NUREG-0740

Long Range Research Plan

FY 1983 - 1987

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1. INTRODUCTION

1.1 Background

To better coordinate NRC research planning with the budget cycles, the Commission directed the Office of Nuclear Regulatory Research (RES) (through a memorandum from the Office of the Secretary (Chilk) to the Acting Executive Director for Operations (Dircks), dated April 22, 1980) to develop a long-range research plan (LRRP). It was further felt that this plan would assist the Commission in establishing appropriate priorities and in ensuring effective utilization of NRC resources. The LRRP is a 5-year planning document that identifies regulatory issues and lays out programmatic approaches for research to be done as part of the resolution of these issues. The plan will be updated at least annually as tasks are completed and new requirements and guidance are provided by the Commission, either through their Policy, Planning, and Program Guidance (PPPG) documents or by direct comment on the LRRP.

The LRRP is being developed by RES with recommendations from the user offices (NRR, NMSS, SD, and I&E) on their current or perceived further research needs. In November 1980, a Draft LRRP was forwarded to the user offices for comments on those portions that related to their needs. In addition, the draft plan was forwarded to the Office of the Executive Director for Operations (EDO), the Office of Policy Evaluation (PE), and several other staff offices for their comments on the overall LRRP. In general, their recommendations have been incorporated into the LRRP. Any significant comments not incorporated because of differences between RES and the other program offices will be highlighted in the paper that forwards the LRRP to the Commission. Additionally, any items or comments on the LRRP that affect agency policy or that should be included in the PPPG will be noted in the forwarding paper to the Commission. The process by which the user offices provided input to the plan and subsequently reviewed and commented on it represents their general endorsement of the need for the research programs in their areas of interest.

After their initial LRRP review, the user offices again review the research program during the budget cycle when they specifically endorse programs in their related areas for the coming fiscal year. Thus, the sponsoring offices and RES participate together in the development and justification of the research budget. Because the program offices have endorsed their programs during the planning and budget development phase of the budget cycle, it will not be necessary for the user offices to again endorse the programs during the budget execution phase. The EDO has asked the Office of Management and Program Analysis (MPA) to develop explicit procedures for interoffice cooperation in the planning and budget review process.

The RES staff expects that the ACRS will review and comment on the plan each year and that the Commission will subsequently review the plan for approval. Based on comments from the Commission and guidance provided in the PPPG, specific projects and budget will be developed.

The resource requirements for the LRRP for FY 1983-1987 are based on the Office of Management and Budget (OMB) FY 1982-1983 mark, with an allowance for inflation in FY 1983. The budget and resource information are being forwarded to the Commission and program offices separately because of the predecisional nature of the information. The LRRP is specifically directed toward NRC regulatory activities and emphasizes the safety of operating reactors and other fuel cycle facilities. The plan attempts to provide a balance between work directed at NRC current or near-term regulatory activities and work in areas with potentially broader or longer-term results. For this first LRRP, only limited effort was directed at assigning priorities to research programs, especially between decision units. However, the limitations on budget, the research requests from user offices, and the ACRS comments on the budget have influenced to some extent the assigning of program priorities within a decision unit. As suggested by the ACRS, for example, RES is evaluating the use of risk-assessment techniques as an aid in the development of research priorities; however, it should be noted that this method would be only one of the means used to determine what programs are proposed for funding and at what level.

A contingency long-range program plan has been developed for a few research areas--fast breeder and gas-cooled reactors--primarily because past Congressional action has directed that work be conducted in these areas; however, these programs have not been included as a formal part of this LRRP because of budget constraints and the need for further congressional direction on these programs. The programs planned would be compatible with an expanding effort by the Department of Energy (DOE) in these areas and with a decision by the administration to go ahead with a fast-breeder demonstration plant.

It should be noted that the programs proposed in this LRRP are work that RES and other program offices (through their requests and endorsements) feel is appropriately sponsored by the NRC. NRC-sponsored research is aimed at developing a technical basis to support regulatory decisions, rulemakings, and the development of standards and at resolving generic safety issues. The nuclear industry and DOE also 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. There should be cooperation and coordination between the NRC, DOE, and the nuclear industry to ensure that the appropriate level of effort is directed at resolving safety issues and to prevent unnecessary duplication of effort. The RES technical staff attempts to stay abreast of work sponsored by the nuclear industry, DOE, and foreign organizations through meetings, discussions, and the exchange of information. In addition, the RES staff attempts to sponsor appropriate regulatory research that does not duplicate other efforts.

Recent legislation (Light Water Reactor Research Act-P.L. 96-567) requires DOE "to establish a safety, development and demonstration program for developing practical improvements in the generic safety of nuclear power plants during the next five years"; this legislation, in turn, will require the NRC and DOE to enter into an immediate dialogue on what new effort they, together with the nuclear industry, might undertake. These efforts will help to ensure a well-coordinated national program, precluding duplication and avoiding programmatic conflict with Commission reactor safety research programs. A probable outgrowth of these discussions will be the need for a new memorandum of understanding between the two agencies.

1.2 Long-Range Role of NRC Research

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 the public health and safety, protection of environmental quality, and assurance of national security--calls for the exercise of the regulatory functions of rulemaking, licensing review, and inspection and enforcement to ensure that licensed nuclear activities satisfy established safety and environmental standards.

