effects experiments on the governing phenomena, both in the Annular Core Research Reactor (ACRR) and the laboratory, to furnish a data base for model development and assessment and to cover the necessary range of parameters on a cost-effective basis. The third part will consist 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. These two codes will 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 will consist 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 SFD program is the PBF Phase-1 series of five tests that will be performed in FY 1983 and FY 1984.

The major research products will be:

- Report on risk-dominant phenomenological uncertainties in severe LWR accidents (1984).
- SCDAP-Mod 2 with whole core analysis capability (1984).
- Report on initial ACRR debris coolability (degraded core cooling series) experiments (1984).
- Report on analysis of PBF Phase-1 tests (1984).
- Report on Pacific Northwest Laboratory (PNL) high-temperature out-of-pile oxidation experiments (1984).
- Report on applicability of LMFBR debris coolability models to LWR conditions (1984).
- Initial version of MELPROG (1984).
- Initial report on independent examination of selected TMI-2 core samples (1984).

- Report on ACRR debris formation series of experiments (1985).
- Assessment of SCDAP-Mod 2 (1985).
- Report on ACRR quench-debris series of experiments (1986).
- MELPROG-Mod 1 code released (1986).
- Report on analysis of PBF Phase-2 tests (1986).
- Assessment of final version of SCDAP (1986).
- Final report on ACRR LWR degraded-core-cooling experiments and coolability models (1986).

5.5 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.

5.5.1 Major Regulatory Needs and Their Justifications

1. Data from all areas of hydrogen research such as generation, ignition conditions, and mixing are needed to support the proposed rule on interim requirements related to hydrogen control (FY 1983-1985).
   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) (FY 1984).
   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. Hydrogen control system information for the "Pending Operator License Application" (OL Rule) (FY 1984).
   Justification: The rule requires an assessment of hydrogen control systems for various accident scenarios and containment types. The hydrogen safety research program will be providing a technical basis to support and confirm those assessments.

4. Technical data and information on hydrogen generation and control to help formulate the Commission policy on hydrogen regulations not covered by the existing rule (FY 1983-1985).
   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.

5. 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" (FY 1983-1985).
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.

6. Specific data on hydrogen combustion phenomena for ice condenser and Mark III containments (FY 1984).
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.

5.5.2 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.

A supporting program is studying the prevention and mitigation of hydrogen combustion. Mitigation options include oxygen depletion and pre- and post-accident inerting. This program includes studying the effects of CO_2 and water fog evaporation. A deliberate flaring technique in conjunction with high-point vents is 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 are:

- o Analysis of three to five specific plants (1984).

- o Assessment of effects of aerosols on hydrogen control system (1984).
- o Preliminary hydrogen transport code with flame acceleration model (1983).
- o Coupled transport and burn codes (1985).
- o Combustion experiments and autoignition (1983).
- o High-point vent mitigation (1984).
- o Hydrogen mitigation tests.
  - Pre-inerting and oxygen depletion (1983).
  - Improved hydrogen mitigation systems (1984-1986).

5.6 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 safety system injection, and (3) mitigating structures or devices.

5.6.1 Major Regulatory Needs and Their Justification

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

1. Basemat penetration rates for severe accident risk analysis (FY 1984).
2. Heat generation and release for risk analysis of containment performance during severe accidents (FY 1984).
3. Gas and aerosol release for risk analysis of containment performance during severe accidents (FY 1985).
4. Rapid steam generation with potential for containment failure due to missile generation from steam explosions or to nonexplosive steam overpressurization (FY 1988).
   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 four needs listed above. The sources for the containment challenge are not adequately known, and more research is needed to quantitatively evaluate (1, 2, and 3) effects of the interaction between hot fuel debris and concrete basemat materials; (4) effects involving

rapid steam generation; and (1-4) the quantification of parameters used to provide a basis for verifying analytical models used in risk assessment and accident management planning.

5.6.2 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 code. 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 principal products will be:

- CORCON code verification (1983).
- Experimental data for modeling the interaction of hot solidified melts with concrete and long-term cooling characteristics of solidified core melts (1985).
- Predictive, mechanistic models for the explosive and nonexplosive interaction of core melts with water in dropping and reflood contact modes in the Fully Instrumented Test Series (FITS) facility (1985).
- Verification of fuel-melt/concrete and fuel-melt/retention materials interaction models in CORCON to be obtained from large-scale test facilities (1985).

5.7 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, and fuel-structure interactions in the reactor vessel cavity.

5.7.1 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 1984). The code is structured to be readily modified to reflect new data and models as they are developed.
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.

5.7.2 Research Program Description

This research element 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 from which aerosol source term information will also be generated.

The major products will be:

- Integration interface between CORCON code for core-melt/concrete basemat interaction and CONTAIN code for containment performance (1983).

- CONTAIN code verification (1984).

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

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

- CONTAIN code maintenance and minor improvements (1983-1988).

5.8 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. 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.

5.8.1 Major Regulatory Needs and Their Justifications

The Commission has requested that the radioactive source term be reassessed by early 1983 (Policy and Planning Guidance (PPG), NUREG-0885). The source term research provides the necessary data base for the development of severe accident policy in the following areas:

1. Emergency planning and response requirements (FY 1983-1984).
   Justification: Current emergency planning requirements have been influenced by severe accident source term estimates developed for the Reactor Safety Study and subsequent PRAs. Certain of these requirements and the implementation and application of the Environmental Protection Agency (EPA) protective action guides (PAGs) should be reevaluated if there are significant reductions in the predicted source term. For example, reassessment of the need for prompt notification systems to a 10-mile distance may be

warranted. The need for improved source term information with regard to reevaluating EPA requirements is discussed in NUREG-0771 and in an NRC evaluation of emergency planning zone practices at 5- and 10-mile distances from reactors. This research provides technical information to be used for reevaluating the studies upon which current emergency preparedness requirements are based (see item 2 of Section 9.5.1).

2. PRA consequence calculational methods (FY 1983-1984).
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. The revised source term data base will be particularly useful in current evaluations of the effectiveness and need for additional plant features to reduce the risk of severe accidents. Evaluations using the older and pessimistic source terms could significantly overstate the effectiveness of added features in reducing risk.

3. Equipment qualification (FY 1983-1984).
Justification: The NRC is in the process of developing a final rule on equipment qualification. Definition of the radiation environment (dose levels) for equipment qualification is one aspect of this rule. Source terms currently in use for equipment qualification are based on past design basis accident analysis assumptions (e.g., Regulatory Guides 1.3 and 1.4), on observations made during the TMI-2 accident, and on calculated release fractions reported in the Reactor Safety Study. These source terms may require complete revision. EPRI is conducting research that will contribute significantly to NRC's decisions in this area.

4. Siting policy (FY 1983-1984).
Justification: Nuclear power plant siting requirements have the potential of alleviating most consequences of the impact upon the public of accidental releases of radionuclides. Significant modification to the radiological source released from the plant during severe accidents will 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 (PPG, NUREG-0885). Major results will be available in the FY 1983-1984 period. Final results are due in FY 1986.

5.8.2 Research Program Description

The strategy of the fission product release and transport program is to develop models and 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 5.10). 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 also being developed.

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 will be completed in 1983.

The major research products will be:

- o Revised best-estimate severe accident source terms for use by the Commission to reassess the source term currently used (1983).
- o Data report for benchmarking fission product release models from irradiated LWR rods up to 2400 K (1983).
- o Data report on core-melt aerosol release models from large (10 kg) out-of-pile fuel bundle experiments (1983).
- o Mechanistic FASTGRASS model for fission product release from fuel (1983).
- o Data report on behavior of prototypic aerosols in a condensing steam atmosphere for testing models in aerosol code (1983).
- o Documented computer code model for fission product release during core-melt interaction with concrete (1984).
- o Improved version of TRAP-MELT fission product and aerosol transport code (1983-1984).
- o Data report from experiments with fission product chemistry in the gas or aqueous phases to confirm or improve fission product behavior model (1984).
- o Large-scale (Marviken) fission product and aerosol transport test package to be used to assess TRAP-MELT code (1985).
- o Data report for benchmarking models for release of fission products from fuel that has experienced 3100 K from Phase II of PBF in-pile experiments (1985).

5.9 Containment Failure Mode

This element treats three possible failure modes: faulty valve operation, 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. Research is not needed in this area. 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.

5.9.1 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 (FY 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 (FY 1987).
   Justification: The implementation of a safety goal will require, as part of the PRA calculations, computational models describing the performance of containments. In particular, the implementation of a "containment performance criteria," i.e., conditional probability that a containment will function given an accident, will require an ability to relate variability in leakage behavior to variability in structural parameters.

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 (FY 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 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.

5.9.2 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.

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 isolation 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 5.5, 5.6, and 5.7), as well as with the risk code development element (see Chapter 8). There will also be interaction with other U.S. programs. Contributions to this program from EPRI are anticipated in the form of analytical predictions of capacity to be compared against test results. This program will be coordinated with the containment capability program being contemplated by the Department of Energy and the Industry Degraded Core (IDCOR) program on containment overpressure response.

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 1983. Reinforced concrete models will be tested in 1984.

The planning of dynamic, unsymmetric pressure tests 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:

- Estimated steel containment capacities compared with experimental static pressure results (1984).
- Experiments on valve performance and electrical penetrations (1984).
- Initial test series on models of major penetrations (1985).
- Predicted capacities for prestressed and reinforced concrete containments compared with experiments under static pressure (1985).
- Predictions of steel containment capacity under dynamic pressure loads compared with experimental results (1986).
- Predictions of capacity for prestressed and reinforced concrete containments under dynamic pressure loads compared with experimental results (1987).
- Initial tests of containment models under simulated seismic loading (1988).

5.10 Fission Product Control

The fission product control program is developed to evaluate the effectiveness of 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 severe accidents.

5.10.1 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 rulemaking [FY 1984]; (2) revising Regulatory Guides 1.3, 1.4, 1.7, 1.52 [FY 1985]; and (3) revising the standard review plans for ESF systems [FY 1985]. 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.

5.10.2 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 U.S. 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:

- Report on evaluation of performance and scrubbing efficiency for suppression pools as part of BWR generic design review (1984).

- Code development and verification for evaluation of PWR ice-condenser effectiveness (1984).

- 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).

- Code development and verification for performance and effectiveness of PWR/BWR containment sprays under severe accident conditions (1984-1985).

- o Report and input for code development on hydrogen burning and its effect on aerosol concentrations and selected ESF-system performance (1985).
- o Evaluation of existing PWR/BWR ESF-system design (1985).
- o Code development and verification for evaluation of generic design of ESF systems for standardized nuclear facilities (1986-1987).
- o ESF reliability testing code development and verification (1987-1988).
- o Code development and verification for aging aspects of selected generic ESF systems (1987-1988).

5.11 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 benchmark and value/impact assessments discussed in Sections 5.12 and 5.13.

5.11.1 Major Regulatory Needs and Their Justification

1. Numerous additional supporting calculations, sensitivity studies, etc., for the MARCH and MATADOR codes (FY 1983).

2. On a longer-term schedule than for Need 1, development for regulatory use of a risk code more readily understandable and amenable to modification, MELCOR to fulfill this need (FY 1985).
   Justification for Above Needs: 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 5.12 and 5.13 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 Sections 5.12 and 5.13 analyses. Correction of known deficiencies in these codes is necessary prior to their use in this context. The work of this section will provide the required code modifications.

5.11.2 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 nature of the deficiencies is such that some are amenable to short-term upgrading while others require longer time or supporting phenomenological research. For this reason, the code development is to be performed in two parallel paths, one relatively short term, the other long term. These two paths are described in more detail below.

1. MARCH-2/MATADOR Development Program

   The MARCH-2/MATADOR development program has as its objective the short-term modification of the present versions of these codes. Because of the need for improved codes on the short-time schedule of this research (see Sections 5.11 and 5.12, in particular) and other regulatory matters (e.g., plant operating license reviews), this code development will improve particular aspects of the codes but will not attempt to alter their basic structure. Thus modifications will be made, for example, to improve upon certain too simplistic models, to correct identified errors, to replace specific basic data with more recent data, etc.

2. MELCOR Development Program

   The MELCOR development program is intended to produce the longer-term replacement computer code 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 products from this element will be:

- MARCH 2 and MATADOR codes documented for users (1983).

- MELCOR code documented for users (1985).

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*The MARCH code analyzes in-plant 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.

5.12 Accident Consequence and Risk Reevaluation

In this section, the risk codes discussed in Section 5.11 are being applied in concert with products of other sections (e.g., 5.1, 5.9) to produce up-to-date assessments of the consequences and risk of severe accidents in LWRs.

5.12.1 Major Regulatory Need and Its Justification

Up-to-date analyses of the predicted consequences and risk from severe accidents in LWRs (FY 1983-1987).

Justification: As noted in Section 5.11, 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 5.13) needed for this rulemaking can be performed. Thus, this work directly supports resolution of the severe accident rulemaking.

5.12.2 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 5.11), they will be put to use to reanalyze the consequences of accident sequences determined to be important in previous risk studies. Further, as these consequence analyses are completed, they are to be combined with the sequence likelihood results, 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 (1983-1987).

5.13 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.

5.13.1 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.

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.

5.13.2 Research Program Description

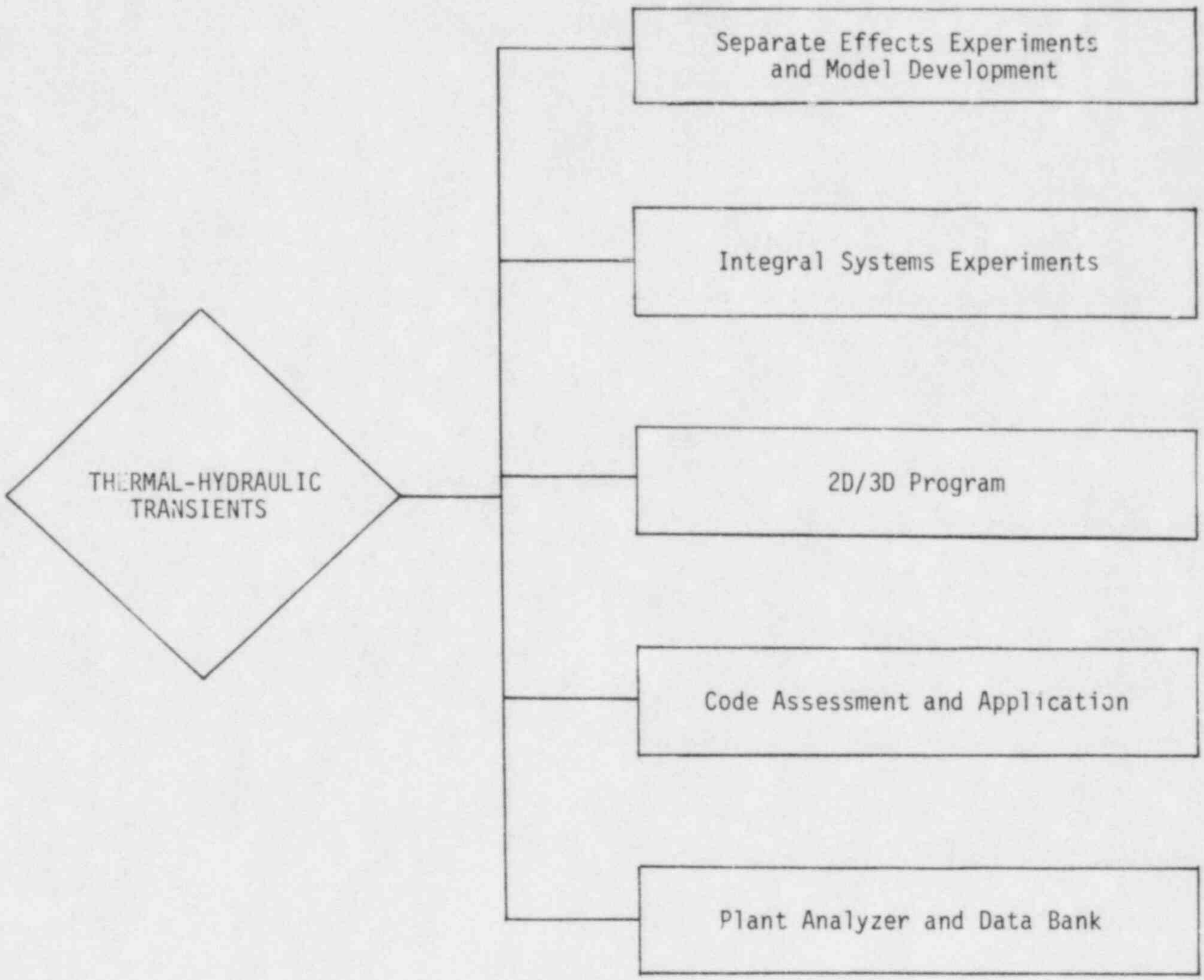
In this element, analyses of the risk-reduction potential and costs associated with a spectrum of possible plant modifications for severe accident prevention and mitigation are to be performed. Included in these possible modifications are, for example, filtered-vent containment systems, alternative shutdown heat removal systems, and stronger containments. The objective of such analyses is to identify those modifications that appear to present the most 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 products will be:

- Identification of narrowed set of most promising features for severe accident prevention and mitigation (1983).
- Identification of the risk-reduction value and cost associated with specified set of features (and combinations of features) (1983).
- Revised risk reduction and cost estimates using updated information (1984-1985).

Thermal-Hydraulic Transients

6



<u>FUNDING LEVELS</u>
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
THERMAL-HYDRAULIC TRANSIENTS	$37.7	$31.4	$25.6	$23.5	$20.9

6. 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 2 years. The data obtained through the separate effects research programs enabled code developers to produce models that represent physical phenomena. The assembled models were 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 represented 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, 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 is planned to 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, the 2D/3D program, code assessment and application, and plant analyzer and data bank.

6.1 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 during degraded core cooling and other plant transients. 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 and water hammer, 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.

6.1.1 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 Chapters 2 and 3) (FY 1984). Justification: 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 no experimental data on the mixing of cold ECC water with hot reactor vessel water in the downcomer. This information is needed to determine possible thermal stresses on the reactor vessel.

2. A basis for understanding and validating transient and LOCA fluid flow and heat transfer models used in systems analysis codes (FY 1983-1988). Justification: To assess operator guidelines proposed for management of a range of LOCA sequences, system analysis codes, frequently using special fluid flow models, are used. The regulatory assessment 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 the following revisions to Appendix K to 10 CFR Part 50 in addition to those revisions discussed in Section 6.3.
   - o Revised post-CHF heat transfer correlation (FY 1983).
   - o Second general revision (FY 1984-1986).
   - o Analysis model for steam cooling (FY 1985).
   - o Fuel element blockage criteria (FY 1984).
   - o Less than 1"/sec cooling criteria (FY 1985).
   - o Evaluation of long-term decay heat (FY 1985).

   Justification: 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. The Policy and Planning Guidance document has established the priority for this work.

4. Revision and assessment of heat transfer package in the RELAP and TRAC codes for use in the evaluation of operator guidelines (FY 1988). Justification: The heat transferred from fuel rods to water is calculated in current RELAP and TRAC code versions according to correlations such as the "re-wet criterion" and critical heat flux (CHF) that are purposely chosen to be conservative for use in 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 for the balance of the plant (BOP) in advanced codes for use in evaluating anticipated transients and ATWS effects (FY 1985).

Justification: Current steam generator models in TRAC and RELAP are not based on directly relevant experimental data. As a result, unknown error is present in analyses involving the use of the BOP models in the TRAC and RELAP codes.

6.1.2 Research Program Description

Ongoing experiments include grid spacer effects, nonequilibrium post-CHF heat transfer, PWR loop and steam generator flow oscillations, 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 and to result 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 planned to provide (1) separate effects testing of important transients and (2) scoping tests for input to the proposed B&W integral test 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 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 valuable support in evaluating the PTS issue discussed in Need 1 of Section 6.1.1.

The purpose of the steam/water flow program is to obtain data and subsequently models that describe various interactions of steam and water in reactor piping. Northwestern University will be obtaining data on condensation heat transfer, interfacial friction, flooding, and water hammer at various angles of pipe inclination to simulate LWR piping.

The major research products will be:

- PWR loop oscillation phenomena report (1983).
- Simplified B&W loop simulation for understanding separate effects phenomena and for B&W experimental facility formal design, to be used in evaluating proposed test facilities to resolve issues (1983).
- Inverted annular flow model (1983).
- Condensation heat transfer model (1983).
- Interfacial friction model (1983).
- Stability map for water hammer at different angles of pipe inclination (1983).
- Application method for use in TRAC or RELAP-5 systems codes (1983).
- Thermal fluid mixing models (1984).
- Downcomer fluid mixing model (1984).
- Pressure vessel fluid heat transfer model (1984).
- Qualified grid heat transfer enhancement models (1984).
- Bundle post-CHF data analysis report (1984).
- Validated steam generator data from tests (1984).
- Post-CHF heat transfer model (1985).

6.2 Integral Systems Experiments

This element includes experimental simulations of integral thermal-hydraulic systems of PWR and BWR reactors. The U.S. facilities involved are the Semiscale 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 will be operated under an international consortium.) Foreign agreements and understandings provide for data exchanges with comparable facilities in the FRG (PKL), Commission of the European Community (CEC) (LOBI), and Japan (ROSA IV). Transients simulated include the full-break-size spectrum of LOCAs, loss of feedwater, steam line break, steam generator tube rupture, and various safety and control system failures. This element is closely associated with TRAC and RELAP-5 code development and assessment in that the TRAC

and RELAP-5 codes are tested against experiments conducted in these facilities in order to improve and validate the codes (see Section 6.4).

6.2.1 Major Regulatory Needs and Their Justifications

1. Experimental data from an integral test facility on PWR station blackout with total loss of ECC for review of operator guidelines (FY 1983).

2. Experimental steam generator tube rupture data for code validation and review of operator guidelines (FY 1984).

3. Experimental steam line break data for evaluation of operator guidelines, plant recovery technique assessment, and validation of code capability (FY 1984).

4. Integral loop test data to assess operator guidelines for management of high-occurrence and risk-dominating events (FY 1985).

5. Primary coolant system feed and bleed data for use in evaluating procedures for plants with low-head high-pressure injection (HPI) systems under upset conditions (FY 1984).

6. Data on small hot-leg breaks with and without primary pump operation for evaluating feed and bleed and pumps-on/pumps-off procedures (FY 1984).

7. Fuel clad swelling and rupture, blockage, and flow diversion experimental data to validate the blockage models in Appendix K to Part 50 (FY 1986-1987 assuming planned fuel test programs continue).

8. Validated data on BWR small-break behavior to be used in assessment of proposed operator guidelines (FY 1983).

9. Experimental and analytical data on BWR ATWS response for regulatory review of ATWS (FY 1983).

10. Data on BWR natural circulation for NRR use in analysis of ATWS and loss of forced flow (FY 1984).

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

12. Data for the evaluation of a station blackout in BWRs to evaluate severe accident capability and operator guidelines (FY 1985).

Items 1 through 4 are expected to be met in Semiscale, items 5 through 7 in LOFT, and items 8 through 12 in the FIST facility.
Justification: The following justifications apply to all 12 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.

6.2.2 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 and later will concentrate on accident and transient sequences that have been identified as having a high probability of occurrence and that, when equipment failures or gross operator errors occur, can lead to degraded cooling or core melt. The sequences identified in the IREP and NREP studies will provide the bases for test definitions. The tests will be designed to evaluate both the identified sequences and the prescribed operational procedures. Computer code predictions will be assessed on the basis of the test data. This will provide further understanding of the abilities of large systems codes (such as RELAP-5 and TRAC) to predict transient conditions as well as LOCA conditions.