The mission of the Office of Nuclear Regulatory Research (RES) within the NRC is to provide the research information needed as part of the basis for sound understanding of regulatory issues and establishing effective regulatory policies and practices for evaluating licensee proposals and activities. RES carries out this mission through the development of risk-assessment methods for evaluating regulatory issues and the application of those methods to broad problem areas; through improvement of the understanding of phenomena necessary for analyzing safety, safeguards, and environmental impact; and through identification and definition of means for improving the consistency and coherency of the level of protection afforded by NRC regulations.

In the process of carrying out its mission, the Commission formulates basic 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 has available complete information for this purpose. In many cases, the information and data needed must be obtained by research.

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 an accident such as the one at Three Mile

Island (TMI) as a basis for reassessing technical criteria with the goal of facilitating Commission actions to improve the regulatory process.

While the staff legitimately should ensure the completeness and rigor of regulatory assessments, the nature of the regulatory process tends to require that the staff give priority to review in light of accepted standards, that is to say, precedent generally rules unless challenged. Experience indicates that there are serious practical difficulties in expecting those who are called upon to make important decisions in specific cases to continually question the scope, character, and content of the process by which they make those decisions. But, in view of the insights gained from such operating experiences as the TMI accident, it is also necessary to maintain within the NRC an institutional means for independent assessment of the completeness and validity of the NRC regulatory process. In addition to providing technical support for other regulatory activities, the NRC research program supports this function.

NRC research objectives are broadly defined as follows:

1. Better understanding of regulatory issues

   Research on methods for assessing risk is used to identify and evaluate the principal contributors to risks, to estimate uncertainties, to assess risk implications of proposed actions, and to help ensure that NRC requirements are properly balanced and cost effective.

2. Improvement of the licensing process

   Research to reduce the uncertainties in and to increase the clarity of regulatory assessments (that is, to make the relationship between licensee performance and risk more explicit) helps to make licensing decisions more understandable and, thus, more acceptable to the licensee, the public, and the NRC staff. These improvements are directed toward increasing the effectiveness of regulation by enhancing industry's understanding and its commitment to safety.

3. Improve efficiency or reduce licensing time

Research to improve review and evaluation techniques used in the regulatory process helps to make advanced technology more readily available for NRC application.

4. Improve perspective

Research to identify and quantify the benefits of the licensed activities makes possible more competent risk-benefit analyses and more informed public review and oversight.

5. Improve safety

Research investigating ways to improve the level of protection to the public health and safety of the environment in the operation of nuclear facilities focuses on the design of engineered safety features and other features to mitigate the consequences of accidents.

So that the reader may better understand the relationship between the objectives of the long-range research program and the regulatory programs of the agency, a discussion of how the LRRP categorizes and combines program subelements, regulatory functions, regulatory programs, and regulated activities follows.

NRC regulations imposed on licensees can be categorized as:

1. Accident prevention and mitigation (public health and safety),
2. Effluent control,
3. Occupational radiation exposure, and
4. Safeguards (against theft of special nuclear material and sabotage).

The regulatory functions are categorized as rulemaking, licensing review, and inspection and enforcement.

In addition, the Commission is required by the National Environmental Policy Act (NEPA) to do environmental impact assessments for major rulemaking or licensing actions. These assessments include consideration of human health effects, ecological effects, and socioeconomic effects.

The nuclear operations covered by these Commission actions can be categorized as

1. Nuclear reactors
2. Fuel cycle facilities
3. Transportation (shipment of radioactive material by licensees)
4. Waste-management activities
5. Material licensees
6. Medical use of isotopes

Different NRC staff offices have technical responsibilities in the same areas, especially in relation to environmental impact assessments. A prime example of this shared concern is the assessment of radiological health effects, which is relevant to all nuclear activities and to all regulatory program areas. On the other hand, technical matters involving accident prevention and mitigation (safety) tend to be more specifically related to particular regulated activities. The outline used for this long-term research plan reflects areas of commonality; it does not make program distinctions with significance only to the internal organization of NRC. To facilitate understanding and management of the research program, the plan is organized around the following major elements:

A. <u>Reactor Safety Research (RSR)</u>

   1. LOCA and Transient Research
   2. LOFT
   3. Plant Operational Safety
   4. Severe Accident Phenomena and Mitigation Research
   5. Siting and Environmental Research

B. Systems and Reliability Research (SRR)

1. Light Water Reactor Risk Assessment
2. Fuel Cycle Risk Assessment
3. Human Reliability Research
4. Reliability Engineering, Operations Research, and Decision Theory
5. Consequence Analysis

C. Safeguards, Fuel Cycle and Environmental Research (SAFER)

1. Waste Management
2. Safeguards
3. Fuel Cycle and Material Safety
4. Radiological Protection
5. Environmental Impact Assessment

One aspect of the outline used in this plan is that the individual program presentations deal with problems or issues of widely varying significance. This is a consequence of the intent to explain each element of the research program by relating it clearly to regulatory program objectives. This intent cannot be satisfied if smaller programs are too aggregated. On the other hand, larger programs can be explained to any desired level of detail.

One NRC functional area involves more than licensee regulation. This is the area of Federal emergency response (for example, the NRC response in the event of a serious accident, sabotage, theft incident, or nuclear threat requiring NRC action and/or the support of other agencies). The NRC emergency response functions are currently under review, and research needs in a few specific areas have been identified. These are discussed separately in the plan.

In the LRRP, research programs have been developed from consideration of the following aspects of the NRC regulatory programs:

1. NRC regulatory objectives: These are generally an expression of the desired level of protection or safety, stated in terms that are specific

to the activity being regulated. Efforts are underway within the NRC to define a safety goal first for nuclear power plants and subsequently for other regulatory programs.

2. Technical capabilities required to achieve the regulatory objectives: For all regulatory programs, this involves identifying and characterizing the hazards or sources of risk associated with the regulated activity; identifying and evaluating the effectiveness of possible safety measures aimed at reducing risk by prevention or mitigation; and assessing qualitatively or quantitatively the relative contribution to societal risks made by each hazard.

3. Current technical status of the NRC capabilities: This can be expressed in terms of the scope (or completeness) of NRC considerations; the depth of understanding of the phenomena involved; the level of uncertainty associated with regulatory assessments; and the ease (or clarity) with which the basis for making regulatory decisions can be communicated to the licensee and to the public.

The NRC research objectives are derived directly from an appreciation of weaknesses or gaps arising from an assessment of the current level of NRC capabilities. Research is done to test and improve the scope of NRC assessments through detailed expert analysis and review; to improve understanding of phenomena and to reduce uncertainties through analytical, empirical, and experimental research; and to improve NRC's ability to communicate by systematically formulating and documenting the technical bases for regulatory decisions.

PART A. REACTOR SAFETY RESEARCH

2. LOCA AND TRANSIENT RESEARCH

Regulatory reviews and Commission rulemaking matters require supportive experimental data and analytical models which address adverse plan accident transients, operator recognition of degraded core conditions and plan recovery models and plan monitoring systems. The basic objectives of loss-of-coolant accident (LOCA) and transient research efforts therefore are:

1. To examine the reactor system and component performance that occurs during a wide range of LOCAs, concentrating on the smaller breaks where the information needs are the greatest.

2. To examine the reactor system performance and system interactions that may occur during a wide range of transients.

3. To provide the licensing staff with data and tested safety analysis computer codes that are assessed against a meaningful data base which can be used to audit licensee requests for regulatory decisions.

These objectives will be achieved through:

1. continued use of the Semiscale facility to examine PWR plant accident scenarios;

2. a planned upgrade of the Two-Loop Test Apparatus (TLTA) which will provide a facility to investigate boiling water reactor (BWR) plant transients (in a manner similar to Semiscale experiments);

3. continuation of core heat transfer component experiments in existing separate effects facilities to provide a means for model development and code assessment;

4. performance of fuel tests under conditions that cause severe fuel damage, the formation and coolability of fuel debris beds, and the generation and control of hydrogen; and

5. completion of systems codes, including assessment against actual physical data, to analyze reactor transients and accidents.

The research program detailed in the following subsections addresses these objectives and will be accomplished during FY 1983-FY 1987. In addition, data generated by industry (EPRI, vendor) or international sources (JAERI, FRG) will be used so that the need for new facility investments will be minimized. The Lessons Learned Task Force reports (NUREG-0585, -0578) and NRC Action Plan (NUREG-0660) have been used to formulate specific program plans, together with interactive discussions with the Office of Nuclear Reactor Regulation, the Advisory Committee on Reactor Safeguards (ACRS) staff and the Commission.

2.1 Semiscale

Semiscale is a PWR system experimental facility that has the unique features of being versatile, able to perform experiments on short-time schedules, easy to modify such that a wide range of experiments can be performed, and capable of assessing accident monitoring instrumentation proposed for use in nuclear power plants. The facility will be used to address the following NRC issues:

1. Perform natural circulation experiments to better understand the phenomena, provide data for code assessment, and provide data on indicators of natural circulation in a power plant. Because under accident conditions power plants depend on this mode of heat removal at high pressure, it is important that a positive indication of natural circulation exist and the phenomena involved be well understood.

2. Natural circulation experiments with non-condensible gas in the primary system will be investigated. Non-condensible gases have the potential of blocking natural circulation from occurring and, therefore, leading to inadequate core-cooling transients. During these experiments, the system response to different quantities of non-condensible gas will be determined in an effort to establish critical level of non-condensible gas and possible detection schemes available to plant operators.

3. Small-break LOCA experiments will be conducted with and without upper head injection to assess this new ECCS design support of NRR and ACRS review of these plants.

4. Plant transients will be performed that have either high probability of occurrence or high risk of fuel damage. Data from these experiments will be used for code assessment and phenomena understanding to support accident mitigation, methods for termination of transients before reaching degraded core conditions and in support of the plant system behavior program.

5. Provide data to assess the vendor design of liquid-level detectors used in power plants. This experimental data will be used to evaluate current and proposed vendor liquid-level system to insure that the system gives operators an unambiguous indication of liquid levels.

The results of this Semiscale work will be an improved understanding of reactor accident scenarios, including how the operator can diagnose these accident conditions and possible alternative actions. Also, a major part of this program will be to provide data for computer code assessment by NRC and licensee (through the required problem analysis requested by NRR). These codes will be used to address accuracy of emergency plant operating procedures and to support rulemaking procedures.

2.1.1 Regulatory Objectives

The broad objective of the Semiscale program is to conduct integral systems experiments to provide information for:

1. Understanding of adverse plant transients, potential operator reactions, plant recovery modes and special plant features (i.e., upper head injection).

2. Rulemaking support derived from the test results which provide an insight into degraded plant and core conditons.

3. Assessing plant monitoring systems, such as liquid level detectors, by using Semiscale as a test bed, thereby providing a means to assess new plant safety features (i.e., mimimum safegurads required) and operator interpretation.