A Test Advisory Group, composed of representatives from NRC, B&W, B&W owners, and EPRI, has studied the data needs for licensing issues unique to B&W reactors. A report evaluating the data needs and the existing integral systems experimental facilities and containing recommendations for additional experimental work is nearing completion. This report will form the basis for

a joint NRC-industry research effort to resolve the licensing issues related to the B&W reactor designs. It is expected that related tests and analyses will be completed in FY 1987.

Management and control of the LOFT program has been transferred to the DOE and the OECD LOFT project, and funding is now being shared by approximately 15 countries and agencies. The NRC will be a member of the new project and has indicated its preference for tests of the type indicated in items 5 through 7 of Section 6.2.1. 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, will be completed in late FY 1983. At that time, simulations of a wide variety of BWR coolant system transients will be 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 FY 1984 and a similar assessment of code capability will be made in FY 1985. If a need is shown for further data to support the final versions of the codes, continued testing will be conducted in FY 1986.

The major research products will be:

- Semiscale MOD-2B data used to assess steam generator tube rupture and loss-of-power models (1985).
- Evaluation of licensing issues based on tests at B&W experimental facility (1987).
- Postirradiation examination of LOFT ballooned fuel bundle to be used to assess licensee models (1987).
- Assessment of advanced computer codes using LOFT thermal-hydraulic test data (1984).
- Assessment of safety and licensing issues using LOFT results (1985).
- Evaluation of BWR transients using FIST Phase I test results (1985).
- Assessment of BWR calculational techniques using preliminary analytical models (1987).
- Evaluation of BWR transients using FIST Phase II data (1988).

6.3 2D/3D Program

The 2D/3D program is a joint research program with the FRG Ministry of Research and Technology (BMFT) and the Japanese Atomic Energy Research Institute (JAERI) to study the thermal-hydraulic behavior of the emergency core coolant during the refill and reflood phases of a large-break LOCA in a PWR. Since the TMI accident, a study of some aspects of a small-break LOCA has also been included in the work scope.

6.3.1 Major Regulatory Needs and Their Justifications

Provide experimentally verified information on the following five areas needed for the evaluation and possible revision of Appendix K to Part 50 (FY 1986).

1. Steam binding effect.
   Justification: The steam binding effect represents a major and uncertain source of conservatism in evaluating the large-break LOCA and should be addressed before current ECCS designs can be fully assessed in the best-estimate sense. This will be evaluated in large experimental facilities that will provide the three-dimensional effects on entrainment and deentrainment.

2. ECC bypass.
   Justification: The effectiveness of current ECCS designs is largely dependent on the extent of ECC bypass. Large-scale test data are needed to evaluate the small separate effects and integral systems experiments and the analytical models so as to give a more precise estimate of the existing conservatism.

3. Determination of the core blockage effect.
   Justification: The effect of blocking the fuel rod flow channels due to ballooned rods has been addressed in the FLECHT-SEASET facility. The tests at the Cylindrical Core Test Facility and at the Slab Core Test Facility will verify the multidimensional flow around large-scale blockages.

4. Three-dimensional effect of reflood process.
   Justification: Preferential flow paths resulting from varying flow resistances in the core and nonuniform distribution of water pooling in the upper plenum may cause a redistribution of flows that may result in local cooling or local overheating. Results obtained in the FLECHT-SEASET facility will be tested in the large-scale 2D/3D facilities.

5. Countercurrent flow limitation (CCFL) in hot legs.
   Justification: Small-scale data indicate that the natural circulation of coolant around the primary loop changes to a reflux condensing type of circulation as the void percentage increases in U-tube steam generators. Under this mode, liquid and vapor flow in opposite directions in hot legs and thereby create a possibility of CCFL, thus limiting the capability to remove heat. The CCFL in full-scale hot legs will be obtained in the Upper Plenum Test Facility (UPTF), and these data will be used to assess analytical models in the advanced computer codes.

6.3.2 Research Program Description

The 2D/3D program has been formulated to address evaluation of the safety margins in five areas of the Appendix K rule and to provide these dimensional data for code improvements and code assessment. The five areas in which conservatisms in Appendix K are being evaluated are the ECC bypass phenomena, requirement to subtract water injected during the ECC bypass period, three-dimensional effect on the reflood process, effect of core blockage during reflood, and upper plenum deentrainment during reflood. Another major issue being addressed by the 2D/3D program is providing three-dimensional data for code improvement and code assessment for the two/three-dimensional computer codes for TRAC and TRAC/COBRA. The 2D/3D program provides the only full-scale three-dimensional separate effects data during the refill and reflood portion of a large-break LOCA.

The 2D/3D program was initiated jointly with the FRG Ministry for Research (BMFT) and with JAERI to minimize the cost in resolving the issues raised above. To reduce the cost further, the 2D/3D program was limited to the PWR LOCAs, refill and reflood phases of a large-break LOCA, and the natural circulation phase of a small-break LOCA because these areas were believed to represent the greatest uncertainty and hence greatest risk in the LOCA-initiated event.

FRG completed in 1981 the design of a full-scale UPTF with simulated core, steam generators, and loops. The construction has just started and will be completed by the end of 1985. This facility will be used for, among others, upper plenum entrainment and deentrainment tests, combined hot-leg and cold-leg injection tests, and the downcomer bypass tests.

The NRC has provided the advanced two-phase flow instrumentation to the present Japanese and German test facilities and will do so for the future test facilities. The NRC has also been providing analytical support by performing facility design calculations and pre-test and post-test analyses of experiments using the advanced multidimensional two-fluid transient analysis code (TRAC).

The major research products will be:

- o Determination of flow blockage effect in fuel bundles during reflood phase to help assess coolability of blocked core as specified in Appendix K to 10 CFR Part 50 (1984).
- o Assessment of computer code TRAC as compared with Appendix K model for calculating two-dimensional flow effects in core (1985).
- o Determination of steam binding effect during reflood phase as it relates to requirements in Appendix K (1986).
- o Determination of extent of ECC bypass during refill phase as required by Appendix K (1986).
- o Determination of three-dimensional flow effects in reactor vessel during reflood phase to help assess extent of hot spots in vessel (1988).

- Assessment of computer code TRAC as compared to Appendix K evaluation model for calculating three-dimensional flow effects in upper plenum (1988).

- Determination of entrainment and deentrainment of liquid in upper plenum and hot legs to help assess steam binding effect and top-quenching of reactor (1988).

- Determination of CCFL in hot legs during small-break LOCA (1988).

- Examination of earlier Appendix K evaluation models based on small-scale data to see if they are still applicable to large-scale data (1988).

6.4 Code Assessment and Application

This element includes the application of computer codes to the analysis of transients in full-scale LWRs and the assessment of these analytic capabilities against experimental data.

6.4.1 Major Regulatory Need and Its Justification

Assessed and validated systems analysis computer codes that are needed to assist NRR in auditing safety evaluations submitted by applicants (FY 1985). Justification: In order to make licensing decisions that ensure safety and have the proper cost/benefit relationship, best-estimate analyses of transient behavior in LWRs are required. Independently developed and assessed systems analysis computer software are used as available by NRR for audit purposes. These analyses help licensing personnel uncover unexpected consequences and thus better assess proposed accident management schemes.

6.4.2 Research Program Description

The strategy for the research in this element is 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 developers for more extensive testing. Collection and analysis of nuclear data and plant design and operating histories are ongoing efforts that support this element. Approximately one-third of the codes are developed jointly with outside groups (EPRI, foreign governments).

This research program will develop code packages for analysis of 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.

Essentially all independent assessments will be completed in early FY 1987. This coincides with the completion of all major code development in 1985. After 1987, there will be a small but important program for independent assessment of PWR and BWR analyzers. Code application is planned to increase over

the next several years as resources are shifted from code development and independent assessment.

The major research products will be:

- Final versions of the advanced multidimensional two-fluid transient analysis code (TRAC) (1985).
- Assessment of TRAC-PF1/MOD-1, RELAP-5/MOD-2, and TRAC-BD-1/MOD-1 (1984).
- Incorporation of multirod fuel code for analyzing severe core damage accidents resulting in $>$ 2500°F (SCDAP) into simplified version of a systems code for analysis of LWR behavior under degraded core conditions (1985).
- Integration of system codes to benchmark and audit risk analysis methods (1986).
- Assessment of COBRA/TRAC (1985).
- Assessment of PWR system plant analyzer (1987).
- Assessment of BWR system plant analyzer (1988).
- Use of plant analyzer and plant data bank to analyze transients in full-scale LWRs to resolve licensing and safety issues (1986-1988).

6.5 Plant Analyzer and Data Bank (Includes Code Improvement and Maintenance)

This element includes improvement and maintenance of the computer codes discussed in Section 6.4, as well as the development of user-oriented capabilities to use these codes in the form of an automated plant analyzer with plant-specific output displays. It also includes the acquisition and manipulation of plant data needed to develop input specifications for plant-specific analyses.

6.5.1 Major Regulatory Needs and Their Justifications

1. User-convenient system analysis codes for use in evaluating transients and accidents with the ability to input operational manipulations at midpoints during long transients (FY 1985).
   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 should be fast and should allow user interaction and 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.

2. Geometric and operating data for all licensed plants to allow plant-specific calculations to be performed (FY 1983-1988).

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.

6.5.2 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.

While the major effort is devoted to use of the codes as programmed for large central computers, a small effort is being undertaken to investigate the potential benefits of reprogramming existing codes for small special-purpose, high-speed computers that would serve as dedicated machines. A demonstration of this technique will be conducted early in 1983, and a decision concerning the future direction of this work will follow. A plan for the major part of the plant analyzer work will be ready about the same time as the above-mentioned demonstration.

The specifications of the plant analyzer reflect our current experience with the RELAP-5 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 to follow a transient. An interactive feature, currently being specified, would significantly reduce computational effort and speed the acquisition of results. There are, however, meany technical features to be specified and evaluated before the scope of such a feature is fixed. 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.

The major research products will be:

- Demonstration of typical PWR plant analyzer, with incorporated plant data bank (1984).
- Data for selected nuclear power plants added to data bank (1984-1988).

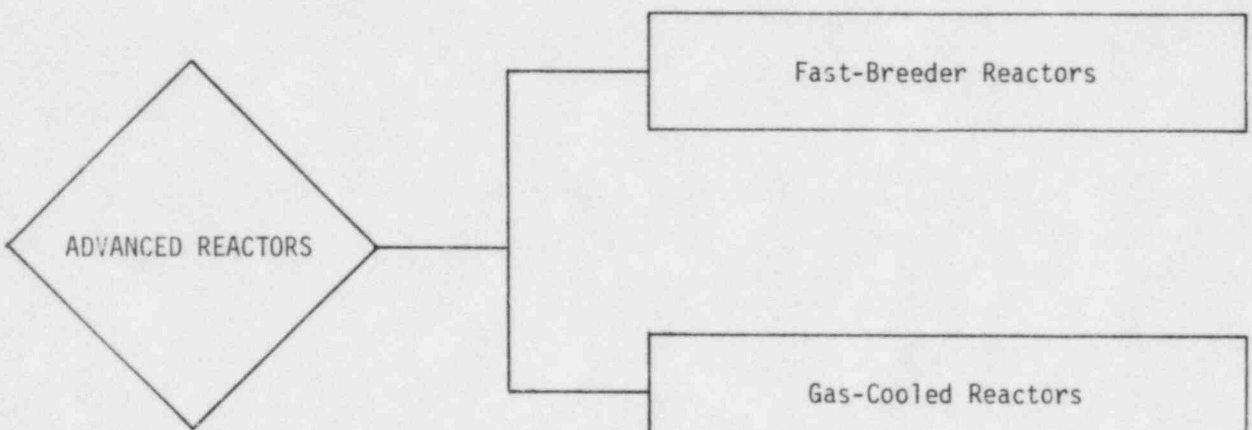
- First version of PWR system plant analyzer completed and demonstrated for use by NRC personnel (1986).

- First version of BWR system plant analyzer completed and demonstrated for use by NRC personnel; maintenance continued for all NRC codes being used for analysis; PWR plant analyzer and associated LWR-plant data bank improved and maintained (1986).

- PWR and BWR plant analyzers improved and maintained, along with necessary maintenance of LWR-plant data bank; continued maintenance for all NRC codes being used in analysis (1987-1988).

Advanced Reactors

7



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
ADVANCED REACTORS	$9.9	$9.0	$9.5	$9.5	$9.5

7. ADVANCED REACTORS

This 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 will provide the NRC with data and analytical methods to make licensing decisions on liquid-metal-cooled fast-breeder reactors (LMFBRs). In the past few years, this program has been directed at generic development and verification of safety analysis codes and of experimental data needed for evaluating LMFBR safety. As a result, a number of computer codes (COMMIX, SIMMER, SSC, CONTAIN, etc.) 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. For the next few years, the scope of research will be directed at resolving Clinch River Breeder Reactor (CRBR) licensing issues. The longer-range objective of the program in FY 1984-1988 will be to complete research needed for the CRBR operating license review and, on a schedule consistent with the Department of Energy (DOE) LMFBR development program, develop a basis for licensing a large commercial LMFBR that is likely to be significantly different in design from the CRBR.

7.1.1 Major Regulatory Needs and Their Justifications

In support of CRBR licensing:

1. A number of safety issues that need to be assessed by the NRC in order to license CRBR, including:

    a. Decay heat removal by natural convection cooling (FY 1985).

    b. Assessment of the energetics of a core disruptive accident (CDA) and coolability of core debris in sodium (FY 1985).

    c. Consequences of failures in the heat transport system such as broken pipes and frozen pumps on thermal margins (FY 1985).

    d. Consequences of complete loss of offsite and onsite power (FY 1984).

    e. Interaction of liquid sodium with concrete and the effects of sodium fires and hydrogen combustion on the containment integrity (FY 1986).

    f. Consequences of core meltdown and challenge to the containment structure by molten fuel (FY 1986).

    g. Consequences of failure of plant protection system (FY 1986).

h. Consequences of malfunction of plant control system (FY 1986).

i. Definition of radiological source term (FY 1984).

Justification: The licensing review of the CRBR construction permit is under way. RES has been directed to place its major LMFBR program emphasis on support to the NRC-CRBR Program Office in resolving the technical and regulatory issues needed to license CRBR.

2. In addition to the immediate regulatory needs for licensing CRBR, NRR and the Advisory Committee on Reactor Safeguards (ACRS) have recommended research for developing LMFBR-specific generic licensing criteria for future LMFBRs. General design criteria, siting criteria, and regulatory standards and guides are needed. Future large commercial LMFBRs will require additional safety research. If EPRI and DOE coordination on LMFBR design progresses as expected, the NRC program will be coordinated with EPRI and DOE efforts to minimize duplicative efforts.
Justification: LMFBRs are substantially different from LWRs. The development of generic design criteria and regulatory standards will be required as a long-range objective of the program. This development of generic design criteria and regulatory standards will be closely coordinated with the DOE LMFBR base technology program to ensure relevance and applicability for commercialization, to avoid duplication, and to focus support for NRC responsibilities for timely licensing of future plants in accordance with national policy.

7.1.2 Research Program Description

The strategy for the research in this element is to apply the computer codes and experimental data base that have been developed in the past to the specific needs of the CRBR licensing review. Code improvement and experimental programs will be directed at resolving issues that arise in both the construction permit and operating license stages of this licensing review. In addition to the immediate research for supporting the licensing review of CRBR, research to develop generic licensing criteria for future LMFBRs will be undertaken.

The family of major computer codes developed under this program include SIMMER for analysis of core disruptive accidents, the Super Systems Code (SSC) for analysis of the response of the entire plant to accidents and transients (natural circulation, pipe break, etc.), COMMIX to study thermal hydraulics of components, and CONTAIN for analysis of containment conditions during accidents.

The analytical code development program will include improvements in SIMMER to address the heterogeneous core in CRBR, application of SSC and COMMIX to accident analysis and heat removal problems, and application of CONTAIN to defining the loads of the CRBR filter-vent system. The experimental work on sodium-concrete and molten-core/concrete interactions should be completed, and models of these phenomena should be available for use in CONTAIN in FY 1984. Experimental work in the Annular Core Research Reactor (ACRR) to provide an accident energetic data base and validation for SIMMER analysis may be required through licensing for operation of CRBR (through FY 1987).

The major research products will be:

- o Analysis of CRBR thermal hydraulics and design basis accidents (1985).
- o COMMIX and SSC validated against Phoenix and SNR 300 (1984).
- o CONTAIN applied to CRBR (through 1985).
- o Fuel aerosol simulant test (FAST) completed (1984).
- o Source term modeling completed (1985).
- o Analysis of CRBR energetics with SIMMER for operating license (1987).
- o Data/models on transition-phase fuel removal processes (1984).
- o 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).
- o Data/models on LMFBR debris coolability limits on bottom-cooled structures (1985).
- o Data/models on clad and fuel relocation and blockage formation during CDA initiation phase (1986).
- o Data/models for assessing long-term ex-vessel debris coolability in sodium (1987).
- o Assessment of CDA termination by fuel-removal processes (1988).

7.2 Gas-Cooled Reactors

This element includes 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 high-temperature gas-cooled reactors (HTGRs), certain appropriate codes and regulatory guide modifications, 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 with current budgets is relatively small. It reflects the thinking that NRC research supports the

regulatory program; NRC research does not support the development of the technology or the license application. The research 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, within the constraints of time and budget, 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 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 siting 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 prestressed concrete reactor vessel liner 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 will be:

- o Resetting of priorities for NRC safety research needs and long-range plans (1983).
- o New HTGR edition of the standard format and content of safety analysis reports for nuclear power plants (1984).
- o Plans for detailed risk analysis of HTGR (1984).
- o Graphite failure criteria and failure mechanism model (1984).
- o Completion of in-core testing of convective flow mixing and natural convection phenomena for code verification (1985).
- o Completion of thermal barrier and liner cooling system requirements (1985).
- o Advanced analysis code for HTGR fission product plateout and liftoff to supersede SUVIUS code (1985).
- o Adaptation of CHAP and ORECA codes to predict severe accident radiological releases (1986).
- o 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).
- o Development of ASME Code revisions and code cases for evaluation of PCRV structure, penetration closures, and liner cooling (1985-1987).
- o Advanced computer code for analysis of HTGR containment system response during depressurization events (1987).

It will be necessary to augment the plans described above if an actual licensing application is received. At that time, 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. Consideration will also be given throughout the period to advanced safety and licensing needs that may be developed as a result of industrial advancements such as those for application of the HTGR to high-temperature process heat or as a result of further development of the currently contemplated smaller, modular HTGR design with its planned walk-away safety features.

Risk Analysis

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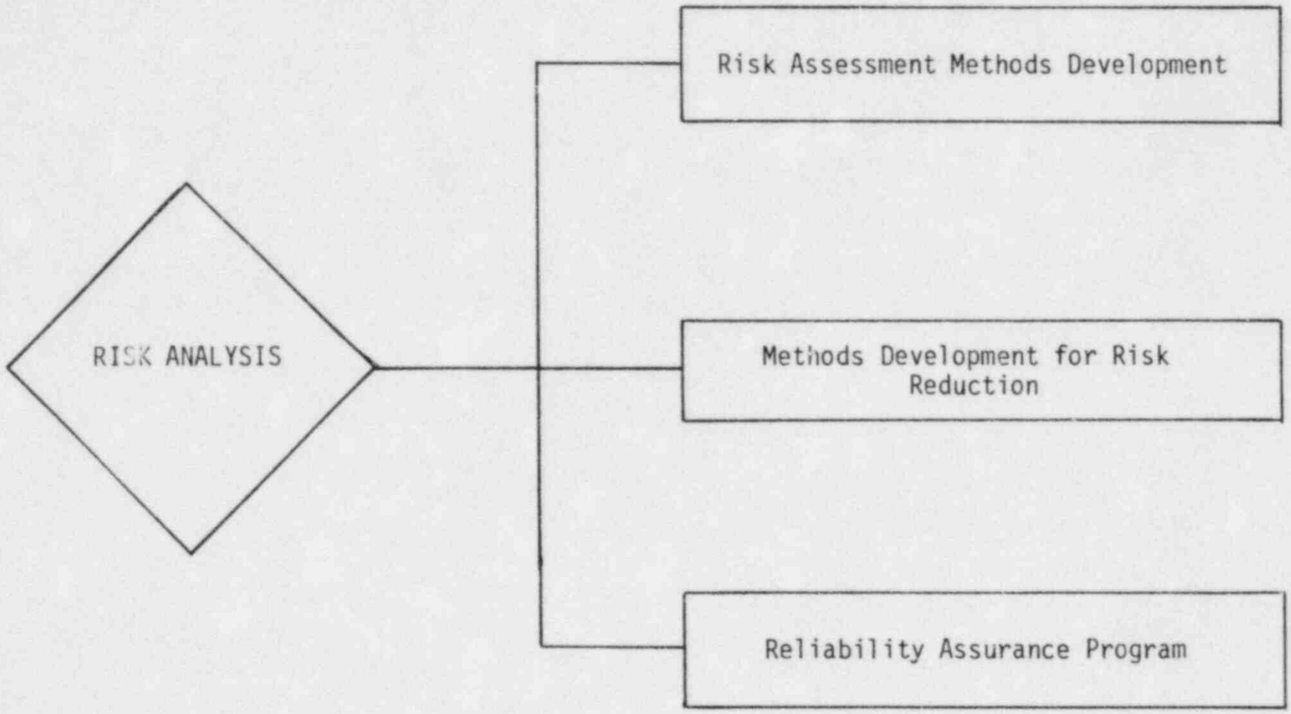

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FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
RISK ANALYSIS	$9.4	$12.0	$17.9	$15.7	$10.1

8. RISK ANALYSIS

This chapter describes the research being carried out to support the application of probabilistic risk assessment (PRA) methods to the regulation of nuclear power reactors. This work includes the development of models, methods, documented procedures, and other analyses required to support Commission decisions on a broad range of critical issues relating to power reactor safety.

This program is divided into three elements corresponding to the principal topical areas within this program, i.e., risk assessment methods development, methods development for risk reduction, and reliability assurance.

8.1 Risk Assessment Methods Development

Research in this element is directed toward developing and documenting methods for quantifying the probabilities and consequences of severe reactor accidents and toward reducing the uncertainties in such estimates. (A more detailed plan for efforts described in this section is scheduled for delivery to NRR by the end of March 1983.)

8.1.1 Major Regulatory Needs and Their Justification

1. Accident Likelihood

   a. Improved data base for estimating component and system failure rates, to support implementation of the Safety Goal and other licensing evaluations of significant safety issues (FY 1983 and beyond as additional data become available).

   b. Methods for the systematic identification and evaluation of principal reactor accident sequences, to support decisions regarding the severe accident rule (see Chapter 5, "Severe Accidents") and the scope of the Systematic Evaluation Program (SEP) (FY 1983-1984).

   c. Techniques for incorporating the contribution of common-cause failures, including fire, and systems interactions into PRA methods, to support the National Reliability Evaluation Program (NREP) (FY 1983-1985).

   d. Methods for quantifying the effects of severe natural phenomena (e.g., seismic activity and external floods) and of human factors on assessments of reactor risk with the human factor efforts being coordinated with those described in Section 9.4, to support NREP (FY 1983-1985).

2. Accident Consequences

   a. Improved PRA models capable of predicting in-plant fission product transport and deposition, using latest experimental data on accident

phenomenology, to support siting policy development and emergency planning requirements (FY 1983-1985).

b. More sophisticated PRA models for determining the health effects of severe reactor accidents using latest data on the biological effects of ionizing radiation, dispersion in the environment, etc., to support siting policy development and emergency planning requirements (FY 1983).

c. Improved methods for estimating the financial and social impacts of severe reactor accidents, to provide data for cost-benefit analysis and indemnification requirements (FY 1983).