4. Development of simple plant models for analyzing adverse plant transients and assessment of plant systems codes.

2.1.2 Technical Capabilities Required

The Semiscale system that will be used for the FY 1981-1983 testing (designated the MOD-2A configuration) is a volume and power-scaled thermal/hydraulic test facility that is designed to study transient and LOCA events typical of 4-loop PWRs. The main facility components are illustrated in Figure 2-1. Planned FY 1981 tests include:

1. Small and intermediate breaks with and without UHI operation;

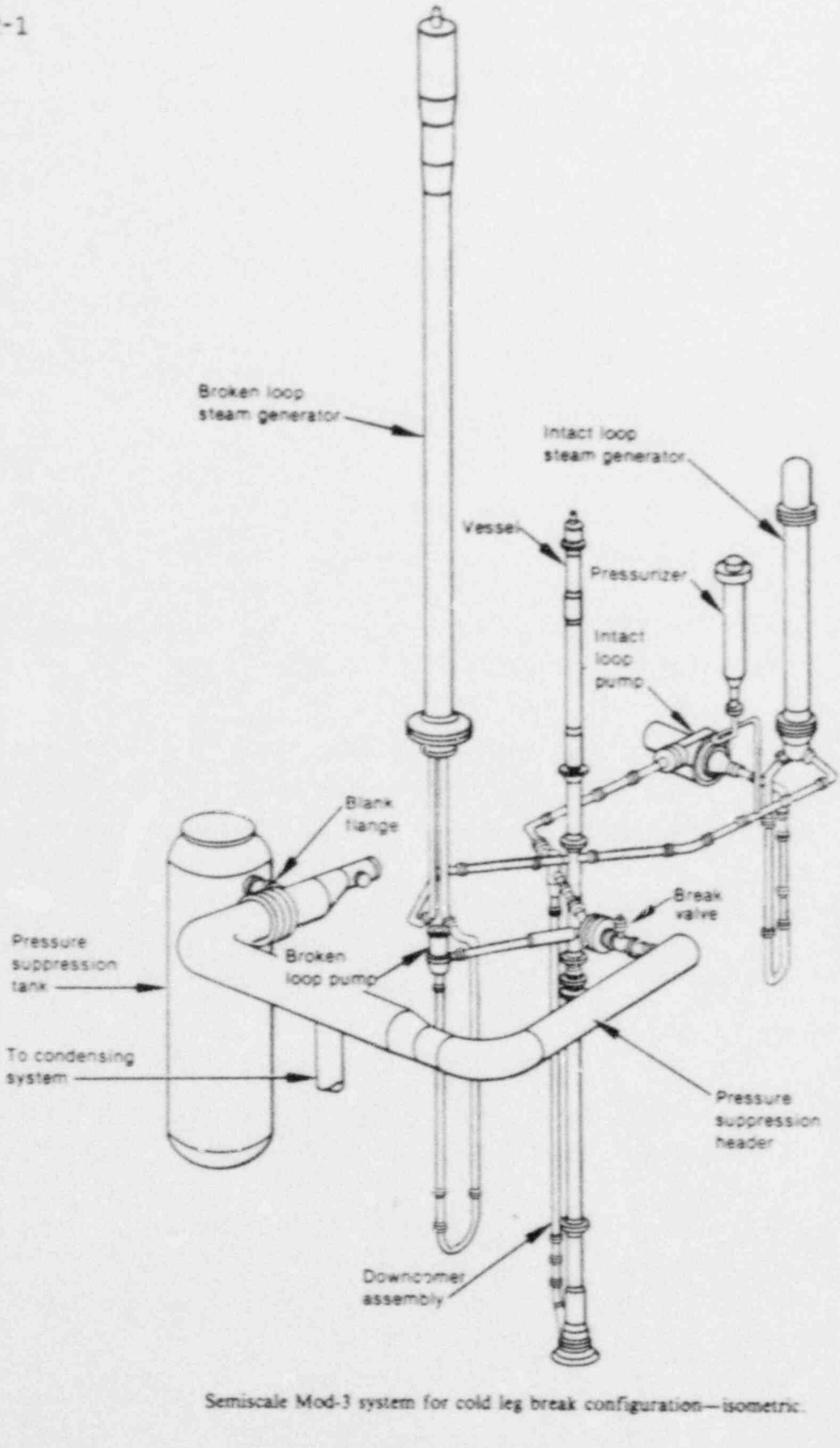
2. Natural circulation, including study of the effects of two-phase behavior and inert gas for a number of primary and secondary subsystem conditions; and

3. Plant transients leading to both steam generator tube rupture and secondary systems' failures.

In addition, Semiscale MOD-2A testing in FY 1984-1985 will address (where feasible) high priority LOFT tests previously planned but not implemented as a result of the phaseout in FY 1983 of LOFT as an operating test facility. It is also expected that Semiscale will be required to conduct licensing-related tests as has been the past and current practice (i.e. pumps on/off tests). This on-call capability affords NRR a significant licensing backup asset.

User need memoranda and ACRS recommendations are key items in establishing data needs from the Semiscale program. Recent requests and NRR comments include:

Figure 2-1



Semiscale Mod-3 system for cold leg break configuration—isometric.

Item	Reference	Areas
NRC (8/80, 11/80)	NRC Action Plan (NUREG-0660,-0737)	ECC, small break, transients, calculation improvement and uncertainty determination, decay heat removal, B&W sensitivity, safety decision making, procedures, venting.
ACRS (2/80, 10/80)	Reviews (NUREG-0657)	MOD 5 (B&W test configuration), transients, facility improvements, data for understanding and for code development and assessment, small-break LOCA, RELAP 5.
NRR (2/80, 10/80) 12/3/79, in preparation)	User need memos to RES	MOD 5, RELAP 5, test programs pertaining to transients and small breaks, UHI, behavior with core damage.

User needs (although mutually developed) are supplemented by comparisons to results obtained from recent tests. NRR and Semiscale project staff also pursue the following activities.

1. Identifying issues by examination of the probability of occurrence and interfacing with diverse information sources, then questioning the adequacy of current knowledge to determine where additiona work is needed;

2. Reviewing licensing issues with emphasis on high priority areas; and

3. Providing for data and analysis capability in areas of NRR licensing needs.

Extensive NRR-RES staff interaction and experimental result evaluations precede issuance of formal user requests and are used to develope Semiscale program plans.

2.1.3 Status of Capabilities

Since 1965, data and analyses have been provided by Semiscale for many PWR configurations. Many LOCAs have been simulated, principally in support of code development and assessment, rulemaking, and licensing concerns. Recent concentration has been on realistic evaluation of issues and response to the NRC Action Plan (NUREG-0660), including steam generator tube breaks (FY 1977), TMI (FY 1979), support to NRR audits (FY 1979-1980), pumps on/off (FY 1980), and station blackout (FY 1980). The recent facility upgrades included the correction of PWR nontypicalities such as component elevations, heat loss rate, and fluid leakage and the installation of a liquid-level detector in a vendor reactor vessel for evaluation.

2.1.4 Research Program Objectives

The objectives of the FY 1982-1983 test program will be:

1. Small LOCA experiments [with and without UHI, secondary systems(s) failures (e.g., steam and feedwater line breaks, steam generator tube failures)];

2. Adverse plant conditions (e.g., station blackout, St Lucie-type transient);

3. Selective plant transients (e.g., North Anna overfeed of 1/79, overcooling and return to power);

4. Evaluation of plant operational procedures and operator procedures and operator interactions;

5. Licensing issues and on-call requests; and

6. Evaluation of new plant monitoring instrumentation (i.e., liquid-level detection systems).

2.1.5 Research Program Plan

Testing in the Semiscale facility will be carried out using the MOD-2A configuration. Although this facility arrangement is currently representative of a Westinghouse 4-loop PWR, PWR plant similarities (with the exception of Once-Through steam Generator, (OTSG), effects) will permit application of this data base to a broad spectrum of PWR accident scenarios.

Simulation of Babcock & Wilcox (B&W) plant would require a MOD-5 upgrade. Although MOD-5 has received both NRR and ACRS endorsement, funding requirements are not included in this 5-year plan due to budget constraints. Rather, discussions are continuing with B&W and B&W users to determined if supplemental industry funding could be established to proceed with a MOD-5 upgrade. Perhaps, B&W plant users may elect to fund B&W plant transient testing independently. Our plans are to proceed therefore with the program plans outlined below.

FY 1981	Complete system conversion to MOD-2A (UHI simulation) configuration and facility shakedown; conduct small-break test series of natural circulation systems tests (with two-phase mixture and with noncondensibles); initiate steam generator tube break test series.
FY 1982	Complete steam generator tube break test series; conduct station blackout plant scenario tests; support NRR licensing issue requirements.
FY 1983	Conclude plant transients simulation test series; continue licensing support; initiate tests previously identified as high priority LOFT tests to be conducted after FY 1983, e.g., 16 percent breaks, ATWS, steam generator tube failures with other LOCA effects, postaccident core recovery scenarios.
FY 1984-1986	Conduct LOFT substitution tests, continue licensing safety issue support.

FY 1987 Provide licensing on-call support at reduced levels.

2.2 Separate Effects and Model Development

Separate effects and model development research consists of experimental programs to address BWR system response (BWR transient facility and SSTF), PWR core heat transfer full length emergency cooling heat transfer-separate effects and systems effects tests (FLECHT-SEASET and THTF), unresolved safety issues (USIs) and core heat transfer and steam generator performance model-development effects. Program plans include:

1. Testing BWR system transients and small breaks in an upgraded 2-loop test apparatus (TLTA) facility to provide data for code assessment, phenomena understanding and evaluation of operator guidelines. Experiments will be conducted that represent high probability or high-risk accident scenarios within the capability of the facility.

2. PWR core heat transfer during uncovered core conditions and natural circulation will be investigated utilizing data from THTF and FLECHT-SEASET, respectively. Data already collected from thermal hydraulic test facility (THTF) will be used to evaluate the effectiveness in steam cooling due to core uncovery and film boiling that proceeed degraded core conditions. These results will be used to support regulatory decisions, rulemaking activities and code development and assessment in two parts in this research program.

3. Experiments to evaluate the containment emergency sump performance during the recirculation phase of an accident will be completed in FY 83. These experiments address an unresolved safety issue (USI-A43) concerning the potential of degraded sump recirculation.

4. Model development efforts performed support the code development and assessment in this program. Heat transfer models for the post-critical heat flux (CHF) and steam condensation are being developed and assessed for both PWR and BWR. Steam generator thermal-hydraulic performance

during small-break LOCA and transient is also being investigated. These efforts will allow better prediction of small-break LOCA and transient behavior to address regulatory decisions and evaluation of plant transients leading to degraded core conditions.

5. Model-development efforts will be initiated to provide heat-transfer information for degraded core conditions. These models will be used in system codes to provide analysis of transients that result in degraded core conditions.