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 draft Safety Goal explicitly requires the quantitative consideration of risk. Similarly, policies presently being considered by the Commission (SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation") and Phase III of SEP and the NREP would require that future Commission decisions regarding the imposition of additional safety requirements on operating plants be justified on the basis on their potential for risk reduction. 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).

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, without large 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 14.2.

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 before PRA can be routinely used in NRC decisionmaking.

8.1.2 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 plans for the implementation of the Safety Goal and the SEP. 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 these methods through the performance of a series of limited PRAs. 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 5) and the SEP and longer-range activities required to support application of the Safety Goal to standard plants and new construction permit (CP) applications.

The major research products will be:

- o Component and system reliability data base developed from evaluations of plant operating data, licensee event reports (LERs), and vendor information (1984 with annual revision to reflect most recent data and operating experience).

- o Identification and review of accident sequences, including accident precursors (1983-1984); reports on likelihood of additional accident sequences and related probabilities (1985).

- o Procedures for incorporating results of Seismic Safety Margin Research Program (see Section 14.5) into PRA methodology (1985).

- o Human factors research results (see Chapter 9) to be incorporated into PRA Procedures Guide (1986).

- o Revision of accident consequence codes reflecting most recent experimental data and more realistic physico-chemical phenomena.

    - First version--MARCH-2, MATADOR (1983).
    - Second version--MELCOR-1 (1985).
    - Third version--MELCOR-2 (1986).

8.2 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.

8.2.1 Regulatory Needs and Their Justification

1. Methods to identify, on a generic basis, the potential for risk reduction of alternative design concepts for all classes of power reactors, to support severe accident rulemaking (FY 1983, 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 implement the ALARA provisions of the proposed Commission Safety Goal (FY 1983, 1985-1986).

3. Documentation of uniform procedures for carrying out value/impact analyses for use by both NRC and the industry (FY 1985).

4. Methods for balancing occupational exposures and public risk (FY 1983).

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 to the severe accident rulemaking, NRC must decide whether to require augmented safety features on those plants being reviewed under Phase III of the SEP and how the ALARA provisions of the Safety Goal are to be applied in future licensing actions.

A prerequisite for such decisions is the availability of reliable methods that will permit the staff to evaluate and compare the risk-reduction potential and costs of practical options for preventing and mitigating the effects of severe accidents. Methods and procedures must be developed and documented to ensure that these options can be comparatively evaluated and that such evaluations are made on a consistent basis.

8.2.2 Research Program Description

In the near term, this program will establish methods for the analysis of the risk-reduction effectiveness of generic safety system modifications to support decisions on severe accident rulemaking (see Section 5.13). In the longer term, the program will apply these evaluation methods to the review of a few standard plant designs in order to establish the feasibility of applying ALARA 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:

- Generic assessment of costs and risk-reduction potential of alternative safety features applicable to specific classes of LWRs (1983, 1985-1986).
- Evaluation of feasibility of using PRA to improve reliability of existing plant systems (1985-1986).
- Establishment of standardized procedures for calculating value/impact of proposed rules, guides, etc. (1985).

8.3 Reliability Assurance Program

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.

8.3.1 Major Regulatory Needs and Their Justification

1. Definition of reliability assurance program requirements for nuclear power plants and the formulation of associated regulations and technical specifications (FY 1985).

2. Methods for assessing the risk significance of plant technical specifications as a frame of reference for licensing decisions on limiting conditions of operation, requests for exemptions, or enforcement actions (FY 1984).

3. Methods for the rapid assessment of the safety significance of LER data (FY 1984).

4. Development of specifications for operator actions to be taken in response to events determined to be plant-specific dominant accident sequence chains (FY 1984).

5. Assistance in providing risk perspectives to inspection and enforcement modules for CP, preoperation, and operating license (OL) inspections, which take into account current analyses of accident sequences, plant operating data, and accident likelihood (FY 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 influence on an overall plant risk assessment.

   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.

8.3.2 Research Program Description

This program develops methods to provide assurance that an acceptable level of safety is maintained over the life of the power plant. The major research products of this element will be:

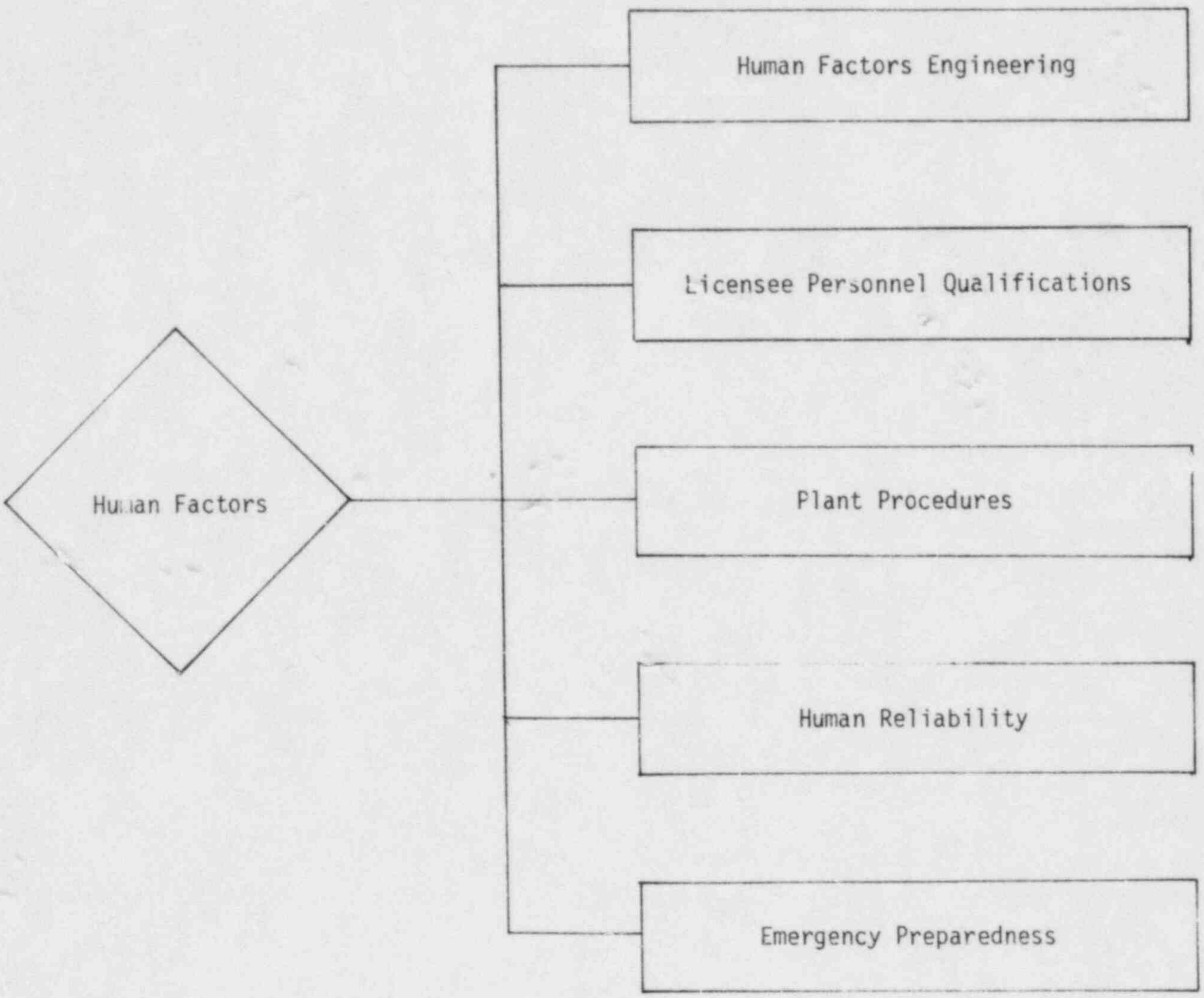
- Recommendations for establishing cost-effective reliability assurance programs at nuclear power plants based on PRA of plant vulnerabilities, the PRA to be a living document for use in all aspects of plant operation, maintenance, and training.

  - Demonstration program (1985).
  - Full-scale application (1987).

- Draft recommendations for operator training and procedures keyed to risk-dominant accident conditions (1985); final recommendations (1986).
- Documented methods for including risk perspectives in licensing reviews of technical specification changes, limiting conditions for operation, exemptions, or enforcement actions (1985).
- Based on risk assessment insights and updated analysis of dominant accident sequences, procedures to focus inspection and enforcement activities on risk-important requirements (continuing program starting in 1985).

Human Factors

9



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
HUMAN FACTORS	$6.2	$7.3	$8.2	$8.3	$8.3

9. HUMAN FACTORS

This program will provide the technical basis to support current and anticipated regulatory needs in the application of human factors to nuclear facilities in support of the NRC Integrated Human Factors Program Plan and will provide the technical basis to support current and anticipated regulatory needs for emergency preparedness at nuclear facilities. The research includes work on control room design and evaluation criteria, personnel qualifications and staffing, plant procedures, human reliability, and emergency preparedness.

9.1 Human Factors Engineering

The research provides the technical basis 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.

9.1.1 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 SASA tasks (see Section 5.2), 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 (FY 1983-1987).
   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. Specifically, 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; "Human Factors Acceptance Criteria for the Safety Parameter Display System," NUREG-0835; 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 backfitting or placing new designs in existing control rooms (FY 1984-1988).
Justification: The design characteristics of computer-based information and control 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 semi-automatic operations, and a mismatch between machine and human functions and capabilities. This work will support near-term regulatory needs for evaluation of backfitting computer-based data systems such as SPDS 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 (FY 1983-1987).
Justification: This research was requested by the Advisory Committee on Reactor Safeguards 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 5.2) and the Seismic Safety Margin Research Program (SSMRP) (see Section 14.5), and External Events (see Chapter 11). 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.

9.1.2 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 products will be:

- o 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 (1984).

- o 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).

- o Data and information base to help develop functional requirements and evaluation criteria for alarm filtering systems, disturbance analysis systems, computerized procedures manuals, artificial intelligence systems, and other computerized systems in the man-machine system (1988).

- o Guidelines for future human engineering standards and criteria for control room and display, control, and communication systems (1987).

- o 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 decisionmaking, and training to be available (1988).

9.2 Licensee Personnel Qualifications

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

9.2.1 Major Regulatory Needs and Their Justifications

1. Determination of the appropriate education, training, examination, and licensing requirements for control room personnel and categorization of decisions made by these personnel, the knowledge they must have to make these decisions, and a means of assessing how well an individual meets these criteria; to be the basis for amendments to a regulation and revisions to a regulatory guide (FY 1984-1986).
Justification: This work is needed to provide the technical basis for regulatory decisions concerning upgrading the qualifications of licensed operators to reduce the potential for human error and its contribution to

risk. These decisions will be reflected in revisions to 10 CFR Part 55 and Regulatory Guide 1.8.

2. Determination of the appropriate use of, capabilities of, and requirements for nuclear power plant simulators in training programs and as examination tools, to serve as the basis for regulatory actions, including amendments to 10 CFR Part 55 and revisions to Regulatory Guides 1.8 and 1.149 (FY 1984-1986).
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 regulatory decisions with regard to simulator capabilities, requiring the use of simulators in training programs, using simulators as examination tools, and determining acceptable alternatives to simulator training.

3. Determination of the appropriate qualifications, training, examination, and licensing requirements for nuclear power plant support personnel, to reduce the potential for human error and its contribution to risk and to be the basis for regulatory actions, including revision to a regulatory guide (FY 1984, 1986).
Justification: The impact of nuclear power plant support personnel on safety varies as a function of both the position and the plant design. This research is needed to provide the technical basis necessary to support regulatory decisions regarding the adequacy of current requirements for the qualifications, training, examination, and licensing of support personnel for existing and planned nuclear power plant designs and will be used to develop regulatory positions in Regulatory Guide 1.8.

4. Determination of the appropriate qualifications, training, examination, and licensing requirements for personnel at fuel cycle facilities, to be the basis for a regulatory guide (FY 1986-1988).
Justification: Historically, qualification, training, examination, and licensing requirements 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.

5. A systematic method that could be used by NRC to evaluate whether additional operator training is an acceptable means of compensating for the operational problems and associated risk that result from a particular design or construction error, to be the basis for appropriate regulatory actions, e.g., a regulatory guide (FY 1988).
Justification: This research is needed to provide the technical basis for criteria to be used by the regulatory staff to determine whether additional training is an acceptable means of compensating for a particular design or construction error or whether hardware changes should be required.

9.2.2 Research Program Description

The strategy is to develop the technical basis for changes to the regulations (e.g. 10 CFR Parts 55 and 72) and the regulatory guides (e.g., 1.8, 1.149, HF 608-4) 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 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:

- Program plan for application of systems approach to training (SAT) to nuclear power plant training (1983).
- Using SAT, criteria selection of malfunctions that should be modeled in nuclear power plant training simulators (1984).
- Empirical data on nuclear power plant operator performance from training simulator experiments (1985).
- Assessment of current qualification and training practices for operations and support personnel at fuel cycle facilities with respect to practices of other industries (1986).
- Training assessment methodology based on SAT for unlicensed operators and support personnel at nuclear power plants (1987).
- Training assessment methodology based on SAT for operators and support personnel at fuel cycle facilities (1988).

9.3 Plant Procedures

This element provides the research that is needed to develop the technical basis for the methods and criteria used by NRC to assess and upgrade, where needed, plant operating procedures necessary for safe operation of nuclear power plants and fuel cycle facilities. Information from PRA (see Chapter 8) and SASA (see Section 5.2) 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, operating procedures, maintenance procedures, and testing procedures. This element does not include administrative procedures for management of these facilities.

9.3.1 Major Regulatory Needs and Their Justifications

1. A technical basis for regulatory decisions regarding the adequacy and effectiveness of nuclear power plant emergency (e.g., single-failure and multiple-failure accident sequences) and normal operating, maintenance,

and testing procedures and for assessing operator performance and training by using upgraded procedures, to develop regulations and regulatory guides (FY 1984-1986).
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, 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. The long-term research will use the technology developed in the near future to address other reactor types (e.g., advanced LWR, LMFBR, advanced gas reactor (AGR)).

2. Evaluation of alternative techniques and formats for presenting procedures, including computer-based CRT systems and other proven concepts, to provide the basis for regulatory positions, criteria, and guidelines for upgrading procedures (FY 1985-1988).
Justification: This research complements Section 9.1, "Human Factors Engineering," and supports resolution of Section I.C.9 of the TMI Action Plan (NUREG-0660). It will provide the technical basis required to support regulatory decisions on assessing current and anticipated procedure presentation practices (e.g., computer-based CRT display) 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 determining the adequacy and effectiveness of operating, maintenance, and testing procedures for fuel cycle facilities, to be used in developing regulatory positions, criteria, guidelines, and regulations for assessing and upgrading such procedures (FY 1986-1988).
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 decisions and standards regarding operating, maintenance, and testing procedures for fuel cycle facilities, including fuel fabrication, storage, transportation, reprocessing, and waste management. The technology developed for nuclear power plant operations will be used where appropriate for this research.

9.3.2 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 nuclear facility operating, maintenance, and testing procedures. 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:

- Methodologies for evaluating generic emergency operating procedure guidelines and operating, maintenance, and testing procedures for PWRs and BWRs (1984).

- o Adaptation of computer-based analysis technique for assessment of nuclear power plant procedures presentation based on qualifications and abilities of the plant personnel (1984-1985).

- o Application of methodologies to provide data for assessing plant procedures and optimizing regulatory requirements (1985-1986).

- o Adaptation of computer-based analysis technique for assessment of plant technical documentation (manuals) (1986-1987).

- o Criteria for application and assessment of alternative techniques and formats for presenting procedures (1985).

- o Assessment of impact of computer diagnostics and automation on procedures and regulatory requirements (1986-1987).

- o Methodologies for evaluation of operating, maintenance, and testing procedures for advanced reactor (LMFBR, HTGR, AGR) and fuel cycle facilities (1986-1988).

9.4 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.

9.4.1 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 (FY 1983-1986).
   Justification: Human error in the operation and maintenance of NPP 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 (FY 1983-1986).
   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 as regulatory design requirements for advanced complex man-machine safety systems (FY 1985-1986). Justification: Human error (omission, commission, extraneous acts, sequential, time) is of critical importance in the design of complex man-machine systems, especially those crucial to safety. A long-range goal (FY 1985 and beyond) of the human reliability program is the transformation of human error technical base data and other addenda into design guidelines for complex man-machine systems that provide improved plant and public safety.

9.4.2 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; and (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. This research will be coordinated with INPO, EPRI, utilities, user groups, other Government agencies, and foreign countries.

The major products of this research will be:

- Handbook supporting human reliability analyses of NPP operation and maintenance safety-related events (1983).
- Workbook (procedures manual) for carrying out human reliability analyses of NPP man-machine safety-related events (1983).
- Prototype human reliability data bank (1984).
- Human error probability data from operating plants, industry training simulators, and expert judgment adequate to support selected human reliability analyses (1984).
- Human reliability data bank implementation plan (1985).
- Computer-based maintenance model for developing human error data to support selected human reliability analyses (1985).
- Human reliability data bank combining varied human error data acquisition media and automated storage and retrieval techniques (1986).

- o Probabilistic risk assessment specialist aids (e.g., handbook, workbook, human reliability models) for human reliability analyses of varied NPP safety-related events (1986).

- o Human error criteria for assessing effectiveness of man-machine safety systems (1987).

- o Man-machine design criteria for advanced NPP safety systems (1988).

9.5 Emergency Preparedness

The research and standards program in emergency preparedness provides the technical basis and standards 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.

9.5.1 Major Regulatory Needs and Their Justifications

1. A technical basis for emergency preparedness regulations for fuel cycle and material licensees, to be used in developing upgraded regulations and regulatory guides (FY 1983-1986).
   Justification: This research is responsive to Commission direction and will provide the technical basis for making regulatory decisions relevant to upgrading emergency preparedness for fuel cycle and material facilities.

2. Upgraded emergency preparedness regulations and regulatory guides for LWR plants and research reactors based on research on severe accident management and source terms, as well as staff experience, petitions for rulemaking, and public comment, in order to provide assurance that appropriate protective actions can and will be taken to mitigate the consequences to the public of an accident. This effort will include a reevaluation of the studies upon which current emergency preparedness requirements are based (FY 1983-1985).
   Justification: This effort is responsive to Commission direction, research on severe accident management and source terms, as well as public petitions and staff experience, and will provide a technical basis for regulatory decisions relevant to upgrading emergency preparedness for research reactors and nuclear power reactors.

3. Identification and verification of 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 (FY 1984-1985).
   Justification: Appendix E to 10 CFR Part 50 requires that the licensees/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.

4. Determination of protective action effectiveness under various sets of representative site conditions, accident sequences, and plant containment types and determination of the impact on protective action

effectiveness based on the ability to predict accident progress (e.g., containment failure time) (FY 1984-1986).
Justification: Appendix E to 10 CFR Part 50 requires that EALs be established for which the licensee will recommend the appropriate protective action to offsite officials. This research will provide a basis to develop additional 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.

5. 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 (FY 1983-1985).
Justification: Work will be continued to develop adequate methods 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.

6. Identification and determination of the relationship of the factors that influence interpretation of containment monitor readings (e.g., break size, monitor type, monitor location, core damage, reactor type) relative to assessing the course of an accident (FY 1984-1987).
Justification: As a result of the TMI Action Plan, the licensees have installed high-range containment monitors. These monitors provide key indications of core conditions, safety system status, and source term composition. The licensees are to establish the relationship between the monitor reading, core condition, and source term composition. The NRC needs an independent technical basis on which to evaluate the adequacy of these relationships for use by the NRC Operations Center in performing its functions.

7. Determination of how much site meteorological data must be available at the time of and during an accident in order to adequately characterize the plume emergency planning zone (EPZ) needed for effective implementation of protective actions (FY 1984-1988).
Justification: The current regulatory position stated in Supplement 1 to NUREG-0737 is that the licensee must be able to sufficiently characterize the plume EPZ to allow for effective implementation of protective measures. An improved technical basis is needed to augment the current heavy reliance on professional judgment for determining whether the licensee's ability to characterize site meteorology is adequate.

8. 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 (FY 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.

9. Criteria for evaluating the adequacy of the calibration methods and standards for instrumentation to be used during accident conditions (FY 1985-1987).
   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.

10. 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 (FY 1985-1987).
    Justification: A great variety of portable instruments are 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.

9.5.2 Research Program Description

The strategy is to build on existing emergency preparedness experience with research that provides a 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) as well as with State and local government authorities will continue.

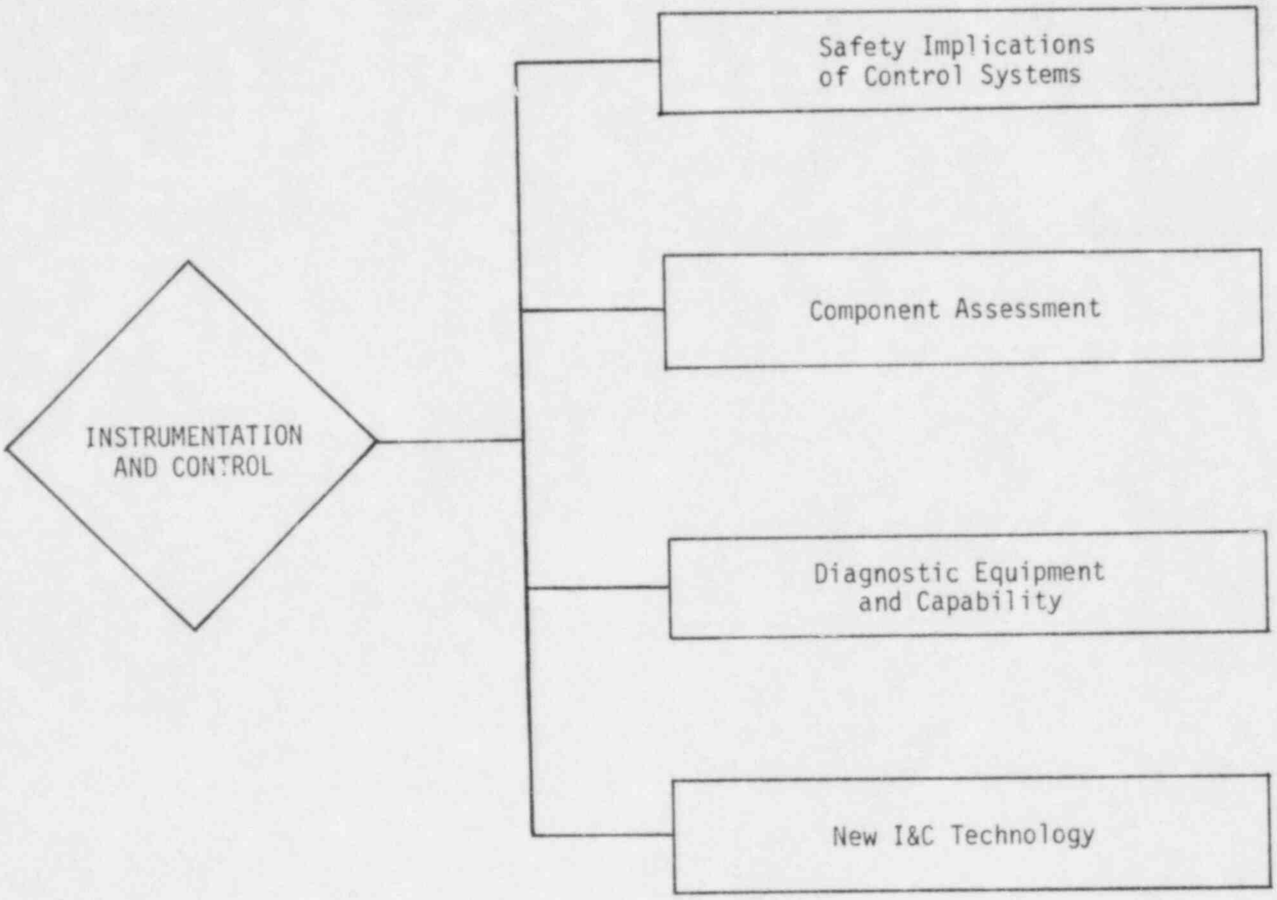
In August 1980, the upgraded emergency preparedness regulations for nuclear reactors were issued by the Commission. These regulations required the installation by February 1982 of prompt public notification systems. Confirmatory research is now being conducted, in conjunction with FEMA, that will enable inspectors to determine adequate compliance with these regulations. Additionally, research is continuing to assess the human factors involved with activating the prompt public notification systems and to review the interfaces for incident response among Federal, State, and local governmental authorities and licensees. Findings of this research may result in modification to the existing regulations. Research is also under way to develop an improved technical basis for emergency preparedness requirements for fuel cycle and material licensees. Research in FY 1984 through FY 1988 will provide information to form a technical basis for formulating inspection procedures and to support the assessment of the adequacy of emergency preparedness.