Separate effect experiments and model development activities will provide data for assessing core heat transfer phenomena, complement integral system experiments since these tests can be conducted with controlled boundary conditions, provide a means to develop models for plant system analysis codes, and provide data for rulemaking activities.

2.2.1 Regulatory Objective

The regulatory objectives of planned separate effects experiments and model development are:

1. Support for Commission decisionmaking. These efforts will be directed towards providing a sound data base supporting rulemaking hearings related to degraded core and plant safety features.

2. Data for analytical code and thermal/hydraulic model assessment. Separate effects experiments and model development experiments will be used to assess code and modeling adequacy for large LOCA, small breaks, plant transients and to support NRR's vendor code assessment efforts.

3. Research for unresolved safety issues (USIs). Research and studies addressing USI questions will continue to be developed, contracted for, and managed for NRR.

4. Research related to NRR's current licensing needs. This effort provides an on-call capability for NRR to investigate plant review and licensing issues through experiments in existing facilities, but with short-term turnaround.

5. Evaluation of the performance of proposed plant monitoring instrumentation and control systems.

2.2.2 Technical Capabilities Required

As previously noted, the accident at TMI-2 prompted immediate attention to small breaks and plant transients. NRR has identified research needs that require concluding data analyses and model assessments. These are:

NRR-79-20 (8/18/79)	Confirmatory Research on LWR Heat Transfer
NRR-79-27 (11/20/79)	Two-Loop Test Apparatus (TLTA), Small-Break Tests
NRR-79-30 (12/27/79)	Two-Phase Natural Circulation and Pump Performance
TAP-43 (12/80)	Containment Emergency Sump Performance

The broad scope of NRR-79-20 and specifics of NRR-79-27 and NRR-79-30 will require (1) concluding experiments in the facilities discussed below, (2) concluding data analysis and model evaluation from past separate effects experiments, and (3) collaborating results obtained from separate effects experiments with integral-system tests carried out in Semiscale and LOFT.

TAP-43 details the efforts required to achieve resolution of USI-43, which is being managed for NRR.

2.2.3 Status of Capabilities

The major separate effects programs were developed during and following the Appendix K hearings in 1973 and were therefore principally large-LOCA oriented.

Following the TMI-2 accident, the majority of separate effects and model development efforts were directed to research small-break and PWR operational transients. The thrust was, and is, to conclude large LOCA-research, use (if possible) facilities for current information needs and to phase out large, nonpriority programs. The pressurized water reactor blowdown heat transfer (PWR-BDHT) program at ORNL was redirected to investigate uncovered bundle heat transfer in FY 1979; these experiments, plus concluding large-LOCA experiments, were completed in FY 1981. This program will be phased out in FY 1981. Other separate effects programs can be summarized as follows:

Program	Large-Break LOCA Experiments	Small Breaks and Plant Transient Capability
BWR BD/ECC (TLTA Facility)	BWR bundle thermal hydraulics: and ECC performance research (completed)	Facility will be upgraded to investigate BWR small breaks and plant transients
FLECHT SEASET (Reflood heat transfer)	PWR bundle reflood response tests (will be phased out in FY 1982)	Evaluation for two-phase natural circulation tests underway
BWR REFILL/ REFLOOD (BWR spray distribution)	ECC spray distribution refill, reflood and countercurrent flow phenomena (will be phased out in FY 1982)	Some countercurrent flow and mixing phenomena tests applicable to small breaks will be conducted.

As noted above, these facilities have limitations and most will be phased out. However, a Commission paper for upgrading the Two-Loop Test Apparatus (TLTA) is being processed. An upgraded TLTA will provide a facility to investigate BWR-related accidents similar to Semiscale's investigating PWR accidents. These separate effects experiments investigate specific and identified concerns, using controlled and measured boundary conditions. Thus the data complements the Semiscale and LOFT system effects experiments and also provides a data set to assess model and/or code predictive capabilities.

On the other hand, basic phenomena-related concerns are addressed through small-scale model development experiments using the small lab experimental approach that may be scoping in nature. Typical phenomena investigated include condensation-controlled mass exchange, transition boiling, and rewet. For the most part, these programs have been basic research programs, funded at universities, and designed to investigate controlling phenomena. New programs have been undertaken as a result of NRR-identified plant-related needs. Examples of such work include steam generator behavior during transients and BWR multichannel stability research. Future work will include the study of thermal/hydraulic behavior applicable to degraded core conditions and related to code developments. The small size of the experimental equipment used permits newly identified phenomena to be more quickly investigated because major facilities are not involved.

2.2.4 Research Program Objectives

The research objectives of separate effects programs can be summarized as follows:

Program	Objectives
BWR BD/ECC (TLTA Facility) accident	A Commission paper for authority to proceed with upgrading TLTA to allow a better simulation of a BWR small-break and other BWR transients. Initial planning and design of the upgrade are projected to start in late FY 1981, with a short small-break test series projected for FY 1983. A 1 to 1-1/2 year period of BWR transient separate effects testing and analysis is then planned, starting in FY 1984.
FLECHT SEASET	This program is designed to investigate two-phase natural circulation phenomena that are representative of small-break accidents. Reflood testing will be phased out in FY 1982. The two-phase natural circulation tests will be concluded in FY 1983, and the results will be reported in FY 1984.
BWR-Refill/ Reflood	The refill/reflood program, initiated in FY 1979, was designed to evaluate ECC spray distribution within a BWR upper plenum region and to determine flooding limitations

induced by upper core hardware in support of Appendix K LOCA evaluation requirements. This program also provides model development in support of the BWR TRAC code development and will provide data useful for small-break analysis. Experimentation is scheduled to be completed in FY 1982. Data analysis and the reporting of results are to be concluded in FY 1983.