The major research products for this element will be:

- o Evaluation of instrumentation for measuring radioiodine in the field (1985).
- o Technical basis for emergency preparedness requirements for fuel cycle and material licensees (1983).
- o Evaluation of accident plume determination within EPZ (1985-1986).
- o Evaluation of protective action decisionmaking (1984-1985).
- o Evaluation of emergency action level identification (1984-1985).
- o Evaluation of relationship of containment monitor indication to accident conditions (1984-1985).
- o Evaluation of adequacy of radiological instrumentation under accident conditions (1985-1988).
- o Evaluation of adequacy of calibration methods and standards for radiological instrumentation (1985-1988).

Instrumentation and Control

10



10

FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
INSTRUMENTATION AND CONTROL	$6.2	$6.4	$6.9	$6.3	$6.4

10. INSTRUMENTATION AND CONTROL

This program will involve research to improve and confirm the availability of methods of reactor and associated process systems protection, control, and instrumentation to minimize the probability of abnormal operation or accidents and to mitigate the consequences of an accident if one should occur. The major elements of the program include safety implications of control systems, component assessment, diagnostic equipment and capability, and new instrumentation and control (I&C) technology.

10.1 Safety Implications of Control Systems

This element provides research needed to evaluate malfunctions of plant control and related instrumentation and electrical systems to determine the impact of these malfunctions on plant operational safety and equipment important to safety, particularly where such events could lead to unanticipated transients or accidents. In addition, there is a need to evaluate automatic analog and programmable digital computer-based control and protection equipment and computer-based information systems for use by operators to determine their suitability for nuclear power plants.

10.1.1 Major Regulatory Needs and Their Justifications

1. Information needed for the resolution of USI A-47, "Safety Implications of Control Systems," and USI A-49, "Pressurized Thermal Shock" (FY 1984).
   Justification: The resolution of these issues has a high priority because of potential risk-reduction significance and use in regulatory decision-making for both operating plants and those under licensing review. Research is necessary to provide information and the technical basis for the needed regulatory judgments.

2. Evaluation of the risk to public health and safety presented by the possible failure or malfunction of systems intended for nonsafety purposes, to be the basis for a regulatory guide (FY 1985).
   Justification: Systems provided for nonsafety purposes (e.g., turbine controls, reactor power level controls) could possibly initiate events that should also be considered as protection system design basis events if severe transients can be initiated by failures within these nonsafety systems or if failures of more than one nonsafety system can be caused by some common-cause or common-mode failure.

3. Determination of the need to establish an intermediate classification between the systems most important to safety and those systems not important to safety, to evaluate the need for a regulatory guide (FY 1986).
   Justification: All I&C equipment designated important to safety is not equally important to actual safety. Some equipment that may have some importance to safety is not required to meet any particular requirements for equipment important to safety, although it probably should. The same degree of reliability or assurance of operability in a severe environment would not be needed for all these I&C systems.

4. Evaluation of the safety of programmable digital computers with associated isolation devices and other advanced concepts in safety control, alarm, and information systems, to be the basis for regulatory guides (FY 1986).
   Justification: Digital computers are likely to be used more extensively in plant safety control and display systems in the future. Such systems will have safety implications that were not considered in current regulations and that could potentially hinder the operations of the safety system. In order to be able to properly evaluate proposed designs using programmable digital computers, or their advanced concepts, NRC must independently assess various design options and specify acceptance criteria.

10.1.2 Research Program Description

The strategy for the research is to provide major inputs to the resolution of USI A-47. Control systems of various plant designs will be evaluated to determine their effectiveness in preventing transients and accidents as well as their potential for causing a transient or making worse an event in progress. To help ensure the validity of these evaluations, this work will be performed in close cooperation with each utility whose plant is being evaluated.

The control system evaluations will include consideration of random failures and common-cause, common-mode, and cascade failures and their effects on plant dynamics. Electrical system failures and their interactions with plant controls will be evaluated. Transients with potential to cause safety limits to be exceeded will be identified. These events will be further evaluated to determine their likelihood of occurrence and the possible severity of the consequences of such events.

Included in the research will be conventional analog devices used in current nuclear power plants as well as digital computer-based protection, control, alarm, and displays that may be proposed as replacement equipment for future plant upgrading. Evaluations will be made of digital computer hardware systems in which both control and protection functions are performed to establish whether such systems can be designed and operated with a low probability of undesirable interactions between the control and protection functions. Digital computer software verification techniques will be evaluated to ensure that computers can be safely used in control systems, in protection systems, or in systems that could perform both functions. Digital computer-based systems for use in alarm and information display systems will also be evaluated.

Control and protection systems and instrumentation proposed for use in these systems in LMFBRs will be evaluated for their adequacy, and additions to modifications of regulatory guidance needed for such systems and equipment will be identified.

All instrumentation, control, and electrical power systems and components important to safety will be reviewed to determine their relative importance to safety. On the basis of criteria corresponding to a graded approach of at least two classes of systems and components important to safety (Class 1E and Class 2E), a classification of these systems and components will be possible.

Appropriate regulatory guidance will be developed to accomplish this with emphasis on criteria for reliability, qualification, and testing.

The major research products will be:

- o Methodology for performance of augmented failure mode and effects analysis (FMEA) for control systems (1984).
- o Resolution of USI A-47, "Safety Implications of Control Systems" (1984).
- o New or revised regulatory guidance for systems not required for safety (1985).
- o Regulatory guidance for setting priorities for alarms (1985).
- o Criteria for use of digital computers in systems important to safety (1986).
- o Design, qualification, and testing of regulatory guidance for I&C systems important to safety, not considered "safety-grade" (1987).
- o Regulatory guidance for protection, control, and instrumentation systems for LMFBRs (1988).

10.2 Component Assessment

The component assessment program will determine needed activities to ensure that proper performance of I&C and electrical systems and equipment important to safety can be achieved in a nuclear power plant. This program provides research needed to evaluate individual instrumentation and electrical system hardware components to (1) determine the mechanisms that can result in component or system malfunctions and to identify ways of reducing the likelihood of undesired failure modes and (2) assess the effects of design, manufacture, installation, test, and maintenance practices on the ability of these systems to meet their performance requirements under adverse situations. The component assessment program is related to the RES Qualification Testing Evaluation (see Section 4.1) and the NRR Equipment Qualification Programs in which the equipment qualification tests are being evaluated for their adequacy. In particular, the component assessment program provides identification of degradation mechanisms (see (1) above) and various practices (see (2) above) to ensure that proper test configurations and acceptance criteria are chosen for qualification tests or analyses.

10.2.1 Major Regulatory Needs and Their Justifications

1. 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 I&C and electrical equipment important to safety (FY 1985).
   Justification: Regulatory review of I&C and electrical power system component qualification is based on the assumption that the modes of failure or degradation relating to a component's use have been determined

and that these are used to form the basis for qualification test acceptance. 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 better control the failure or degradation modes.

2. Information on the current state of the art and current practices for measuring variables recommended in Regulatory Guide 1.97, including, for example, identification of instrumentation deficiencies and calibration of equipment for measuring extremes of parameters, to provide a technical basis for implementing the guide (FY 1984).
Justification: Information for following the course of an accident, based largely on engineering judgment, has been recommended in Regulatory Guide 1.97. The confirmatory research is needed to provide the technical basis for regulatory decisions on the acceptability of specific proposals addressed to Regulatory Guide 1.97 that may be submitted by various licensees. Research is also needed to ensure the adequacy, or identify the upgrading required, of current equipment and practices.

3. Confirmation that components for measuring important parameters during the course of an accident are available and adequate, to be the basis for revising Regulatory Guide 1.97 (FY 1985).
Justification: In reviewing applicant or licensee submittals on implementation plans for Regulatory Guide 1.97, the NRR staff must be aware of the availability of accident-monitoring instruments and the ability to ensure their operability in an accident in order to respond to requests for exceptions or deviations from the recommendations of Regulatory Guide 1.97.

4. Confirmation of techniques for measuring response time of some instrument components important to safety, to be one of the bases for revising Regulatory Guide 1.118 (FY 1985).
Justification: In accordance with Regulatory Guide 1.118, there is the need to confirm the availability of a remote in situ technique for response time testing of pressure sensor/sensing line systems. Response time testing techniques for neutron detectors may provide valuable information on the operational availability of these detectors.

5. The safety consequences of challenges brought about by electromagnetic interference (EMI) and lightning events on I&C equipment important to safety, to be the basis for a possible regulatory guide (FY 1986).
Justification: EMI and lightning events could possibly cause multiple failures in I&C equipment important to safety, particularly systems using sensitive solid-state devices. Such multiple failures could lead to events that were not previously analyzed and that may exceed the protection system design bases.

6. Evaluation of the test frequency for engineered-safety-feature actuation systems (ESFAS) to establish a proper balance between testing to ensure operational availability and wear that could impact on reliability, for use in the revision of Regulatory Guide 1.118 (FY 1986). (See Section 2.4 on the aging of electrical and mechanical components, a program closely coordinated with this need.)

Justification: Frequent testing of ESFAS equipment may cause excessive wear that could increase the potential for failure of such systems, thus potentially affecting safety.

7. Evaluation of the adequacy of current guidance in Regulatory Guide 1.106 on thermal overloads of electric valve operators to determine the need to revise the guide (FY 1988).
   Justification: Current guidance provides that overload protection be bypassed during accident conditions or be conservatively set so that inadvertent trip will not occur. A balance needs to be developed to ensure that the risk created by a burned-out motor due to bypassed protection is not greater than the motor's being inoperative as the result of an overload trip.

10.2.2 Research Program Description

The strategy for this research is to assess the availability and performance of instrumentation hardware for monitoring parameters in a nuclear plant that is important for diagnosing the onset of off-normal conditions and for use during the course of an accident. This research will also identify possible degradation and failure modes for I&C and electrical components important for safety.

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

Data from this program will be factored into the standards activities to ensure that the standards and regulatory guides focus on the important characteristics for each generic component. Where appropriate, these data can be factored directly into the licensing and inspection and enforcement programs in the form of equipment qualifications 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). I&C and electrical components that are unique to an LMFBR will be assessed with regard to availability and performance during postulated off-normal conditions and for use during the course of an accident. Degradation and failure modes for those components that can be expected in an LMFBR environment will also be identified, and current regulatory guidance will be modified where necessary.

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

I&C components considered to be sensitive to particular transient electromotive or magnetic interference conditions will be examined to determine the extent of the sensitivity and to help identify protective measures that may be useful.

The major research products will be:

- Regulatory guidance for testing response time of pressure and neutron sensors (1984).
- Regulatory guidance for design, manufacture, installation, and maintenance of terminal blocks and pressure transducers (1985).
- Regulatory guidance for response time testing of pressure transducers with their associated sensing lines (1985).
- Regulatory guidance for protection against EMI (1986).
- Regulatory guidance for individual instruments used to follow the course of an accident (1988).
- Revisions to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (1983 and 1988).

10.3 Diagnostic Equipment and Capability

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

10.3.1 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 that will help (1) prevent damage to equipment and systems important to safety or avoid unnecessary radiation exposure of personnel and (2) prevent or ameliorate accidents, to provide a technical basis for licensing criteria and regulatory actions on operating reactors (FY 1986).
   Justification: Diagnostic techniques and instrumentation (e.g., noise diagnostics) can provide warning of impending failures or malfunctions so that corrective action can be taken to prevent those failures and possible impairment of safety. Research is needed to assess the need for and the practicality of such techniques.

2. A technical basis for regulatory decisions concerning the adequacy of diagnostic instrumentation (FY 1985).
   Justification: To establish an NRC position on what additional, if any, diagnostic instrumentation should be required, an evaluation is required to assess the availability, practicality, and performance of diagnostic instrumentation during normal and off-normal operation as well as during shutdown conditions. Research is needed to provide this information.

3. Information on diagnostic instrumentation to be provided to reactor operators in a clear and easily understandable form enabling them to recognize the onset of reactor anomalies, to provide a technical basis for licensing criteria (FY 1985).
   Justification: Diagnostic information must be made available to reactor operators to enable recognition of reactor anomalies and to minimize the possibility of misinterpretation and incorrect actions.

10.3.2 Research Program Description

The strategy for this research is to evaluate various alternative means of diagnosing unanticipated events so that plant transients, equipment damage, and accidents can be avoided. A cooperative effort involving two utilities and coordination with EPRI will be included to ensure validity of results to operating LWRs. Conventional analog and digital instruments will be evaluated in addition to techniques using noise signature analysis of the a.c. component of analog instrument signals.

Instruments being employed in current nuclear power plants will be evaluated for their usefulness in providing information that can warn of impending failures or malfunctions so that actions can be taken to prevent plant system malfunctions or equipment damage.

An assessment will be made of an automated continuous on-line surveillance and diagnostics system to detect and diagnose operating PWR and BWR faults. The demonstration of a continuous on-line surveillance and diagnostics system at the TVA Sequoyah Unit 1 PWR will be continued, and plans are being made to demonstrate the system at one of the Peach Bottom BWRs.

An evaluation will be made of the capability of candidate continuous on-line radiation monitors to provide an early detection of failed fuel in the reactor core.

Signals provided to conventional analog instruments will be evaluated using techniques intended to identify resonances in the a.c. component of those signals that are indicative of abnormalities in plant equipment.

It is anticipated that the surveillance and diagnostic research group working in the above areas will be called upon by NRC to investigate reported anomalies at operating reactors in which they can apply results of the research efforts.

The major research products from this element will be:

- Formulation and demonstration of diagnostic techniques using a.c. signal components (1984).
- Regulatory guidance on use of conventional analog and digital instruments for diagnostic purposes (1986).
- Regulatory guidance on design, qualification, and testing of diagnostic instrumentation (1987).

- o Demonstration of an automated continuous on-line surveillance and diagnostics system at operating PWR (1984) and at operating BWR (1987).

10.4 New I&C Technology

This element provides research needed to evaluate technological advances in the state of the art of I&C and electrical equipment developed for other fields of technology and determines their applicability and acceptability for use in nuclear power plants (NPPs).

10.4.1 Major Regulatory Needs and Their Justifications

1. A technical basis for the use of state-of-the-art I&C technology that has been developed in other fields, to be used as the basis for the development of new or improved systems for NPPs (re: improved safety charter as stated in paragraph f of the NRC Fiscal Year 1978 Authorization Act [Public Law 95-209]) (FY 1986).
   Justification: The advances of I&C technology in other fields provide equipment that is potentially more reliable and efficient or that has improved performance over existing equipment. The application of some of these advances to commercial NPPs could enhance the overall safety and thereby reduce risk. However, most current regulatory requirements and guidance are based on older technology and may inhibit the use of the new technologies. This may also cause regulatory delays when new technology is proposed for use in a NPP.

2. A review of NRC regulations, regulatory guides, branch technical positions, and other technical guidance on I&C to determine the changes necessary to permit the use of acceptable alternative approaches (FY 1988).
   Justification: A review should be performed to determine if additional alternative approaches should be adopted in NRC regulatory documents or if such documents should be modified to allow other alternative approaches by making the documents less restrictive.

3. Methodology to enable the potential value and impact of I&C regulatory proposals to be consistently and fairly evaluated on the basis of their relative importance to the overall system involved, to be used in a standard review plan for proposed I&C hardware (FY 1987).
   Justification: I&C hardware comprise only a part of the system that involves actuated equipment such as pumps and valves and other process system hardware such as vessels and heat exchangers. When changes are proposed for I&C hardware, the value and impact of these changes have been difficult to evaluate from a regulatory viewpoint because the I&C components are only a fraction of the entire "system" needed to accomplish a particular function.

10.4.2 Research Program Description

Improvements in overall reactor safety in the future will frequently have to trade off between process system modifications and changes in I&C. Because of the relatively lower cost of changing I&C, this choice is likely to result in a number of these changes. Such I&C changes will consider the use of advances in I&C technology since the affected plant(s) was built.

In view of the above, the strategy for this research is to review advanced instrumentation, control, and protection techniques used in other industries and determine whether these techniques can be used in the nuclear industry in a safe and effective manner. Preliminary arrangements have been made for a cooperative effort with cognizant professional societies, including the Instrument Society of America (ISA) and IEEE, and with EPRI to ensure that the equipment and systems with the greatest potential are examined.

Instruments using digital signals for measuring process variables, microprocessors and minicomputers for use in controls, and operator displays will be evaluated.

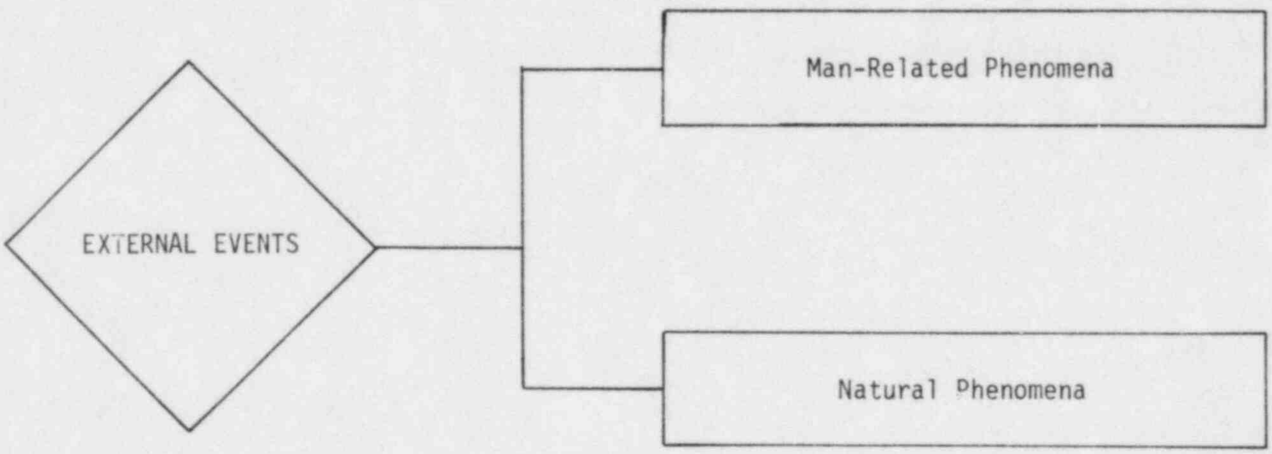
Current regulatory guidance and regulations will be examined to ensure that new techniques that can be employed in a safe and effective manner are not prohibited, especially when these techniques may be used in replacing existing equipment in operating NPPs.

The major research products from this element will be:

- o Jointly sponsored (ISA, EPRI, NRC) conference on use of new I&C technology in nuclear power plants (1984).
- o Regulatory guidance on use of advanced instruments (adaptation of technology from other industries) (1986).
- o Regulatory guidance on use of advanced control and protection techniques (adaptation of technology from other industries) (1987).

External Events

11



11

FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
EXTERNAL EVENTS	$12.5	$16.0	$12.7	$12.3	$12.0

11. EXTERNAL EVENTS

External events in the form of man-related and extreme natural phenomena pose a threat to the safe operation of nuclear facilities. The character of these events and the probabilistic distribution of their magnitudes affect facility design, operation, and siting, as well as the level of risk associated with an operating facility. Uncertainties in the characterization of these events and their probabilities lead to conservatism in regulation and uncertainty in risk assessment. It is important that the resulting uncertainty in risk estimates be adequately and properly assessed and, if significant, reduced. This chapter discusses both man-related phenomena and natural phenomena.

11.1 Man-Related Phenomena

Some man-related events that occur outside nuclear facility buildings have the potential of compromising the safety of the facility. The probability of safety-related equipment failure or degradation of equipment availability or effectiveness, the availability or effectiveness of the plant operator, or the severity of accidents may all be affected by these events. These events may occur during transportation or in the manufacture of hazardous materials, may be the result of transport vehicles such as aircraft striking the plant or related equipment, or may be the result of a military accident (e.g., crash of a military aircraft).

When such events happen, they can affect the facility in a number of ways. Release of toxic or irritant chemicals or obscuring smoke could reduce the effectiveness of plant operators or completely incapacitate them. Aircraft impact on plant structures or the impacts of explosion-generated missiles or the pressure waves from these explosions could destroy or degrade safety-related equipment. Heat from fires and vapor clouds of gases that are in explosive or flammable concentrations could also pose a threat by causing malfunction of emergency or other safety systems. Safety-related equipment might also be degraded by corrosive, conductive, or nonconductive vapor clouds or aerosols. These materials could affect electronic control circuits, relays, switches, electrical motors, other insulation, etc. They could clog filters, reduce cooling efficiency by blocking airflow to vital equipment or by coating cooling surfaces, coat instrument faces or other optical surfaces, or distort sensing apparatus.

11.1.1 Major Regulatory Needs and Their Justifications

1. Information concerning the possible adverse effects on safety-related plant equipment of materials resulting from external man-related events, to be the basis of a series of reports (FY 1983-1988).
   Justification: There is a probability that corrosive, conductive and non-conductive, or flammable gases or aerosols could prevent or significantly degrade the operation of safety-related equipment. Little evaluation has been done on this subject to date.

2. Information that will allow more reliable analysis of dispersion of both small and large releases of nonradioactive gases and aerosols that might compromise the operation of safety-related equipment in nuclear facilities, to be the basis of a series of reports (FY 1986-1988).
   Justification: The dispersion of buoyant and nonbuoyant explosive, toxic, or corrosive vapor clouds is not sufficiently well understood to realistically assess the risk from offsite spills of such materials. The change in character of the hazard with dispersal is also not well understood (i.e., gases that chemically react during dispersal and the explosion or deflagration mechanisms of gas/air mixtures). Model verification by experimental means is essential to reduce uncertainty in risk calculations and to remove unnecessary conservatism in the regulatory process. NRC should be only partially responsible for this research. The petroleum, chemical, and insurance industries as well as the Department of Defense and the Coast Guard have a need for similar information. Interagency cooperation is essential since experimental verification is very expensive.

3. Information on the mechanisms, times, and consequences of operator incapacitation resulting from external events, to be the basis for a series of reports (FY 1986-1988).
   Justification: No comprehensive studies have been made to understand the consequences to the safety of plants of the inability of the operator to function at full capacity. Models for determining mechanisms and times of incapacitation from exposure to various gases have been developed but are incomplete and unverified. If consequences are unacceptably high, regulatory requirements for additional control automation may be justified.

4. Refined methods for analyzing potential impacts of explosion-generated missiles and aircraft or other transport vehicles on nuclear facilities, to be the basis for a series of reports (FY 1986-1988).
   Justification: A number of calculational and procedural uncertainties exist in the methods used by the staff to analyze the consequences of missiles or aircraft striking areas of plants other than containment. Analysis of fire/explosion/impact characteristics of aircraft combined with plant systems susceptibilities are necessary in order to determine possible consequences.

11.1.2 Research Program Description

No resources are available in FY 1984 for initiating the program on man-related phenomena. Following the 2-year trial period of the Commission's Safety Goal for operating nuclear power plants and revision of the accident source term, it is planned that this research will be started.

In order to systematically deal with the contribution to risk from external events, this program will identify and evaluate those types of man-related events that might adversely affect the safe operation of a nuclear power plant. Programs will be developed to identify the potential sources of toxic, corrosive, flammable, and explosive clouds and to describe their transport, dispersion, and chemical characteristics. Other programs will address missiles from offsite explosions and aircraft impacts to establish the spectrum of such events that might occur.

The program will address the ways in which the performance of safety-related systems outside containment might be impaired by the presence of toxic or corrosive clouds or the effects of fire or pressure waves from onsite or offsite explosions. Information on the functioning of equipment under these adverse conditions will be obtained from literature and experimental test programs. The potential for impaired operator judgment and effectiveness, or total incapacitation due to toxic or irritant gases, will be studied as well as mechanisms to control or eliminate the possibility of such occurrences. This information will be correlated with research on operator stress being conducted in the human factors program. The vulnerability of emergency power supplies and other safety-related equipment to adverse chemical environments, fire, pressure waves, and impacts of missiles will be studied.

The research products will be:

- Technical reports on mechanisms of dispersion of hazardous material and validation of these models (1986-1988).
- Technical reports characterizing the extent of the effects of adverse hazardous materials on safety-related equipment (1987-1988).
- Technical reports analyzing consequences of damage or degradation of safety-related equipment or systems (1986-1987).
- Technical reports detailing mechanisms and consequences of operator incapacitation (1986-1988).

11.2 Natural Phenomena

Extreme natural phenomena such as earthquakes, floods, and severe weather represent the natural hazards to nuclear facilities. A major source of uncertainty in assessing the potential risk from seismic hazards results from limited knowledge of the source areas of Eastern United States seismicity, the propagation characteristics of seismic energy, site-specific attenuation, earthquake recurrence intervals, and the probabilistic distribution of seismic events. Significant uncertainties also continue to exist relative to safety margins in the design of nuclear facilities associated with meteorological and flooding hazards.

11.2.1 Major Regulatory Needs and Their Justifications

1. Data concerning seismic source zones in the Eastern United States, including Charleston, New Madrid, New England, and others, 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 (FY 1988).
   Justification: In the Western United States, the cause of seismicity is reasonably well understood, and the location of the major faults is known. Except for the New Madrid seismicity, the distribution of seismicity in the East, including New England and the vicinity of Charleston, S.C., has not been well defined. No working hypothesis for the cause of the seismicity has been generally accepted by the geoscience community. The major faults and associated strain fields, including those near New Madrid, are poorly known.

2. Information on how seismic energy is attenuated or apparently amplified, to be the basis for amendments to Appendix A to 10 CFR Part 100 and for developing a regulatory guide (FY 1986).
Justification: The significance of the uncertainty in defining the design vibratory ground motion for nuclear power plants has recently been highlighted. The probabilistic risk assessments for Zion and Indian Point have clearly shown that the seismic hazard is a prime contributor to the overall risk and is a source of significant uncertainty in risk assessment.

3. 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 (FY 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 these records and their use in licensing decisions.

4. Methods for estimating the margins of uncertainty and the possibility of underestimating external flood probability with the current procedures, to be the basis for developing a regulatory guide (FY 1986).
Justification: The NRC licensing staff has identified the need for assessing the long-term representativeness of flood records, defining methodology for selecting flood event confidence limits, and assessing the likelihood of failure of flood protection at nuclear power plants. Further, the ACRS has formally recommended to the Commission that priority be given to research on flood event probabilities.

5. An assessment of the margins of safety associated with the use of the existing data bases for extreme and severe meteorological events to reduce uncertainties and eliminate any excessive requirements. This program is nearing completion, and the information will be used as a basis for revising existing regulatory guides (FY 1985).
Justification: Severe and extreme meteorological events (e.g., high windspeeds) may contribute significantly to the overall risk, as indicated by recent PRA studies for nuclear power plants. Uncertainties in the frequencies of occurrence for such phenomena are sources of the uncertainties in the PRA.

11.2.2 Research Program Description

11.2.2.1 Solid Earth Sciences

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 cause 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.), crustal stress measurements, and studies of recent crustal movements. This program involves 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 data sets where available. Other programs involve upgrading seismic instrumentation and analysis of data to establish seismic wave attenuation relationships and to limit uncertainties in ground motion. A network of strong-motion seismographs is being established to gain information on site-specific response and attenuation relationships.

The probabilistic sensitivity study consists of an investigation of the probable ground motion predicted by the various hypotheses on the cause of seismicity in the Eastern United States.

The major products of this research will be:

- o Seismographic network data used on a day-to-day basis by licensing staff, by staff involved in PRAs, and for rulemaking decisions and engineering research projects (1984-1988).
- o Probabilistic sensitivity study of probable ground-motion dependence on the various proposed causes of seismicity in the Eastern United States (1985).
- o Geophysical data for determining crustal structure in areas of suspicious geologic structures (1985).
- o Data from the in situ stress measurement program in the Northeastern United States (1986).

11.2.2.2 Meteorology

The research program for severe and extreme meteorological events is expected to receive decreased emphasis over the next few years as the work in this area nears completion. The purpose of the program is to (1) improve the data bases for the frequency of occurrence and magnitudes of meteorological events to include those such as hurricanes and tornadoes that produce extreme windspeeds and (2) determine their respective hazard probabilities. The national hazard probabilities will then be defined regionally and temporally for use in evaluating the associated risks to nuclear facilities.

The major products of this research will be:

- o Unification of different tornado data sets (1983).
- o Definition of design basis tornado characteristics (1983).
- o Regionalization of design basis tornado characteristics (1984).

- Assessment of the relationship of extreme "fastest-mile" windspeeds to extreme windspeeds for other time durations (1984).

11.2.2.3 Hydrology

The research effort to analyze methods for estimating the margins of uncertainty and the possibility of underestimating flood probability will involve assessing the long-term representativeness of flood records, defining methodology for selecting flood event confidence limits, and assessing the likelihood of the failure of flood protection at nuclear facilities. For assessment of generic ground-water site conditions and mitigative techniques, the work will involve analyzing the present nuclear facility site conditions and available mitigative techniques and developing an interdictive strategy for the ranges of site conditions and time considerations.

Major products of this research will be:

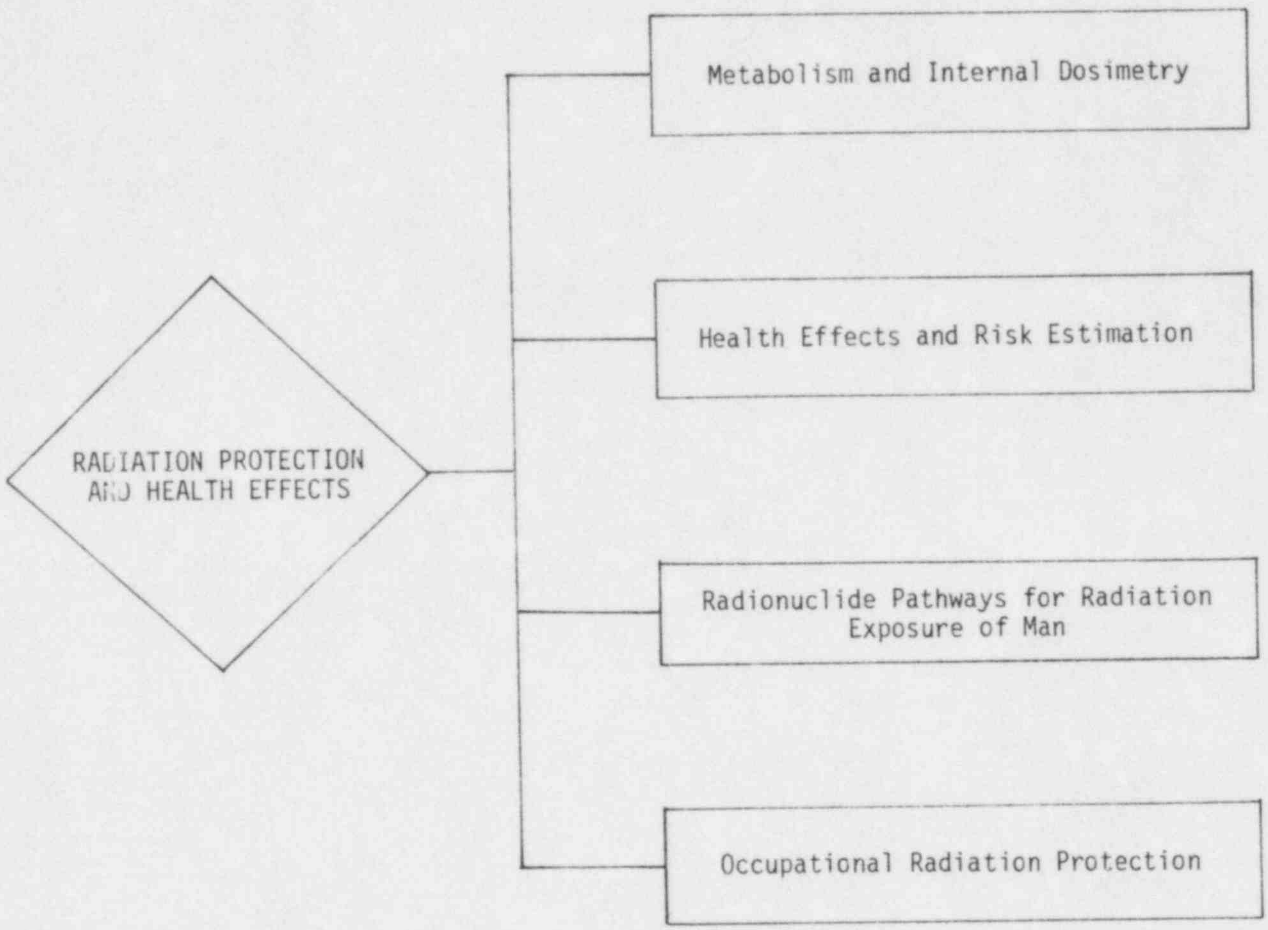
- Regulatory guidance on interdictive strategies for severe event accidents at nuclear facilities dealing with ground-water contaminant transport and available mitigative techniques (1984).

- Regulatory guidance on methodology for selecting flood event confidence limits (1987).

- Technical information reports on long-term representativeness of flood records and likelihood of failure of flood protection at nuclear facilities (1988).

Radiation Protection and Health Effects

12



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
RADIATION PROTECTION AND HEALTH EFFECTS	$5.3	$5.2	$7.2	$8.0	$8.6

12. RADIATION PROTECTION AND HEALTH EFFECTS

The goal of radiation protection is to ensure that any individual and societal risk of health damage resulting from licensed activities are not at or above unacceptable levels and are as low as 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 i sues 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.

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.

12.1 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.

12.1.1 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 (FY 1987).
   Justification: Materials generated during the production of reactor fuel that contain 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 and some activation products, the metabolism of and internal dose from these materials are not well understood. For example, how much of a given radionuclide is deposited and retained in the lung after an intake by inhalation is very dependent on the chemical and physical state of the inhaled aerosol. 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 (FY 1987).
   Justification: Although extensively investigated, uncertainties remain concerning intake parameters for some transuranic elements. Because of differences in physical and chemical states, industrially produced materials are likely not to behave the same as the laboratory materials 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 licensing of reactors using mixed oxides or plutonium fuels. For example, recent studies on the gastrointestinal absorption of plutonium suggest problems with the transfer factors on which present standards are based.

3. Methods for calculating internal doses to be used for incorporating recommendations of the International Commission on Radiological Protection (ICRP-26, 1977) into 10 CFR Part 20 and for developing a regulatory guide on summing internal and external doses (FY 1985).
   Justification: If the ICRP-26 recommendation of a dose limitation system based on health risk is adopted in the revision of 10 CFR Part 20, it will be necessary to provide practical methods for summing internal and external doses for those cases where simplified approximate methods are not adequate. In order to do this, absorbed doses (rads) must be converted to dose-equivalent in units of rem. This requires reduction of uncertainties in

the use of quality factors for different types of radiation. Especially important for alpha particles is the uncertainty in averaging the dose estimate over the whole organ versus considering only the dose to those cells actually exposed.

12.1.2 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 from this element will be:

- Values for the gastrointestinal absorption factor for plutonium (1984), neptunium (1985), and other actinides (1986-1987).
- Metabolic model for inhaled mixed oxides (1985).
- Metabolic model for inhaled yellowcake (1987).
- Assessment of internal dosimetry code (1988).
- Compilation of age- and sex-dependent dose conversion factors (1988).

12.2 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.

12.2.1 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 (FY 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 likely 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 (FY 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 (FY 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.

12.2.2 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.

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 with causative pathology and genetic effects will be observed end points.

A statistical reevaluation of the effects of exposure to radon decay products on lung cancer risk in nonsmoking uranium miners will begin in 1984. Potential scientific merit and feasibility of epidemiological studies of inhaled radon decay products and associated lung cancer risk in two non-hard-rock miner populations, radium dial painters (FY 1984), and coal miners (FY 1985) will also be initiated. 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. The relative merits of several modern statistical methodologies for hazard evaluation will be assessed beginning in FY 1984 to determine how much the estimated radon risk is influenced by the choice of statistical procedures. These studies 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 provide dosimetric data for these radon studies, 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.

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:

- Revised (or reaffirmed) values for neutron quality factor (1986).
- Improved statistical procedures that will find application in other hazard evaluation studies (1986).
- Dosimetric data for assessment of risk for radon and its daughters (1986).
- Evaluation of magnitude of non-radon-related lung cancer risks in uranium miners (1986).

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

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

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

12.3 Radionuclide Pathways for Radiation Exposure of Man

In order to estimate or predict the potential radiation exposure to the public from NRC-licensed facilities and therefore to evaluate the potential health consequences of these exposures, it is necessary to understand and model the transport of radionuclides from the point of release to the point of human intake, i.e., the mouth, nose, and skin. It is also necessary to estimate the variability and uncertainties in the models and in the numerical values of the parameters chosen.

12.3.1 Major Regulatory Need and Its Justification

Validation and improvement of methods used by NRC for predictive assessment of population exposure and revision, as warranted, of regulatory guides describing these methods (FY 1986).
Justification: Existing transport models used to predict the transfer of radionuclides from facility effluents to human intake are adequate for routine release applications. However, many of the numerical values for the parameters used in these existing models are derived from atypical fallout studies or laboratory investigations. Validation of transport models and values is needed to depict more realistic assessments applicable to reactor effluents and field conditions. Past NRC studies of radionuclide transport and accumulation in fresh-water systems have focused on rivers. Additional research is needed to validate or improve predictions for lakes, estuaries, and oceans. More attention is needed on the role played by agricultural practices and climatological conditions in exposure assessments that are dependent on site-specific variations for licensed facilities. One immediate area where these differences may be important is for expected uranium mining and milling in the Eastern United States (e.g., Virginia, New Hampshire). NRC's present models were developed for the more arid conditions of the Western States.

12.3.2 Research Program Description

The strategy is to rely primarily on compilation and analysis of existing data supplemented as necessary by field and short-term laboratory studies. Validation of predictions for aquatic systems will rely on past Clinch River and Columbia River field studies to provide data that can be tested against present NRC aquatic models. Comparisons, using data from the Hudson River estuary and from Lake Michigan both of which have been extensively studied, will also be done.

Variations in transport parameters will be based on examining existing data on differences in regional farming and agricultural practices. These data will

be supplemented by stable elements analyses to quantify transfer factors in cases where unusual practices are observed.

The major research products will be:

- An improved data base of radionuclide concentration factors in fresh-water and marine ecosystems (1986).
- Estimates of radionuclide loss from surface soil due to leaching by rainwater (1985).
- Estimates of the amount of radionuclide removal from ground surfaces by wind in non-arid areas (1986).

12.4 Occupational Radiation Protection

Research in the occupational radiation protection program provides information needed to ensure an adequate degree of radiation protection for workers in NRC-licensed activities. Implementation of this research through NRC regulations and guidance ensures consistency with national and international advances in radiation protection methodology.

12.4.1 Major Regulatory Needs and Their Justifications

1. Development of methods for evaluating the effectiveness of dose-reduction techniques in nuclear power plants, to be the basis for implementing 10 CFR Parts 20 and 50 and for developing regulatory guides as warranted (FY 1985-1988).
   Justification: As nuclear power plants grow older, plant modification, inservice inspections, and major repairs tend to subject workers to higher radiation exposures. Effective dose-reduction techniques are needed to arrest this trend. Research to evaluate such techniques is essential to provide a technical basis for appropriate regulatory requirements to accomplish this and to implement ALARA programs.

2. Improvements in health physics measurement techniques, to be the basis for implementing 10 CFR Part 20 and for developing regulatory guides as warranted (FY 1983-1988).
   Justification: Evidence exists that certain health physics measurements are not being performed with an acceptable degree of accuracy. Without such accuracy, determinations of radiation exposures are not sufficiently reliable to ensure compliance with regulatory dose limits. Inaccurate measurements may also prevent the accumulation of a data base sufficient to identify trends in radiation exposures. These inaccurate measurements are a major source of error in any epidemiological studies of workers intended to improve the dose-effect assumptions. Through continuing research to establish performance criteria for improving health physics measurement techniques and instrumentation, better control of radiation exposures may be achieved under the requirements of 10 CFR Part 20.

3. Evaluation of methods for improved control of radionuclide intakes, including air-sampling techniques and bioassay, to be the basis for developing regulatory guides as warranted (FY 1983-1988).
   Justification: Recent ICRP recommendations have modified the dosimetric models used to assess doses from the internal deposition of radionuclides and have changed their annual limits of intake and derived air concentrations for workers to reflect these modifications. In addition, the ICRP has recommended a system of dose limitation in which the summation of internal and external doses is used. Research is required to evaluate improved methods for control of radionuclide intake as a basis for revising NRC radiation protection regulations and guidance consistent with these advances in methodology. This research will include air sampling, bioassay, and dose calculational methods.

4. Improvements in operational practices related to health physics at nuclear power plants to achieve ALARA exposures (FY 1986).
   Justification: 10 CFR Part 20 indicates that licensees should make every reasonable effort to maintain radiation exposures ALARA commensurate with sound economic and operating practices. Research is needed as a basis for improvements in operational practices related to health physics at nuclear power plants to be implemented as part of the NRC's improved safety program.

12.4.2 Research Program Description

The research strategy is to coordinate the planning of research activities with other Federal agencies and industrial and professional organizations with mutual interests in occupational radiation protection to maximize the use of experimental programs and scientific data for related requirements. Research is directed at technical issues that will provide answers to short- and long-term regulatory needs of other NRC offices and for further implementation of the ALARA concept consistent with advances in radiation protection methodology.

The research program for reduction of occupational dose in nuclear power plants will involve experimental investigation of the effectiveness of decontamination alternatives. Computer codes will be developed and tested to assess the relative effectiveness of these decontamination alternatives. Problems of worker exposure during waste disposal associated with decontamination will be assessed on site. The effects of design and operational changes on reduction in dose resulting from corrosion product buildup will be investigated through studies of sampling, procedural developments, and component design, as well as transport and deposition models. Job analyses, radiation protection training techniques, and worker incentives will also be examined as means of facilitating dose reduction.

Research to improve health physics measurements will encompass the development of additional measurement standards, including national beta radiation standards, calibration programs relevant to radiation spectra typical of NRC-licensed activities, and improved techniques for the measurement of neutron energy spectra and the assessment of neutron doses. Advancements in health physics instrumentation and dosimetry techniques, including onsite applications to improve measurement capabilities, will be studied.

To ensure compliance with NRC radiation protection limits, developments in bioassay techniques and air-sampling methods will be evaluated. Laboratory and in-plant testing programs will be conducted to evaluate new methodology for dose assessment. Updated mathematical models of human uptake, physiological distribution, and excretion of radionuclides will be used to improve computer programs for calculating internal doses and annual intakes from bioassay measurements.

Research will be conducted to develop an acceptable analytical method to determine, for specific nuclear power plant operations, whether additional improvements or changes to reduce occupational exposures are justified. The method will include the costs of additional protective measures and of health effects (theoretical cost). The ICRP-suggested approach (optimization) will be especially considered.

The major research products will be:

- Analysis of the health physics technician's skills as needed to define training requirements (1984).
- Recommendations for improved radiation protection training techniques (1985-1988).
- Identification for licensee management of incentives for dose reduction (1984).
- Methods for the evaluation of new techniques to reduce collective dose at nuclear power plants:
  - New decontamination alternatives for coolant system interior surfaces and fuel surfaces (1984-1988).
  - Design and operational changes to reduce dose from corrosion product buildup (1984-1987).
  - Assessment of problems of worker exposure during waste disposal associated with decontamination (1984-1986).
- Improvements or new techniques in health physics measurements that provide more accurate and reliable determinations of worker exposures, including:
  - Development of beta radiation standards and improved measurement techniques for beta radiation (1983-1984).
  - Recommendations for improved health physics instrumentation calibration (1984).
  - Recommendations for improved techniques to assess worker exposure to neutrons (1984).
- Recommendations for advanced respiratory protection techniques and bioassay methodologies to improve control of radionuclide intakes by workers (1983-1988).
- Recommendations for effective ALARA practices for nuclear power plants (1986).

- Optimization methods for determining the dose at which the total cost (health effects cost plus the cost of protective measures) would be minimized as a basis for evaluating whether improvements or changes to reduce occupational exposures are justified (1987).

Waste Management

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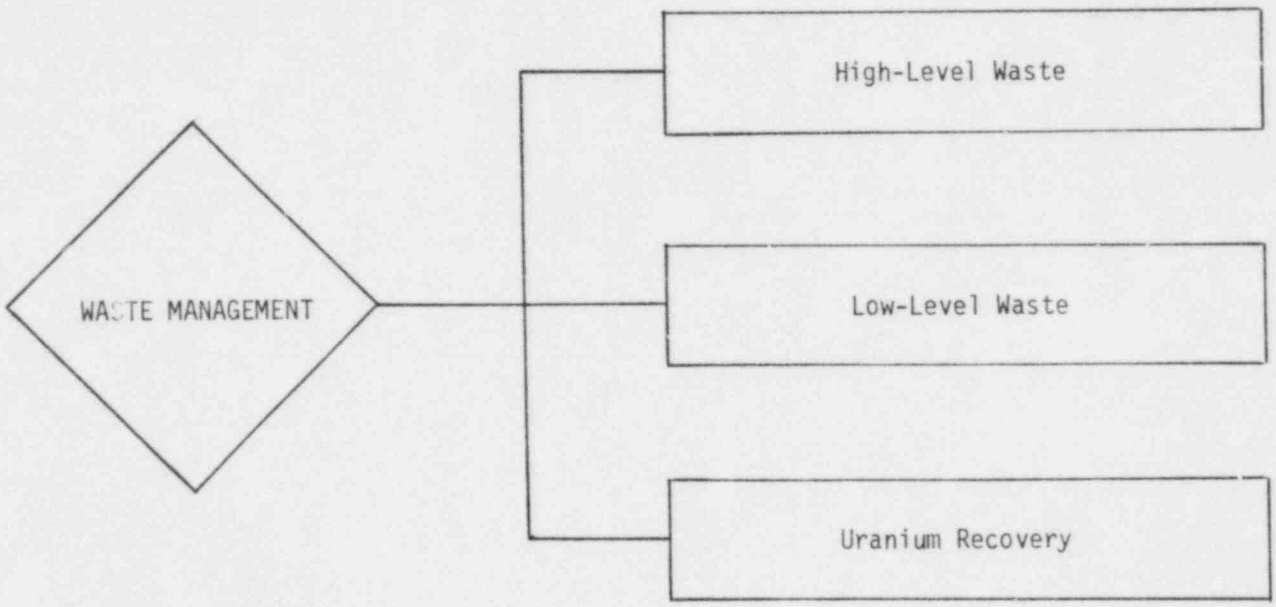

FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
WASTE MANAGEMENT	$9.3	$10.4	$15.5	$14.9	$14.2

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13. 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 and to assess risks associated with waste management operations and environmental impacts. The sources of uncertainty in the assessment of compliance and risk differ in the three program elements of high-level waste, low-level waste, and uranium recovery.