USI-43, Containment Emergency Sump Performance — This program was initiated in FY 1979, with Sandia awarding a contract to Alden Laboratory to build a full-scale sump hydraulics test facility. Testing initiated in FY 1981 is investigating vortexing phenomena and ingestion of air. RES is funding a plant insulation survey to determine the types of materials employed and the potential to generate debris. Completion of these efforts and potential resolution of this USI is projected for FY 1983. RES is managing and coordinating the required research and study efforts for NRR.

2.2.5 Research Program Plan

Model development and correlations verification encompasses small-scale experiments, data obtained from the large-scale separate effects experimental programs, and findings from integral-system tests. In FY 1982-1984, the emphasis will switch from the continued use of small-scale experiments to concentrating on data from separate effects experiments. After this evaluation, the results will be combined with results from similar Japanese Atomic Energy Research Institute (JAERI) and Federal Republic of Germany (FRG) experiments, and from Semiscale and LOFT to develop models and/or correlations. Key areas of understanding are displayed in Table 2-1. It should also be noted that, as a result of NRC contracts with various universities, recognized thermal/hydraulics experts have been available to respond to NRR day-to-day questions (particularly following the TMI-2 accident).

Recent results and projected plans for the separate effects programs are as follows:

Table 2-1 Overview of Model Development and Correlation Verification

NRR User's Need — RES Results and Schedule

Request from NRR	Information Needed	Data Sources: Completion of Exp. (FY)	Data Sources: Evaluation of Data (FY)	Model Verification
Heat Transfer NRR-79-20	Blowdown Heat Transfer			
	Post CHF data	THTF (1981) TLTA (1982) Semiscale, LOFT (1979-1981)	1981-1982 1983 1980-1981	
	Post CHF correlations	Lehigh Univ. (1983)	1983	1984
	Rewet model		1982	1982
	Refill/Reflood			
	Thermal/hydraulic model for unblocked bundle	FLECHT-SEASET (1981) CCTF (1983)	1982 1983	1983
	Thermal/hydraulic model for blocked bundle	FLECHT-SEASET (1981) SCTF (1985)	1982 1983-1986	1984
	2-dimensional effect on reflood	SCTF (1985)	1985-1986	1986
	3-dimensional effect on reflood	UPTF (1986)	1986-1987	1987
	Large-scale refill model	UPTF (1986) BWR refill-reflood	1983	1987

Table 2-1 (continued)

NRR User's Need		RES Results and Schedule		
		Data Sources		
Request from NRR	Information Needed	Completion of Exp. (FY)	Evaluation of Data (FY)	Model Verification
Operational Transient NRR-79-30	<u>Station Blackout</u>:			
	Data	Semiscale (1983)		
	Pumps on/pumps off model			1983
	Model for thermal/hydraulic response to transient			1986
	<u>Loss of Feedwater</u>			
	Data	LOFT (1981)	1982	
	Coolant - solid interaction (thermal/dynamic shocks)			1986
	<u>Alternate ECC</u>:	CCTF (1984)	1985	

Table 2-1 (continued)

NRR User's Need		RES Results and Schedule		
		Data Sources		
Request from NRR	Information Needed	Completion of Exp. (FY)	Evaluation of Data (FY)	Model Verification
Small Break NRR-79-30	<u>Core Uncovering</u>			
	Data on level swelling and steam heat transfer	Semiscale (1981-1982)		
		THTF (1981)	1982	
		LOFT (1983)	1980-1985	
	Phase separation/ distribution model	RPI (1981)		1984
	Steam heat transfer correlation			1983
	<u>Steam generator and Natural Circulation</u>:			
	Data	FLECHT-SEASET(1981)	1982	1983
		Semiscale (1982)	1983	
		CCTF (1984)	1985	
		PKL-1 (1980)	1981	
		MIT (1982)	1982	
	Condensation heat transfer correlation			1983
	Noncondensible effect on natural circulation			1983
	Reflux model			1983
	Flow distribution on steam generator			1984

Table 2-1 (continued)

NRR User's Need		RES Results and Schedule		
		Data Sources		
Request from NRR	Information Needed	Completion of Exp. (FY)	Evaluation of Data (FY)	Model Verification
Degraded Core	<u>Degraded Core Cooling</u>	LOFT (1983)	1982-85	
	Data	Programs from Fuel Branch		
		FLECHT-SEASET blocked bundle (1982)		
		BDHT Steam Cooling (1981)		1982 preliminary
	Model			1986 final

2.2.5.1 BWR BD/ECC Program

FY 1980 Completed large-break LOCA testing; results showed significant bundle cooling occurring during blowdown and early reflood. Conducted two small-break tests (per NRR request) to assist in evaluation of GE's analysis methods used to establish operator guidelines.

FY 1981 Complete analysis of large-break LOCA tests and report. Initial design effort for upgrade of the Two-Loop Test Apparatus (TLTA) for small-break LOCA simulation.

FY 1982 Complete TLTA upgrade and shakedown.

FY 1983 Carry out small-break experiments requested by NRR. Upgrade TLTA further for plant operational transients testing.

FY 1984 Plant operational transients testing.

FY 1985 Complete data analysis, report, and close down program.