13.1 High-Level Waste

High-level-waste (HLW) management involves the regulation of operational safety, occupational radiation protection, and the long-term isolation of HLW. High-level radioactive wastes must be disposed of in compliance with standards and regulations established or to be established by the EPA and the NRC (40 CFR Part 191 and 10 CFR Part 60, respectively). DOE, the licensee, is developing plans to isolate HLW in facilities to be constructed in deep geologic formations that will comply with EPA and NRC regulatory requirements. In licensing an HLW repository, the NRC will assess whether there is reasonable assurance that DOE's license applications, if implemented, would comply with the EPA standard and NRC regulations. NRC's compliance assessment requires the technical capability to analyze relevant phenomena and processes. Therefore, NRC needs a means of evaluating proposed safety and radiological health protection programs and the impact of the repository on the environment and a means of predicting whether long-term releases from the repository will be in compliance with established requirements.

Licensing assessments must be provided in the following six regulatory areas in support of NRC authorization to construct the first HLW repository in 1992 and for NRC approval to emplace wastes in 1998.

13.1.1 Major Regulatory Needs and Their Justifications

1. Criteria for assessing whether engineered facility design and operational methods proposed by DOE for HLW repositories comply with 10 CFR Part 60 requirements for the performance of the geologic repository operations area through permanent closure and will provide occupational radiological safety protection in compliance with 10 CFR Part 20, to be used for assessing the application to construct the first HLW repository (FY 1988). Justification: The licensing staff must determine whether engineered facility designs and operational practices proposed by DOE will provide radiological safety in accordance with the requirements of 10 CFR Part 20. Although the NRC has extensive experience in licensing, inspecting, and enforcing radiation health protection for reactors, the NRC has no similar experience related to a deep underground repository facility. The underground facility will pose unique considerations and health protection uncertainties related to monitoring, waste emplacement, handling, ventilation, and accidental releases that will require a limited amount of research for resolution. Since there are no existing repository facilities for HLW, neither the NRC nor the DOE has operational experience.

2. Analytical methodology for evaluating the licensee's safety analysis reports (SARs) to assess compliance with 10 CFR Part 60 and 40 CFR Part 191, to be used for assessing DOE's final environmental impact statement (FY 1988).
Justification: 10 CFR Part 60 requires that the licensee submit an SAR that will predict performance with respect to site-related features of the repository. The NRC must develop analytical methodology to be able to evaluate the licensee's SAR. Reliable technical bases and methods are needed to enable NRC to evaluate repository performance, given the unique characteristics of an HLW repository. In addition, methods need to be developed to allow the licensing process to deal with the uncertainties associated with predicting future changes to the site that might affect waste isoltion.

3. Criteria for assessing whether waste package, HLW facility design, and operational plans will ensure that waste packages can be retrieved, if necessary, to be used for assessing the application to construct the first HLW repository (FY 1988).
Justification: The proposed performance objectives of the technical rule for HLW disposal would require that the repository be designed to allow the option of retrieval of any or all of the emplaced waste. Specifically, the design must ensure that the retrieval option is preserved for an adequate time to permit any corrective actions shown to be necessary. The design must be compatible with a monitoring program capable of measuring relevant parameters and conditions by confirmed methods. Research is needed to enable the NRC to determine if the DOE's performance confirmation program is capable of providing a basis for the NRC to find reasonable assurance that the repository will perform according to 10 CFR Part 60 and to provide the NRC with the analytical methodology in engineering and rock mechanics for evaluating the DOE's proposed retrievability plans to ensure that the option to retrieve is preserved.

4. Methodology for assessing whether HLW package designs proposed by DOE will comply with long-term radionuclide containment requirements defined in 10 CFR Part 60, to be used for assessing the application to construct the first HLW repository (FY 1988).
Justification: The potential hazard posed by HLW will last thousands of years. The Commission must have confidence that the wastes will remain effectively isolated from the biosphere as required by the EPA standard. Since the performance of engineered systems, notably the waste form and waste package, are intended to contribute substantially to that confidence, there must be assurance that containment of HLW will 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. Laboratory and field tests and experiments, even those lasting for decades, are of short duration compared to the long time at issue. Hence, understanding is needed to allow interpretation of these limited observations so that expected performance over the long term can be assessed with confidence.

The licensing staff must assess the capability of DOE's proposed waste package designs to contain the wastes over long periods of time. Research is needed to study and understand mechanisms of waste package degradation and failure as well as uncertainties in material properties, environmental conditions, and the mechanisms themselves. Methods of examination and testing of waste package performance that will allow scaling of short-term laboratory results to predict ranges in waste package life need to be established. In particular, there is an immediate need to improve understanding of the phenomena affecting corrosion and fatigue of proposed metal containers in thermally perturbed geochemical environments under expected repository conditions.

Licensing review will also include an assessment of the leaching characteristics of the waste form. Since there will be no opportunity to observe actual leaching of radionuclides for a waste form over very long periods, the results of short-term tests will have to be extrapolated to thousands of years. Further, the actual waste form will have undergone an aging process before the leaching process is expected to begin. Understanding of the experimental and testing methods is needed to allow confident scaling of laboratory tests to these long periods. This is one of many performance areas where an independent judgment by the NRC will be critical to adequate confidence in the expected performance of the DOE-proposed waste form.

5. Criteria for assessing whether the engineered facility designs proposed by DOE will comply with the release rate criterion of 10 CFR Part 60 by providing backup retention of radionuclides that may be discharged from waste packages that fail, to be used for assessing the application to construct the first HLW repository (FY 1988).
<u>Justification</u>: The NRC must assess the licensee's engineered facility designs to ensure compliance with the release rate criterion of Part 60, i.e., following loss of containment the release rate of radionuclides must not exceed one part in 10^{-5} per year of the inventory present 1000 years after permanent closure. NRC needs the capability to be able to evaluate models that predict phenomena that control the rate of radionuclide release from the facility, including contributions by waste form, waste container, overpack, and backfill. Release rates involve large unresolved uncertainties as a result of the impact of thermal perturbations, geochemical environmental parameters, and very-near-field hydrological flow conditions on the performance of engineered systems. There is a high-priority need for NRC to understand underlying phenomena and the sources of uncertainties concerning the methods for predicting solubility and leachability of waste forms at elevated temperatures and the anticipated geochemical and hydrological conditions of a closed repository. There is also a need to understand the phenomena and uncertainties concerning the capability of proposed backfills to control influx of water and sorption of radionuclides released.

6. Criteria for assessing the reliability of the HLW facility designs proposed by DOE in conjunction with the natural geologic parameters of the sites proposed for the facilities to comply with the long-term specific radionuclide limits set by EPA (40 CFR Part 191), to be used for assessing the application to construct the first HLW repository (FY 1988).

Justification: NRC must assess whether the proposed site and facilities will comply with the EPA standard for cumulative releases of specific radionuclides to the accessible environment over a period of 10,000 years. In order to establish a capability to predict radionuclide transport and to assess the licensee's transport models, the NRC must understand the geochemical-hydrological interactions that control radionuclide transport in repository environments. This requires understanding of geochemical and hydrological radionuclide transport models and models to predict the flow of water through fractured and porous zones in both saturated and unsaturated rock. Research will also be required to assess the impact of the thermal effects of the waste, site tectonics, structural geology, and rock mechanics on preexisting flow paths.

13.1.2 Research Program Description

The strategy of the HLW research program is to develop an understanding of basic phenomena and processes of HLW geologic disposal to form a technical basis for assessing DOE license submittals for construction and operation of HLW repositories so that the NRC can ensure that long-term performance objectives of 10 CFR Part 60 and 40 CFR Part 191 are met. Coordinated program efforts of laboratory and field experimentation, theoretical studies, and model development will provide an 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.

The research focuses on identifying and assessing uncertainties pertaining to waste package and engineered system performance, geochemical interactions with respect to radionuclide migration, ground-water transport of radionuclides, effects on geological stability and isolation due to perturbation from excavation and impacts from emplaced wastes, overall system performance modeling, and long-term postclosure risks. This program also addresses monitoring methods, instrumentation reliability, and plans for retrieval of emplaced waste packages that fail or for other reasons require remedial actions before closure of the repository.

The major research products will include:

- Model for assessing degradation of borosilicate glass waste forms based on surface kinetics (1984).
- Final report establishing the relationship between quality control of container-manufacturing technology and container characteristics (1985).
- Method for predicting long-term performance of waste packages (including waste form, container, and overpack) (1988).
- Final report on effects of oxidation state on radionuclide mobility (1985).

- Hydrological and geochemical models for assessing radionuclide transport processes (1986).

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

- Confirmation of DOE prediction of radionuclide source term (composition and flux) released from proposed waste packages (1988).

- Methods for evaluating long-term performance of backfill systems proposed by DOE (1988).

- Final reports on methods for evaluating the effectiveness of borehole plugging and sealing (1984) and shaft sealing techniques (1987).

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

13.2 Low-Level Waste

Low-level-waste (LLW) management includes occupational radiation protection, protection of the general population from releases of radioactivity, and environmental protection. LLW must be disposed of 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 be required to assess the compliance of license applications from non-Agreement States with NRC regulatory requirements. Agreement States will be assisted by NRC so that they can manage and regulate land burial of low-level wastes in a manner comparable to NRC safety regulatory practice. In order to determine compliance with 10 CFR Part 61, the NRC must model the phenomena and processes that affect the mobility of radionuclides. The uncertainties in the compliance assessment determinations must be sufficiently small to allow confidence that release criteria are met. 10 CFR Part 61 establishes minimum requirements on the form and content of waste acceptable for disposal. Continuing development of chemical decontamination agents and water chemistry control methods will cause the chemical characteristics of power plant wastes and the characteristics of subsequent solidification products to vary significantly from waste stream to waste stream as well as from batch to batch. Thus, solidification materials (waste form) such as hydraulic cement, bitumen, and vinyl ester styrene need to be evaluated for these variations in specific characteristics with respect to solidification effectiveness and product performance. Further, the NRC must establish adequate methods and procedures to determine compliance with criteria specified in 10 CFR Part 61 on waste classification, waste stability, site selection, facility design, and facility operation and site closure, including long-term institutional controls.

13.2.1 Major Regulatory Needs and Their Justifications

1. Evaluation of the applicability of NRC's existing data bases for determining whether land disposal facility designs and operational practices proposed by the applicant will provide adequate protection from radiation (FY 1986).
   Justification: The licensing staff must determine whether engineered facility designs and operational practices proposed by applicants will

provide radiological protection in accordance with the requirements of 10 CFR Parts 20 and 61. A limited amount of operational safety research will be needed to improve NRC's capability to evaluate the adequacy of the applicant's plans.

2. Analytical methods for evaluating environmental reports (ERs) from licensees to determine compliance with 10 CFR Part 51 and the National Environmental Policy Act (NEPA) (FY 1986).
Justification: NEPA and 10 CFR Part 51 require the licensee to submit an ER that will predict the impact of land disposal facility siting and operation on the environment. NRC must develop analytical methods to be able to adequately evaluate the licensee's ER. Data and experience at some humid sites in the Eastern United States have presented complex site performance characteristics. Predictions of environmental impact for these sites are thus difficult to make.

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

4. Methods and procedures for determining the long-term stability of packaged LLW, to be used in evaluating applicants' methods for ensuring stability (FY 1987).
Justification: Section 61.56 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 and (2) to the staff on evaluating applicants' methods for ensuring waste stability.

5. Criteria for assessing the performance of sites for land disposal of LLW to ensure protection of public health and safety, to be the basis for determining compliance with 10 CFR Part 61 (FY 1986).
Justification: For near-surface disposal of LLW, 10 CFR Part 61 specifies a series of characteristics that a site must have. However, the regulation does not provide detailed guidance on (1) how an applicant is to demonstrate that these characteristic are met, (2) how the NRC is to evaluate whether or not they are met, and (3) the level of confidence required to demonstrate protection of public health and safety. Research is needed to provide this guidance.

6. Criteria for assessing the acceptability of facility designs for land disposal of LLW, to be the basis for determining compliance with § 61.51 of 10 CFR Part 61 (FY 1986).
Justification: Investigations at some of the existing sites have shown that water entered trenches at excessive rates through the trench caps. Trench cap subsidence has resulted from degradation and compaction of waste packages, inadequate waste burial procedures, and inadequate trench

cap designs. Future licensees will need to use improved designs. Research is needed on evaluating facility designs to be able to determine compliance with criteria in § 61.51, as well as to seek further improvements in the safety of designs.

7. Criteria for assessing the acceptability of environmental monitoring plans for the preoperational, operational, and postoperational periods and for evaluating the data, results, and conclusions from the monitoring programs, to be the basis for determining compliance with applicable regulations (FY 1986).
Justification: Although the NRC has developed monitoring criteria for reactor sites and fuel cycle facilities, the pathways from land burial facilities are sufficiently different from the pathways at other nuclear facilities that different monitoring strategies and procedures may be needed. In particular, monitoring will be needed to detect radionuclide migration through unsaturated and saturated soil. Research on statistically valid monitoring is needed so that NRC will be able to ensure compliance with applicable regulations.

8. Methods for evaluating the design, operation, and closure of a land disposal facility that will enhance the protection of any individual inadvertently intruding into the disposal site, to be the basis for developing regulatory guides (FY 1988).
Justification: The criteria in § 61.42 do not specify how inadvertent intruders are to be protected. Sections 61.55 and 61.56 establish several criteria relevant to inadvertent intruders. Additional guidance is needed. Research needs to be conducted to identify methods and requirements appropriate for evaluating proposals for protecting inadvertent intruders.

9. Methodology for determining 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 (FY 1986).
Justification: In order to determine whether the criteria in § 61.41 are met, the NRC needs to establish methodology for predicting radionuclide transport, including how radionuclides become available for transport from waste form or package, and for assessing the licensee's transport models. The NRC must understand the geochemical-hydrological interactions that control radionuclide transport in land burial sites. This requires development and verification of coupled geochemical-hydrological radionuclide transport models to predict transport through both saturated and unsaturated geologic media. The NRC must also be able to assess the geochemical and hydrological parameters used in the applicant's transport models in order to have confidence in the model's output.

10. Criteria for assessing alternatives to shallow-land burial of low-level wastes, especially for wastes that have higher concentrations than are acceptable for Class C wastes, to be the basis for rulemaking, developing regulatory guides, and evaluating applications (FY 1988).
Justification: 10 CFR Part 61 does not yet address the site suitability requirements for land disposal methods other than near surface. Such

methods include aboveground or engineered facility disposal, deep-well disposal, and mined cavities, including consideration of Class C wastes. Research is needed (1) to determine site suitability requirements for such alternative methods of LLW disposal, (2) to provide guidance to applicants on acceptable methods of demonstrating site suitability, and (3) to develop means for staff evaluation of site suitability.

13.2.2 Research Program Description

The strategy for research is to use field data and laboratory experiments to understand the phenomena that determine the performance of LLW disposal facilities. This will be useful in providing guidance to disposers of LLW and 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, which is particularly relevant to 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 chemical 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.

Major research products will include:

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

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

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

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

- Coupled geochemical/hydrological transport models for predicting subsurface migration of radionuclides from shallow-land burial facilities (1985).

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

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

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

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

13.3 Uranium Recovery

Uranium recovery operations involve the extraction of uranium from ores by both conventional and in situ solution mining techniques. These operations result in the generation of large quantities of liquid and solid wastes that contain naturally occurring radionuclides and toxic chemicals. These wastes must be disposed of in compliance with regulations established by NRC and EPA (10 CFR Parts 20 and 40 and 40 CFR Parts 190 and 192, respectively). In licensing uranium recovery operations, the NRC must assess whether an applicant's proposed treatment and disposal of such wastes will comply with the above regulations. NRC's assessment requires the technical capability to analyze relevant phenomena and processes in a manner that minimizes uncertainties that might lead to excessive and unreasonable costs and restrictions while at the same time ensuring adequate protection of the public health and safety and the environment.

13.3.1 Major Regulatory Needs and Their Justifications

1. Criteria for evaluating the long-term stability of waste tailings retention systems following the closing of mills and the decommissioning of mill sites, to be used as a basis for updating a regulatory guide (FY 1985). Justification: NRC and EPA regulations (10 CFR Part 40 and 40 CFR Part 192, respectively) require waste tailings generated from uranium recovery operations to be stabilized for long periods of time following milling operations. Present engineering capabilities do not include methods for predicting the long-term stability of riprapped earthen structures without significant maintenance. This research is directed toward evaluating engineering designs that will require a minimal amount of long-term maintenance.

2. Criteria for better evaluating the seepage and underground migration of waste liquids from conventional uranium milling operations and the liquid leaching solutions used during in situ mining activities, to be used as

bases for regulatory guides on methods, data, assumptions, and models for predicting ground-water transport of toxic or radioactive materials (FY 1983) and on methods and requirements for designing, constructing, testing, monitoring, and closing wells and restoring well fields at in situ uranium solution mining sites (FY 1984).
<u>Justification</u>: The seepage and migration of such liquids and solutions must be minimized and controlled so as to meet the requirements in NRC and EPA regulations (10 CFR Part 20 and 40 CFR Part 190, respectively). The environmental impact from such seepage and migration must also be analyzed in accordance with NEPA requirements and EPA water-quality standards.

3. Criteria for evaluating more accurately the quantities, distribution, and impacts of airborne effluents from uranium recovery operations and methods and cost effectiveness of minimizing such effluents, to be used as a basis for regulatory guides. [These guides are on interim methods for stabilizing tailings piles (FY 1984) and on atmospheric dispersion data acquisition, reduction analysis, reporting requirements, and predictions of releases for tailings piles and other recovery operations (FY 1985).]
<u>Justification</u>: NRC regulations require airborne effluents from active uranium recovery operations to meet the requirements of 10 CFR Part 20 and EPA's 40 CFR Part 190. Such releases must also be evaluated as to quantity and environmental impact in accordance with the provisions of NEPA.

4. Criteria for more efficiently and cost effectively measuring residual concentrations of radioactive constituents to permit reclamation and decommissioning of uranium recovery sites, to be used as a basis for a regulatory guide (FY 1985).
<u>Justification</u>: NRC regulations require compliance with environmental cleanup standards to be contained in EPA's 40 CFR Part 192.

13.3.2 <u>Research Program Description</u>

The strategy for the research is to reduce the uncertainties in methods and models currently used in regulatory assessments to analyze (1) the long-term stability of reclaimed uranium tailings piles, (2) seepage from active tailings systems, (3) migration of liquids used for in situ solution mining, and (4) environmental impacts from airborne radioactive effluents from uranium recovery operations. This effort is aimed at developing technical information, testing analytical methods, and developing predictive models and engineering designs that can be used to expedite and improve the efficiency of licensing reviews and assessments and to reduce unnecessary conservatisms. Thus, more cost-effective licensing requirements and better environmental impact assessments relating to uranium recovery operations are provided.

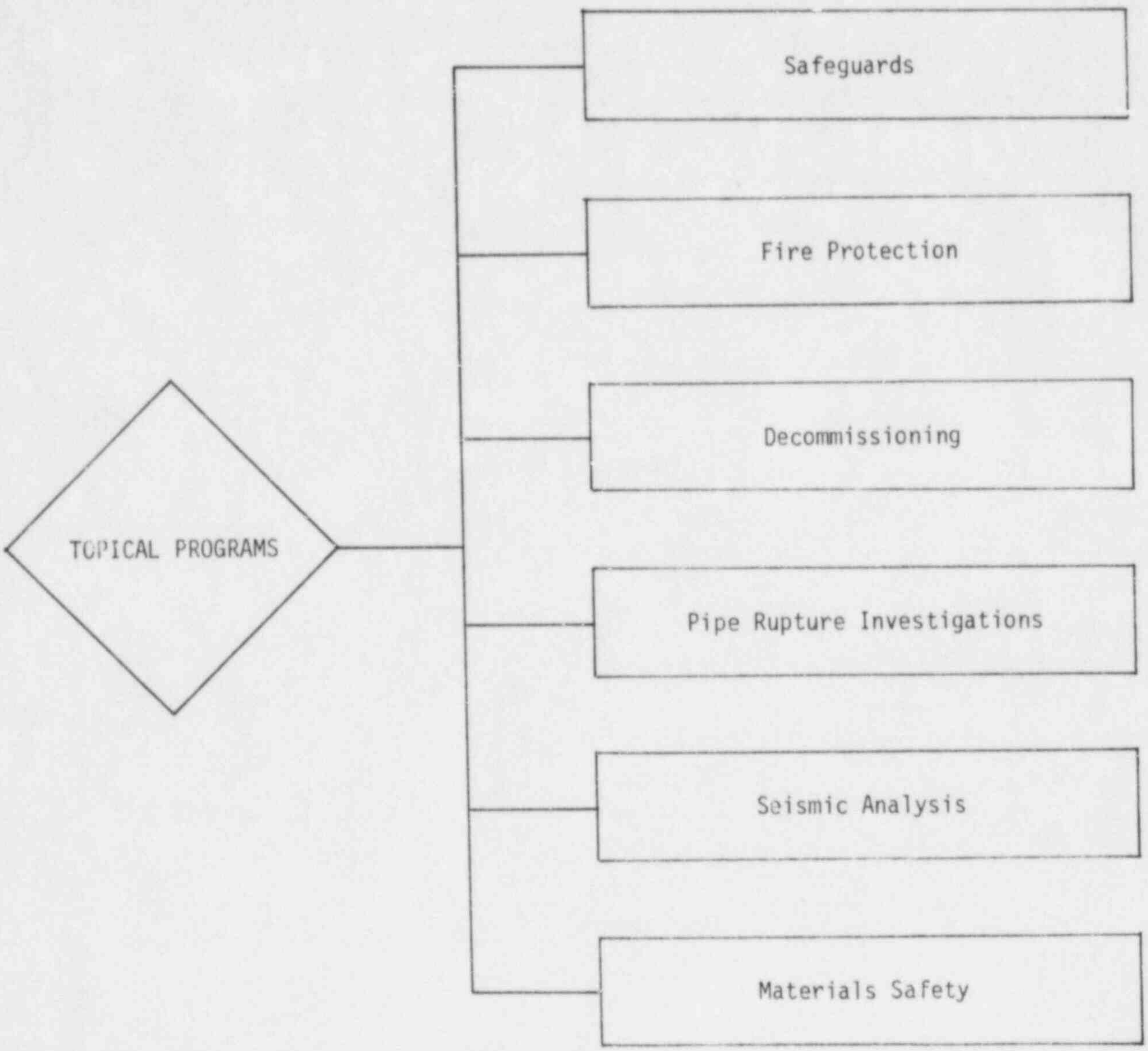
This research program develops methods for analyzing the design and application of riprap for the long-term mitigation of wind and water erosion of the covers of stabilized uranium mill tailings piles. In addition, methods and models are being developed to predict the underground migration and impact of seepage from both conventional tailings disposal operations and in situ solution mining operations. Finally, improved monitoring methods and equipment for assessing releases of radioactive and toxic materials from uranium recovery operations are being developed and tested. These allow the establishment of detailed guidance on monitoring requirements and strategies for evaluating compliance with EPA and

NRC standards. Most of the uranium recovery research is scheduled to be completed by 1988 unless unforeseen needs or events develop in the meantime.

The major research products will include:

- Engineering guidelines for installation of riprap covers (1984).
- Measurement techniques and remedial action in accordance with the Uranium Mill Tailings Radiation Control Act (1984).
- Sampling techniques for detecting contamination of surface and ground waters by tailings leachate (1985).
- Field studies of interim stabilization techniques (1985).
- Development and verification of models for predicting the subsurface migration of liquid wastes from uranium recovery operations (1988).
- Guidance on restoration of in situ leach uranium mines (1985).
- Determination of long-term durability of candidate rock for stabilizing tailings piles (1985).
- Improvements in measurement techniques and equipment for evaluating radon levels in structures (1985).
- Engineering designs and techniques for uranium mill tailings dewatering (1985).

Topical Programs



FUNDING LEVELS
(Dollars in Millions)

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
SAFEGUARDS	$1.0	$1.0	$1.0	$1.0	$1.0
DECOMMISSIONING	0.8	1.0	1.2	1.2	1.0
MATERIALS SAFETY	2.5	3.3	6.4	6.2	5.8

14. TOPICAL PROGRAMS

14.1 Safeguards

A comprehensive program of safeguards research that was initiated in FY 1976 has been essentially completed. The results of this research are being incorporated into proposed rule changes on material control and accounting and access authorization, which are expected to be published for comment in FY 1983. Future research will be limited and will stress the safety-safeguards interface of physical protection systems, human factors, and the effectiveness of alternative or improved physical security systems and procedures.

The major research products will be:

- Refinement of techniques for determining vital equipment (1983-1986).
- Identification and elimination of possible safety/safeguards interface problems at power reactors (1984).
- Identification of human factor concerns in safeguards problems (1983-1984).

14.2 Fire Protection

The fire protection research program will provide experimental and analytical information to evaluate the margins of safety inherent in the current fire protection criteria and to resolve specific issues raised as a result of the staff's experience to date in performing fire protection reviews.

Evaluation of the risks from fires at nuclear plants is difficult because adequate data are not readily available to allow quantification of the probabilities and damage associated with fires without large uncertainties. This program will provide data on the degree of fire damage expected from fires of various postulated sizes and configurations.

The major products of the fire protection research program will provide information in the following areas:

- Compilation of available fire test data potentially applicable to NRC fire protection programs (1985).
- Experimental data to evaluate the margins of safety provided for safe shutdown components and systems by selected combinations of various fire protection features such as spatial separation, fire retardant coatings, fire resistant barriers, and automatic fire suppression systems (1986).
- Analytical fire model capability for extrapolation to configurations of equipment and protective features different from those tested but

found in operating plants or in proposed plants presently undergoing licensing review (1987).

- Data to evaluate fire risk at nuclear plants (1988).

14.3 Decommissioning

The decommissioning program will develop information on technology, safety, and costs needed to establish regulations governing decommissioning of nuclear facilities, to provide bases for estimating financial requirements, and to establish criteria for the design and operation of these facilities that will facilitate their eventual decommissioning, thereby ensuring that the public health and safety is protected. This program covers the decommissioning of power and research reactors and of fuel cycle and non-fuel-cycle nuclear facilities.

The major research products from this element will be:

- Reports on actual decommissioning of reactors to be used in updating the technology, safety, and cost data bases of regulations and regulatory guides (1988).

- Reports on methodologies for surveys of residual radioactivity (1987).

- Updates of technology, safety, and costs for power reactors (1984, 1985, 1987); multireactor sites (1987); and research and test reactors (1988).

14.4 Pipe Rupture Investigations

The first effort in this research deals with establishing or revising criteria for specifying in the design process the nature, number, and locations of postulated breaks in nuclear power plant safety-related piping. Additionally, in a second effort, information is being obtained regarding the behavior of whipping pipes on adjacent pipes both from the experimental and theoretical point of view. The plan for this effort is to establish a sufficient basis so that licensing decisions can be made concerning requirements for pipe whip restraints and asymmetric LOCA and load combinations for primary system piping, components, and supports. The information for the first effort is being developed vendor by vendor with initial cooperation and contribution from Westinghouse and their owners group. The plan for the second effort is being directed toward validating acceptance criteria that are used for evaluating pipe impact resulting from pipe rupture.

The major products from this research element will be:

- Improved understanding of the reliability of Westinghouse primary system piping and the need to:

  - Require pipe whip restraints near such piping,
  - Postulate asymmetric LOCA for such piping in the safety assessment, and

- Combine pipe rupture with earthquakes in the design basis (1983).

- Same as above for Combustion Engineering systems (mid-1984).

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

- Experimental confirmation of how large pipes affect small pipes upon impact as a result of whipping, how small pipes affect large pipes upon impact, and how pipes of equal size behave during impact. In addition, the role of impact velocity on damage will be ascertained (mid-1985).

14.5 Seismic Analysis

This program deals with projects being carried out to better define the response and resistance of nuclear power plant structures and equipment to seismic events, including possible earthquakes beyond the design basis. The strategy is to direct testing, analysis, and data-gathering efforts both toward the individual objectives of each project and the interrelated needs of other seismic analysis research. For example, testing of Category I structures can provide both structural fragility data and nonlinear load input data to be used in piping research. Similarly, research on damping will supply information to be used in benchmarking computer codes and evaluating stiff versus flexible piping.

Priorities for seismic analysis research will be set using the results of the Seismic Safety Margins Research Program (SSMRP) and other risk studies. Efforts will be concentrated on those areas identified as being major contributors to risk and uncertainty. Seismic risk sensitivity studies will provide justification for the need to change design criteria.

Ongoing attempts will be made to collect and analyze test and earthquake experience data from outside groups (EPRI, foreign governments). These data are needed to reduce the large uncertainties currently associated with seismic response parameters and fragilities.

The major research products will be:

- Simplified seismic risk methodology applicable to both PWRs and BWRs (1984).

- Recommendations of alternatives to use of peak ground acceleration as input parameter (1984).

- Determination of changes in floor response spectra for accelerations above Safe Shutdown Earthquake level (1985).

- Benchmarking of soil-structure interaction and structural response analysis techniques (1985).

- Benchmarking of computer codes for buckling analyses of steel containments (1985).

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

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

- Benchmarking of mechanical component computer codes (1987).

- Recommendations for structural load combination criteria (1987).

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

- Quantitative assessment of contribution of design and construction errors to seismic risk (1988).

14.6 Materials Safety

This program deals with projects being carried out to support the regulation of activities involving the processing, transportation, interim storage, and end uses of radioactive materials in facilities other than nuclear power plants. This research encompasses fuel cycle, transportation, and radioisotope utilization.

The major research products from the materials safety program are:

- Risk assessment methods and codes development for fuel cycle (1984-1986).

- Application of risk assessment techniques in decisionmaking on present fuel cycles (1985-1987) and on advanced fuel cycles (1988).

- Accident analysis handbook for explosions, tornadoes, equipment failures, and accidental criticality in LWR fuel cycle facilities (1985).

- Analyses of fuel pellets and cladding of PWR and BWR fuel elements during 5 years of dry storage (1988).

- Verification of calculational models for fixed neutron poison in shipping or storage arrays (1985) and for variation in multiplication factor (1986).

- Updated version of fuel burnup and depletion code ORIGEN-S to determine spent fuel inventories and afterheat values (1985).

- Modal study of transportation safety (air, water, rail, road) (1984).

- Reports detailing the performance of generic cask designs during the tests representing severe accidents (1984-1987).

- Methods for determining temperature distributions in spent fuel assemblies (1986-1987).

- Shipping cask component wear studies (1986-1988).
- Draft environmental impact assessment (1985) and final environmental impact assessment (1986) of recycle of decontaminated materials.

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	5.2
A-14	Nondestructive Examination	2.5
A-17	System Interactions (The research involves human aspects of systems interactions.)	9.1
A-45	Shutdown Decay Heat Removal Requirements	5.2
A-47	Safety Implications of Control Systems	5.2 10.1
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	5.5
A-49	Pressurized Thermal Shock	2.1 Chapter 3 9.1 10.1

TMI Action Plan Item No.		
I.A.4.2	Long-Term Training Simulator Upgrade	9.2
I.B.1	Management for Operations	9.3
I.C.9	Long-Term Plan for Upgrading Procedures	9.4
II.B.8	Rulemaking on Degraded-Core Accidents	8.2 5.11 5.12 5.13

TMI Action Plan Item No.	Subject	LRRP Section
II.C	Reliability Engineering and Risk Assessment	9.5
IV.C	Extending Lessons Learned to Licensed Activities Other Than Power Reactors	9.2 9.4

Appendix B

SETTING PRIORITIES FOR RESEARCH PROGRAM

This appendix describes the methodology and its proposed implementation currently being developed for use by the Office of Nuclear Regulatory Research in setting priorities for research. The results of the initial efforts are also discussed.

1. METHODOLOGY

The method employed to generate priority ranks, the Analytic Hierarchy Process (AHP), was developed by T. L. Saaty (Refs. 1 and 2). 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 example hierarchy is shown in Figure B-1. 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 two criteria for Level 1: C_1 and C_2. These could be regulatory significance, risk/uncertainty relevance, cost effectiveness, etc. Level 2 contains such research areas as reactor risk and waste management. Notice that these research areas appear under both of the Level 1 entities, C_1 and C_2. Under each entity of Level 2, Level 3 may be developed with as many areas as needed and so on until the level that contains the decision units or specific programs and tasks on which decisions are needed is reached. Developing the hierarchy is the critical part of the method.

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 A_i, the matrix would have the form:

$$\begin{array}{c|c} & A_1 \quad A_2 \quad \cdots \quad A_k \\ \hline \begin{array}{c} A_1 \\ A_2 \\ \vdots \\ A_k \end{array} & \left[\begin{array}{cccc} & & & \\ & & & \\ & & & \\ & & & \end{array} \right] \end{array}$$

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 have been made, the "normalized" eigenvector corresponding to the maximum eigenvalue (λmax) yields the weights that we desire. It can be shown theoretically that $\lambda max \geq k$ where k is the order of the matrix. The quantity $CI = (\lambda max - k)/(k - 1)$ provides a check on the consistency of the rankings. The closer CI is to zero, the more consistent the rankings.

The overall priority of an entity in a lower level of the hierarchy depends on the priority of those entities in higher levels with which it is associated. The overall priority is synthesized from these pertinent priorities.

2. IMPLEMENTATION

The AHP is being applied to RES project objectives and to projects by the RES Division Directors under the direction of the Office Director and Deputy Director. The extension of the method to program elements is being performed by the respective program area managers under the direction of the Division Directors.

2.1 Criteria

The first step is to define the criteria to be used in setting priorities for the research program. These criteria, which focus on specific characteristics or attributes of a research project, should be measurable, at least on a relative scale, and should be unambiguous and understandable by others. A general criterion may be broken down into more specific criteria. Each specific criterion (or subcriterion) focuses on a specific attribute of a research project. The specific criteria should be as independent as possible to preclude double-counting project characteristics.

The AHP is useful for resolving a general criterion into more specific criteria. The hierarchy can be developed to whatever level is deemed necessary, and the priorities of the specific criteria can be set by the AHP.

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. Regulatory significance 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.

These general criteria can be broken down into more specific criteria. For such breakdowns, an AHP hierarchy can be constructed for communication purposes. A straightforward scheme is to assign each specific criterion (subcriterion) equal priority. Another is to set priorities for each subcriterion as discussed in the previous section. Even if the subcriteria are not actually used, they are helpful in understanding the meaning of the more general criteria.

2.2 Research Areas

Eighteen research areas were identified. These areas correspond to the chapter headings and the topical programs of the FY 1984-1988 Long-Range Research Plan. Obviously, a thorough understanding of the content and utility of the program areas is essential.

To facilitate the activity, a worksheet was developed. A portion of the worksheet used in shown in Figure B-2. Each Division Director was asked to fill in the worksheet using 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 by a computer program, the weights of the research programs within each criterion were calculated, and the overall weights were determined. 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.

3. REFERENCES

1. Saaty, T. L., The Analytic Hierarchy Process, McGraw-Hill, Inc., New York, 1980.

2. Saaty, T. L., Decision Making for Leaders, Lifetime Learning Publications, Belmont, California, 1982.

Figure B-1

AN EXAMPLE HIERARCHY

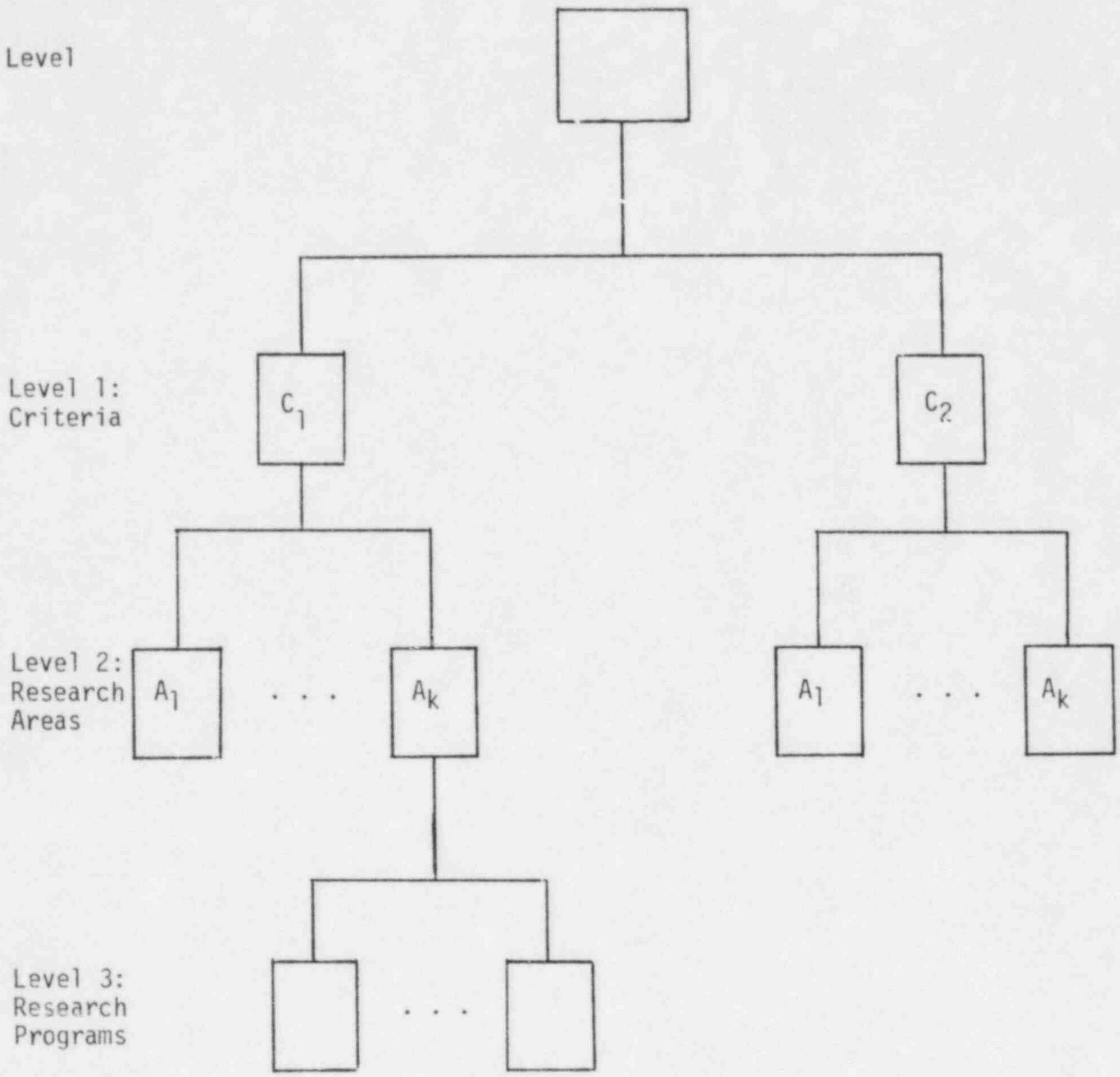


Table B-1

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 INITIAL SETTING OF PRIORITIES

Research Area	Regulatory Significance		Risk Relevance		Composite	
	Weight	Rank	Weight	Rank	Weight	Rank
1. Plant Aging	.0573	8	.0764	5	.0668	5
2. Pressurized Thermal Shock	.1455	2	.1386	2	.1420	2
3. Equipment Qualification	.0620	6	.0702	6	.0661	6
4. Severe Accident	.1745	1	.1578	1	.1661	1
5. Thermal-Hydraulic Transients	.0497	10	.0537	8	.0517	9
6. Advanced Reactors	.0128	18	.0134	17	.0131	18
7. Risk Analysis	.0956	3	.1016	3	.0986	3
8. Human Factors	.0743	4	.0882	4	.0812	4
9. Instrumentation and Control	.0249	12	.0362	10	.0305	12
10. External Events	.0514	9	.0658	7	.0586	7
11. Radiation Protection and Health Effects	.0222	14	.0176	14	.0199	14
12. Waste Management	.0638	5	.0230	13	.0434	10
13. Safeguards	.0138	17	.0158	15	.0148	17
14. Fire Protection	.0300	11	.0344	11	.0322	11
15. Decommissioning	.0199	15	.0133	18	.0166	16
16. Pipe Rupture Investigations	.0234	13	.0278	12	.0256	13
17. Seismic Analysis	.0600	7	.0514	9	.0557	8
18. Materials Safety	.0188	16	.0150	16	.0169	15

Figure B-2

WORKSHEET

Criterion for Comparison: ____________________

Pair No.	Areas to be compared (A vs. B): Area A	Area B	Compare the two areas A and B for the above criterion: Check (√) the more important area: A	B	Numerical value of comparative importance*
1.	Plant Aging	Pressurized Thermal Shock			
2.		Equipment Qualification			
3.		Severe Accident			
4.		Thermal-Hydraulic Transients			
5.		Advanced Reactors			
6.		Risk Analysis			
7.		Human Factors			
8.		Instrumentation and Control			
9.		External Events			
10.		Radiation Protection and Health Effects			
11.		Waste Management			
12.		Safeguards			
13.		Fire Protection			
14.		Decommissioning			
15.		Pipe Rupture Investigations			
16.		Seismic Analysis			
17.		Materials Safety			

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

Appendix C

RESEARCH PROGRAM OUTLINE

Research Program

Chapter 2
Plant Aging

CHAPTER 2. PLANT AGING

This research program will study time-related issues such as aging and degradation of nuclear power plant components. Examination and testing methods to determine the condition of components and interpretation of the test results are also included. This work will provide the bases for the staff to assess industry testing and examination methods and to make licensing decisions on whether operating plants meet appropriate health and safety requirements.

Research Elements

Reactor Vessels (2.1)

Structural integrity of pressure vessels especially as affected by irradiation embrittlement and growth of assumed cracks in service.

Steam Generators (2.2)

Degradation of steam generator tubing and sleeved tubing during service.

Piping (2.3)

Integrity of piping during service as degraded by the water, stress, and temperature environment.

Electrical and Mechanical Components (2.4)

Degradation of electrical and mechanical components during service.

Nondestructive Examination (2.5)

Validation of reliable reproducible nondestructive examination techniques for detection and characterization of cracks and flaws for pressure vessels, piping, and steam generator tubing.

Typical Products

- Basis for fracture toughness requirements under conditions of thermal shock to reactor vessels
- Validation of unified fracture mechanics methodology for licensing evaluation of pressurized thermal shock
- Licensing criteria and standards for in situ annealing of commercial reactor vessels

- Validation of current and advanced NDE results by examination of tubes removed from retired steam generators
- Correlation of residual tube integrity with NDE to validate inservice inspection and tube plugging criteria
- Evaluation of generator cleaning and decontamination methods

- Data base on stress-corrosion cracking in piping for validating predictive models and evaluating proposed fixes
- Data base on loss of fracture toughness in LWR stainless steel piping materials due to long-time aging at temperature
- Data from experiments on flaw size, loading, and material toughness of full-size pipe, including test to failure of large pipes that cracked in service

- Initial scoping study
- Validated criteria on acceptable methodologies to predict and prevent or mitigate significant age-related failures of components

- Validated data base on continuous monitoring for cracks by acoustic emission (AE) and validation of leak monitoring by AE
- A data base from parametric study and field validation trials to develop unified inspection requirements for vessels and piping
- A statistical analysis to determine the number of steam generator tubes needing inspection versus the type and extent of flaws in tubes

Research Program

Chapter 3
Pressurized Thermal Shock

CHAPTER 3. PRESSURIZED THERMAL SHOCK

The safety issue of concern in this research program is the susceptibility of certain older reactor vessels to overcooling transients. A probabilistic analysis will improve the understanding of the likelihood of vessel failure and how this likelihood changes as plants get older. The program is intended to determine licensing requirements through studies in the areas of metallurgy and fracture mechanics, thermal-hydraulic analysis, and risk analysis.

Typical Products

- Probabilistic analysis of likelihood of vessel failure due to PTS at representative plants designed by B&W (Oconee 1), CE (Calvert Cliffs 1), and Westinghouse (H. B. Robinson 2)
- Estimate of the consequences and risk to the public as a result of PTS
- Followup calculations to determine effects of new information, i.e., longer-range results from research in other areas such as human factors and safety implications of control systems

Research Program

Chapter 4
Equipment Qualification

CHAPTER 4. EQUIPMENT QUALIFICATION

This program will study methods for qualifying equipment used in nuclear power plants. Included in this program are such factors as the effects of synergism, order or sequence of tests, and accelerated aging techniques. To provide a basis for licensing decisions, methods for environmental and dynamic qualification of equipment will be validated and new methods will be developed, as needed.

Research Elements

Qualification of Electrical Equipment (Environmental and Functional) (4.1)

Assessment of methods for qualifying safety-related electrical equipment to demonstrate its ability to function both during and following design basis accidents.

Qualification of Mechanical Equipment (Environmental) (4.2)

Technical basis for developing environmental requirements for functional qualification of active mechanical components except for consideration of dynamic (including seismic) loads.

Dynamic Qualification of Equipment (4.3)

Technical basis for developing the dynamic (including seismic) qualification requirements for electrical and active mechanical equipment.

Typical Products

(4.1)

- Evaluation of the effects of aging, radiation dose rate, synergism, and steam exposure on polymers
- Evaluation of aging mechanisms for electrical equipment and establishing methodologies for accelerated aging
- Validated methods for accelerated aging and accident simulation by examining and testing components removed from nuclear plants

(4.2)

- Identification of the range of environmental loads to be evaluated
- Identification of the significant loads or combinations of loads
- Evaluation of criteria to extrapolate test results from one size of component to another

(4.3)

- Identification of the range of dynamic loads to be evaluated
- Identification of the significant loads or combinations of loads
- Evaluation of criteria to extrapolate test results from one size of component to another

Research Program

Chapter 5
Severe Accidents

CHAPTER 5. SEVERE ACCIDENTS

This research program supports the reassessment of the regulatory treatment of severe accidents in nuclear power plants. It includes coordinated phenomenological and risk assessment research to develop a sound technical basis for NRC decisions concerning the ability of reactors to cope with accidents beyond those specified in 10 CFR Parts 50 and 100.

Research Elements

Accident Analysis

Accident Likelihood Evaluation (5.1)
Severe Accident Sequence Analysis (5.2)
Accident Management (5.3)

Fuel & Fission Products

Behavior of Damaged Fuel (5.4)
Fuel Structure Interaction (5.6)
Fission Product Release & Transport (5.8)
Fission Product Control (5.10)

Containment

Containment Analysis (5.7)
Containment Failure Mode (5.9)
Hydrogen Generation and Control (5.5)

Risk

Risk Code Development (5.11)
Accident Consequences & Risk Reevaluation (5.12)
Risk Reduction & Cost Analysis (5.13)

Typical Products

Accident Analysis

- Operating procedures to ensure containment integrity during severe accidents
- Operator procedures for recovery from potentially severe accidents
- Plant response characteristics to selected multiple failure sequences
- Instrumentation to prevent ambiguous information during transients
- Assessment of procedural and design changes to mitigate consequences of an ATWS
- Evaluations of operating data and vendor information for component and system reliability
- Identification and review of accident sequences, including accident precursors

Fuel & Fission Products

- Report on risk-dominant phenomenological uncertainties in severe LWR accidents
- SCDAP-Mod 2 with whole-core analysis capability
- Independent examination of selected TMI-2 core samples
- Verification of fuel-melt/concrete and fuel-melt/retention materials interaction models in CORCON
- Modeling the interaction of hot solidified melts with concrete and long-term cooling of solidified core melts
- Models for the explosive and nonexplosive interaction of core melts with water in FITS facility
- Revised severe accident source terms to reassess current source term
- Large-scale (Marviken) fission product and aerosol tests to assess TRAP-MELT code
- Data for benchmarking fission product release models from irradiated LWR rods
- Data for benchmarking models for release of fission products from experimental fuel tests
- Code for the performance and effectiveness of LWR filtration systems under predicted aerosol loadings
- Code to evaluate generic design of ESF systems for standardized nuclear facilities
- Evaluation of the performance and scrubbing efficiency of suppression pools
- Validated codes for evaluating ESF reliability and aging characteristics
- Validated code for evaluating the effectiveness of the PWR ice condenser

Containment

- Interface between CONTAIN code and TRAP-MELT to provide fission product and energy source data
- Interface between CONTAIN and fission product dispersion code to compute offsite doses
- Capacity predictions for concrete containments and steel containments under dynamic pressure loads
- Initial tests of containment models under simulated seismic loading
- Initial tests on models of major penetrations
- Improved hydrogen combustion mitigation systems
- Effects of aerosols on hydrogen control system

Risk

- MARCH 2, MATADOR, and MELCOR codes documented for users
- Consequence and risk evaluations at roughly 1-year intervals
- Design concepts such as alternative decay heat removal, filtered-vented containment, stronger containments, and core catchers
- Cost analyses of above concepts (and others available to owners)
- Risk-reduction analyses

Research Program

Chapter 6
Thermal-Hydraulic Transients

CHAPTER 6. THERMAL-HYDRAULIC TRANSIENTS

This research program includes the methods and data that provide the bases for reactor coolant system analyses. These analyses are used to determine the adequacy of reactor operator guidelines and procedures as well as to analyze complex plant transients. They are also used to quantify the margins that exist between the actual fuel temperatures that would be experienced in a loss-of-coolant accident and those that would be calculated using the ECCS rule (10 CFR Part 50, Appendix K).

Research Elements

Separate Effects Experiments and Model Development (6.1)

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 during degraded core cooling and other plant transients.

Integral Systems Experiments (6.2)

Experimental simulations of integral thermal-hydraulic systems of PWR and BWR reactors. Transients simulated include the full-break-size spectrum of LOCAs, loss of feedwater, steam line break, steam generator tube rupture, and various safety and control system failures.

2D/3D Program (6.3)

A joint research program with the FRG Ministry of Research and Technology (BMFT) and JAERI to study the thermal-hydraulic behavior of the emergency core coolant during the refill and reflood phases of a large-break LOCA in a PWR.

Code Assessment and Application (6.4)

Application of computer codes to the analysis of transients in full-scale LWRs and the assessment of these analytic capabilities against experimental data.

Plant Analyzer and Data Bank (Includes Code Improvement and Maintenance) (6.5)

Improvement and maintenance of the computer codes discussed in Section 6.4, as well as the development of an automated plant analyzer with plant-specific output displays.

Typical Products

- Bundle post-CHF (critical heat flux) data analysis report
- Post-CHF heat transfer model
- Thermal fluid mixing models

- Assessment of safety and licensing issues using LOFT results
- Evaluation of BWR transients using FIST Phase I and Phase II data
- Assessment of advanced computer codes using LOFT thermal-hydraulic test data

- Examination of Appendix K models (small-scale data) for applicability to large-scale data
- Effect of steam binding during reflood phase related to Appendix K requirements
- Extent of emergency core cooling bypass during refill phase as required by Appendix K

- Use of plant analyzer and plant data bank to analyze transients in full-scale LWRs
- Final version of advanced multidimensional two-fluid transient analysis code
- Systems code for analysis of LWR behavior under degraded core conditions by incorporating multirod fuel code for analyzing severe core damage

- First PWR and BWR system plant analyzers for use by NRC staff
- Demonstration of prototype PWR plant analyzer with plant data bank

Research Program	Research Elements	Typical Products
Chapter 7 Advanced Reactors		
CHAPTER 7. ADVANCED REACTORS This program includes safety research needed to support NRC regulatory activities for all advanced types of nuclear power reactors. It includes research for the liquid-metal-cooled fast-breeder reactors and the high-temperature gas-cooled reactors.	Fast-Breeder Reactors (7.1) Data and analytical methods to make licensing decisions on LMFBRs.	• Analysis of CRBR thermal hydraulics and design basis accidents • Analysis of CRBR energetics with SIMMER code for operating license • Data and models on clad and fuel relocation during initiation phase of core disruptive accident • Data and models to assess long-term coolability of ex-vessel debris in sodium
	Gas-Cooled Reactors (7.2) Develop or verify the chemical, metallurgical, structural, and system performance data and methods necessary to allow the NRC to license HTGRs.	• Advanced analysis code for HTGR fission product plateout and liftoff • New HTGR edition of standard format and content of SARs • Thermal barrier and liner cooling system requirements

Research Program

Chapter 8
Risk Analysis

CHAPTER 8. RISK ANALYSIS

This research program supports the application of probabilistic risk assessment methods to the regulation of nuclear power reactors. This program includes the development of models, methods, documented procedures, and other analyses to support decisions on many reactor safety issues.

Research Elements

Risk Assessment Methods Development (8.1)

Develop and document methods for quantifying the probabilities and consequences of severe reactor accidents and reducing the uncertainties in such estimates.

Methods Development for Risk Reduction (8.2)

Systematic evaluations of the cost effectiveness of current or proposed regulatory requirements, alternative concepts for reactor design and operation, and decisions on backfitting.

Reliability Assurance Program (8.3)

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.

Typical Products

- Data base from evaluations of plant operating data, LER reports, and vendor information for component/system reliability studies
- Identification and review of accident sequences, including accident precursors
- Procedures for incorporating results of the seismic safety margin research program into PRA methodology
- Revised accident consequence codes reflecting the most recent data and more realistic phenomena

- Assessment of costs and risk-reduction of alternative LWR safety features
- Feasibility of use of PRA to improve reliability of plant systems

- Recommendations for cost-effective reliability assurance programs at nuclear power plants based on PRA
- Recommendations for operator training and procedures keyed to accident conditions
- Methods for providing risk insights in reviews of plant technical specifications and limiting conditions for operation
- Risk-based procedures for inspection and enforcement activities

Research Program

Chapter 9
Human Factors

CHAPTER 9. HUMAN FACTORS

This program includes research to provide the technical basis for current and anticipated regulatory needs in applying human factors engineering and emergency preparedness guidance at 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.

Research Elements

Human Factors Engineering (9.1)

Technical basis for evaluating the man-machine relationships at information and control stations and in control rooms, for assessing and recommending human factors standards and guidelines for new or improved designs affecting operators or maintenance personnel and for establishing criteria for regulatory applications of human factors engineering.

Licensee Personnel Qualifications (9.2)

Technical basis for assessing, developing, or confirming the criteria used by the NRC to establish and evaluate the qualifications of licensee personnel to safely operate a nuclear facility and reduce operator-related risk.

Plant Procedures (9.3)

Technical basis for the methods and criteria used by NRC to assess and upgrade, where needed, plant operating procedures necessary for safe operation of nuclear power plants and fuel cycle facilities.

Human Reliability (9.4)

Analysis of errors of nuclear power plant operations personnel and maintenance personnel and their contributions to man-machine safety system failures.

Emergency Preparedness (9.5)

Technical basis and standards for NRC regulatory actions needed to ensure the capability of Federal, State, and local governmental authorities and licensees to mitigate the consequences of an accident at a nuclear facility.

Typical Products

Human Factors Engineering (9.1):

- A comprehensive data base reflecting the operator and crew behaviors in a variety of plant evolutions and accident sequences
- Data base for developing requirements and criteria for alarm filtering systems, disturbance analysis systems, computerized procedures manuals, and artificial intelligence systems
- Guidelines for future human engineering standards and criteria for control room and display, control, and communication systems
- Evaluation of the effects on operational personnel of severe stress due to seismic events and similar sources of stress

Licensee Personnel Qualifications (9.2):

- Application of the systems approach to training (SAT) to criteria for selecting malfunctions that should be modeled in nuclear power plant training simulators
- Empirical data on nuclear power plant operator performance from training simulator experiments
- Training assessment methodology based on SAT for unlicensed operators and support personnel at nuclear power plants
- Training assessment methodology based on SAT for operators and support personnel at fuel cycle facilities

Plant Procedures (9.3):

- Methodologies for evaluating operating, maintenance, testing, and emergency operating procedures for LWRs
- Adaptation of computer-based analysis techniques for assessing techniques for presenting nuclear power plant procedures
- Criteria for applying and assessing alternative techniques and formats for presenting procedures
- Assessment of the impact of computer diagnostics and automation on procedures and regulatory requirements

Human Reliability (9.4):

- Handbook and related workbook (procedures manual) supporting human reliability analyses of safety-related events in the operation and maintenance of nuclear power plants
- Prototype human reliability data bank
- Validated dynamic computer simulation model for reliability studies on nuclear power plant maintenance personnel

Emergency Preparedness (9.5):

- Technical basis for emergency preparedness requirements for fuel cycle and material licensees
- Evaluations of protective action decisionmaking and emergency action level identification

Research Program

Chapter 10
Instrumentation and Control

CHAPTER 10. INSTRUMENTATION AND CONTROL

Research in this program involves improving and confirming the availability of reactor and associated process system protection, control, and instrumentation to minimize the

Research Elements

Safety Implications of Control Systems (10.1)

Technical basis for evaluating malfunctions of plant control and related instrumentation and electrical systems to determine the impact of these malfunctions on plant operational safety and equipment important to safety, particularly where such events could lead to unanticipated transients or accidents.

Component Assessment (10.2)

Determine activities needed to ensure that proper performance of I&C and electrical systems and equipment important to safety can be achieved in a nuclear power plant.

Diagnostic Equipment and Capability (10.3)

Technical basis for evaluating equipment for diagnosing problems in reactor systems to help prevent undesirable plant transients or accidents and help avoid damage to equipment important to safety.

New I&C Technology (10.4)

Evaluate technological advances in the state of the art of I&C and electrical equipment developed for other fields of technology and determine their applicability and acceptability for use in nuclear power plants.

Typical Products

Safety Implications of Control Systems (10.1):

- Resolution of Unresolved Safety Issue A-47, "Safety Implications of Control Systems"
- Guidance for setting priorities for alarms
- Criteria for use of digital computers in systems important to safety
- Guidance for design, qualification, and testing of instrument and control systems important to safety that are not considered "safety grade"

Component Assessment (10.2):

- Guidance for response time testing of pressure transducers with their sensing lines
- Guidance for protection against electromagnetic interference
- Guidance for individual instruments used to follow the course of an accident

Diagnostic Equipment and Capability (10.3):

- Diagnostic techniques using a.c. signal components
- Guidance on use of conventional analog and digital instruments for diagnostic purposes
- Guidance on design, qualification, and testing of diagnostic instrumentation
- Demonstration of an automated continuous on-line surveillance and diagnostics system at operating plants

New I&C Technology (10.4):

- Guidance on use of advanced instruments and techniques used in other industries

Research Program

Chapter 11
External Events

CHAPER 11. EXTERNAL EVENTS

Research in this program deals with uncertainties in characterizing extreme natural and man-related phenomena that pose a threat to the safe operation of nuclear facilities and their probabilities.

Research Elements

Man-Related Phenomena (11.1)

Effect on safe operation of explosion-generated missiles and toxic, corrosive, or explosive vapor clouds.

Natural Phenomena (11.2)

Reduced uncertainty in analysis of seismic, flood, and severe meteorological hazards.

Typical Products

- Effect on safe operation of explosion-generated missiles and toxic, corrosive, or explosive vapor clouds
- Atmospheric transport models for hazardous materials
- Equipment specifications for hazardous materials
- Operator protection procedures
- Mechanisms and consequences of operator incapacitation

- Reduced uncertainty in analysis of seismic, flood, and severe meteorological hazards
- Specification of seismic source zones in Eastern United States
- Information on attenuation of seismic energy
- Data for development of site-specific spectra
- Improvement of design basis tornado specifications

Research Program

Chapter 12
Radiation Protection and Health Effects

CHAPTER 12. RADIATION PROTECTION AND HEALTH EFFECTS

Research in this program will provide technical and scientific information necessary to evaluate (1) the health risks arising from various modes and degrees of radiation exposure and (2) the effectiveness of various strategies and means for radiation protection and control in the work place and the general environment.

Research Elements

Metabolism and Internal Dosimetry (12.1)

Methods for calculating internal doses utilizing improved data on the metabolic behavior of transuranics and of materials from the front end of the fuel cycle.

Health Effects and Risk Estimation (12.2)

Reduced uncertainties in risk from low- and high-LET radiation and models for incorporating early health effects into risk assessment.

Radionuclide Pathways for Radiation Exposure of Man (12.3)

Validated models of pathways for transport of radionuclides from point of release to point of human intake.

Occupational Radiation Protection (12.4)

Improved methods for measuring and controlling occupational exposure and evaluating the effectiveness of dose-reduction techniques.

Typical Products

Metabolism and Internal Dosimetry (12.1):

- Methods for calculating internal doses utilizing improved data on the metabolic behavior of transuranics and of materials from the front end of the fuel cycle
- Gastrointestinal absorption factors for transuranics
- Metabolic models for inhaled mixed oxides and yellowcake
- Computer codes for internal dosimetry

Health Effects and Risk Estimation (12.2):

- Reduced uncertainties in risk from low- and high-LET radiation and models for incorporating early health effects into risk assessment
- Evaluation and utilization of hematologic biodosimetry
- Exposure effectiveness factors for low-level radiation
- Effects of low-level chronic irradiation on spermatogenesis
- Revised values for neutron quality factor
- Effects of inhaling radon and its daughters

Radionuclide Pathways for Radiation Exposure of Man (12.3):

- Validated models of pathways for transport of radionuclides from point of release to point of human intake
- Improved concentration factors for fresh-water and marine ecosystems
- Radionuclide loss from surface soil due to leaching by rainwater
- Removal of radionuclides from ground surfaces by wind in non-arid areas

Occupational Radiation Protection (12.4):

- Improved methods for measuring and controlling occupational exposure and evaluating the effectiveness of dose-reduction techniques
- Improved techniques for radiation protection training
- Incentives for dose reduction
- Reduced collective dose at nuclear power plants
- Improved monitoring of worker exposure
- Improved techniques for respiratory protection

Research Program

Chapter 13
Waste Management

CHAPTER 13. WASTE MANAGEMENT

This research program involves three areas: high-level waste, low-level waste, and uranium recovery. The program is intended to develop the technical capability for assessing operational safety, occupational radiological protection, long-term waste isolation, and the risks associated with these waste management systems.

Research Elements

High-Level Waste (13.1)

Develop means for evaluating proposed safety and radiological health protection programs of a high-level waste repository, the impact of the repository on the environment, and the long-term releases from the repository.

Low-Level Waste (13.2)

Develop models of the phenomena and processes that affect the mobility of radionuclides in low-level wastes and establish methods and procedures to determine compliance with regulations.

Uranium Recovery (13.3)

Develop the technical capability to analyze phenomena and processes relevant to treatment and disposal of wastes resulting from uranium recovery operations to ensure adequate protection of the public health and safety and the environment.

Typical Products

High-Level Waste (13.1):

- Model for degradation of waste forms based on surface kinetics
- Method for predicting long-term performance of waste packages
- Transport model for assessing radionuclide movement through the geologic medium
- Techniques for evaluating effectiveness of borehole plugging or sealing and shaft sealing

Low-Level Waste (13.2):

- Evaluation of interactions of radionuclides with soils to improve predictions of facility performance
- Transport models for predicting subsurface migration of radionuclides from shallow-land burial facilities
- Methods to ensure trench cap stability
- Effect of organic complexants on ground-water transport

Uranium Recovery (13.3):

- Guidelines for installation of riprap covers
- Measurement techniques and remedial action guidelines (to comply with Uranium Mill Tailings Radiation Control Act)
- Techniques for detecting contamination of surface and ground waters by tailings leachate
- Assessment of interim stabilization techniques

Research Program

Chapter 14
Topical Programs

CHAPTER 14. TOPICAL PROGRAMS

These research programs include safeguards, fire protection, decommissioning, pipe rupture investigations, seismic analysis, and fuel cycle and transportation.

Research Elements

Safeguards (14.1)

Provide the technical basis to support current and anticipated regulatory needs in the application of safeguards to special nuclear materials and nuclear facilities.

Fire Protection (14.2)

Develop experimental and analytical information to evaluate the margins of safety inherent in the current fire protection criteria and resolve specific issues raised as a result of the staff's experience to date in performing fire protection reviews.

Decommissioning (14.3)

Develop information on technology, safety, and costs needed to establish regulations governing decommissioning of nuclear facilities and to provide bases for estimating financial requirements.

Pipe Rupture Investigations (14.4)

Establish or revise criteria for specifying in the design process the nature, number, and locations of postulated breaks in nuclear power plant safety-related piping.

Seismic Analysis (14.5)

Improve the knowledge of the response and resistance of nuclear power plant structures and equipment to seismic events, including possible earthquakes beyond the design basis.

Materials Safety (14.6)

Fuel cycle, transportation, and radioisotope utilization.

Typical Products

Safeguards (14.1):

- Refinement of techniques for determining vital equipment
- Identification and elimination of possible safety/safeguards interface problems at power reactors
- Identification of human factor concerns in safeguards problems

Fire Protection (14.2):

- Compilation of available fire test data potentially applicable to NRC fire protection programs
- Data needed to evaluate the margins of safety provided for safe shutdown components and systems by various fire protection features
- Analytical fire model capability for extrapolation configurations of equipment and protective features tested to those in operating plants or in plants undergoing licensing review
- Data needed to evaluate fire risk at nuclear plants

Decommissioning (14.3):

- Reports on actual decommissioning of reactors to be used in updating the technology, safety, and cost data bases of regulations and regulatory guides
- Reports on methodologies for surveys of residual radioactivity
- Updates of technology, safety, and costs for power reactors, multireactor sites, and research and test reactors

Pipe Rupture Investigations (14.4):

- Improved understanding of the reliability of Westinghouse and Combustion Engineering primary system piping
- Evaluation of small-break LOCA probabilities
- Experimental confirmation of the effect of the impact of one pipe on another as a result of whipping for pipes of the same size, a large pipe striking a small, and a small pipe striking a large

Seismic Analysis (14.5):

- Simplified seismic risk methodology applicable to both PWRs and BWRs
- Recommendations of alternatives to the use of peak ground acceleration as an input parameter
- Determination of the changes in floor response spectra for accelerations above the Safe Shutdown Earthquake level
- Benchmarking of soil-structure interaction and structural response analysis techniques

Materials Safety (14.6):

- Risk assessment methods and codes development for fuel cycle
- Analyses of fuel pellets and cladding of PWR and BWR fuel elements during 5 years of dry storage
- Modal study of transportation safety (air, water, rail, road)
- Reports detailing the performance of generic cask designs during tests representing severe accidents

Glossary

ACRONYMS AND INITIALISMS

ACRR	Annular Core Research Reactor
ACRS	Advisory Committee on Reactor Safeguards
AE	Acoustic emission
AGR	Advanced gas reactor
AHP	Analytic Hierarchy Process
ALARA	As low as is reasonably achievable
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
ATWS	Anticipated transient without scram
BMFT	Bundesminister für Forschung und Technologie (FRG Ministry for Research)
BOP	Balance of plant
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
CCFL	Countercurrent flow limitation
CDA	Core disruptive accident
CE	Combustion Engineering
CEA	Commissariat à l'Energie Atomique, France
CEC	Commission of the European Community
CHAP	HTGR system transient analysis code
CHF	Critical heat flux

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	Calculation of reactor accident consequence code
CRBR	Clinch River Breeder Reactor
CREARE	An engineering research and development company
DBA	Design basis accident
DOE	Department of Energy
ECCS	Emergency core cooling system
EMI	Electromagnetic interference
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	Emergency planning zone
ER	Environmental report
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
FLECHT-SEASET	Full-length emergency cooling heat transfer-separate effects and system effects tests
FMEA	Failure modes and effects analysis
FP	Fission product
FRG	Federal Republic of Germany
GRS	Gesellschaft für Reaktorsicherheit
HLW	High-level waste
HPI	High-pressure injection
HTGR	High-temperature gas-cooled reactor
I&C	Instrumentation and control
ICRP	International Commission on Radiological Protection
IDCOR	Industry Degraded Core (program)
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IREP	Interim Reliability Evaluation Program
ISA	Instrument Society of America
ISI	Inservice inspection
JAERI	Japanese Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LER	Licensee event report
LET	Linear energy transfer
LLW	Low-level waste

LMFBR	Liquid-metal-cooled fast-breeder reactor
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 (to include MARCH, CRAC-2, and MATADOR codes)
MELPROG	Melt progression code
ML	Manufacturing license
NDE	Nondestructive examination
NEPA	National Environmental Policy Act
NIH	National Institutes of Health
NPP	Nuclear power plant
NREP	National Reliability Evaluation Program
NRU	Natural-uranium, heavy-water moderated and cooled test reactor, Chalk River, Ontario
OECD	Organization for Economic Cooperation and Development
OL	Operating license
ORECA	HTGR system transient analysis code
ORIGEN-S	Code to determine fuel burnup and depletion
ORNL	Oak Ridge National Laboratory

PAG	Protective action guide
PBF	Power Burst Facility
PCRV	Prestressed concrete reactor vessel
PISC	Plate Inspection Steering Committee
PISC	Program for Inspection of Steel Components
PNL	Pacific Northwest Laboratory
PPG	Policy and planning guidance
PRA	Probabilistic risk assessment
PTS	Pressurized thermal shock
RELAP	Detailed model for thermal-hydraulic behavior in reactor coolant system during transient and loss-of-coolant accidents
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Applications Program
SAFT-UT	Synthetic aperature focusing technique for ultrasonic testing
SAR	Safety analysis report
SASA	Severe accident sequence analysis
SAT	Systems approach to training
SCDAP	Severe core damage analysis package
SEP	Systematic Evaluation Program
SFD	Severe fuel damage
SIMMER	Code to analyze course of core-melt accidents in LMFBRs
SPDS	Safety Parameter Display System
SSC	Super-system code to analyze system transients in LMFBRs

SSMRP	Seismic Safety Margin Research Program
SUVIUS	Code to solve behavior of fission gases in primary coolant of gas-cooled reactors
TRAC	Code to model core reflood and quenching
TRAC/COBRA	Code to analyze LOCA consequences in PWRs with upper-head injection
TRAP-MELT	Code to analyze fission product behavior within LWR primary system under accident conditions up to and including fuel meltdown
UPTF	Upper Plenum Test Facility
USI	Unresolved safety issue

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