2.2.5.2 BWR-CCFL

FY 1980 Completed ECC spray distribution tests used by NRR to qualify calculational methods to determine spray distribution. Completed reflood heat tranfer tests and developed initial models for the BWR-TRAC Code.

FY 1981 Conduct multidimensional refill testing in the 30° SSTF simulating a BWR/6. Continue to upgrade models for BWR-TRAC and initiate TRAC assessment activities.

FY 1982 Conduct multidimensional refill testing in the 30° SSTF simulating a BWR/4 and small-scale 360° refill testing to further quantify 30° SSTF results. Final models provided for BWR-TRAC and TRAC assessment completed.

FY 1983 Complete data analysis, phase out program, and report results.

2.2.5.3 FLECHT-SEASET

FY 1980 Completed 161-rod unblocked bundle (17 x 17 design) reflood tests. Completed steam generator heat transfer experiments.

FY 1981 Conclude unblocked bundle (161 rods) reflood data analysis and report findings. A Research Information Letter (RIL-T67), addressing the significance of low reflooding rates (<1 inch/sec) is planned. Develop steam generator (primary side) heat transfer correlations. Complete 21-rod blocked bundle experiments to establish local flow field effects.

FY 1982 Complete data analysis of 21-rod block bundle experiments, and assess assumptions made in Appendix K regarding blockage calculations. Conduct natural circulation systems-effects tests. Construct 161-rod-with-blockage test facility.

FY 1983 Analyze natural circulation experimental data. Conduct 161 rod-with-blockage test, and report results. Phase out program.

2.2.5.4 USI-43 Containment Emergency Sump Performance

FY 1980 Alden test facility completed and shakedown tests run. Plant insulation survey undertaken to establish types of insulation materials employed and potential for generation of debris under LOCA-induced loads.

FY 1981 Conduct sump performance tests investigating effects of geometrics, water levels, vortex formation, and plant-introduced flow effects. Conclude plant insulation survey. Develop interim findings report (using Alden results and survey results) for USI-43.

FY 1982 Complete Alden sump tests. Perform final audit on insulations employed in operating plants. Prepare final NUREG report for resolution of USI-43.

2.3 2D/3D Program

The 2D/3D program objectives are to provide phenomena understanding and a data base in large-scale and separate effects test facilities that can be used to assess the accuracy of computer codes used by NRC and licensees. Emphasis has been on the refill and reflood phase of LOCAs. This is an international cooperative program among Japan, the Federal Republic of Germany, and the USNRC; NRC will provide instrumentation and analytical support. Major results from these program to accomplish this are:

1. Determine the effectiveness of emergency core cooling during the reflood phase of a LOCA transient, including any three-dimensional effects. This will be accomplished by providing a data base for code assessment and application of these codes to power plant LOCA transients.

2. Determine the effect of core blockage that is expected to occur during inadequate core cooling transients. Data from these experiments will be used to improve and assess code capabilities to predict flow around blockage and heat transfer in blockage regions.

3. Evaluate the effect of ECC bypass during LOCA conditions. These results will be used to assess computer code capabilities to predict penetration of ECC fluid to the lower plenum.

4. Provide additional information on adequate core cooling during small-and medium-break sizes. Test data have been obtained at the German PKL facility on natural circulation and reflux boiling mode of core cooling for a spectrum of break configurations. These data will be used for computer code assessment.

   Alternate ECC injection mode, such as hot leg, cold leg and lower plenum will be evaluated in all the facilities. This will provide data and guidance for making future determination of ECCS during future rulemaking procedures.

2.3.1 Regulatory Objective

The 2D/3D program was initiated in 1978 as a joint international research program among the NRC, the Federal Republic of Germany (FRG) Ministry for Research, and the Japanese Atomic Energy Research Institute (JAERI) to study the thermal/hydraulic behavior of the ECC during the refill and reflood phases of a postulated LOCA in a PWR, including the effects of steam binding and bundle flow blockage on reflood core cooling. Research to resolve questions of ECC bypass in the downcomer during refill/reflood was subsequently added to the program. Following the TMI-2 accident, the program of research was extended to include small-break LOCAs and both natural circulation and reflux boiling mode cooling of the core and core uncovery tests. Large-scale experimental facilities are to be built in Japan and Germany to conduct tests concerning the above LOCA questions. NRC's responsibility in the 2D/3D program includes providing advanced two-phase instrumentation for key measurements and analytical support using the TRAC computer code to provide design and pretest and posttest experimental analyses for all the Japanese and German test facilities.

2.3.2 Technical Capabilities Required

The NRC requires applicants to consider a large-break LOCA in the design of their nuclear power stations. The 3D program will enable NRC to assess the models and assumptions used in licensing criteria for refill and reflood and to develop improved models.

The effectiveness of the ECCS during refill and reflood will determine the level of fuel damage and hence fission-product release. The 3D program will provide information for optimizing the injection locations and activation pressures of various ECC systems to minimize peak clad temperatures and, hence, potential fission-product release.

The added scope of the 3D program, which is concerned with small breaks and natural circulation, will contribute to NRC understanding and provide data for establishing rules and criteria for PWR plant operation to minimize the potential for core damage during this type of accident.

The blocked-bundle tests will provide needed data on cooling a blocked or obstructed fuel rod bundle by providing information on associated fuel temperatures and core-coolant distribution within the core under conditions of gravity and forced-flow cooling during reflood. This information will assist NRC in evaluating the safety provisions for cooling a blocked core and will provide data for determining the extent of core damage and consequent risk to the public.

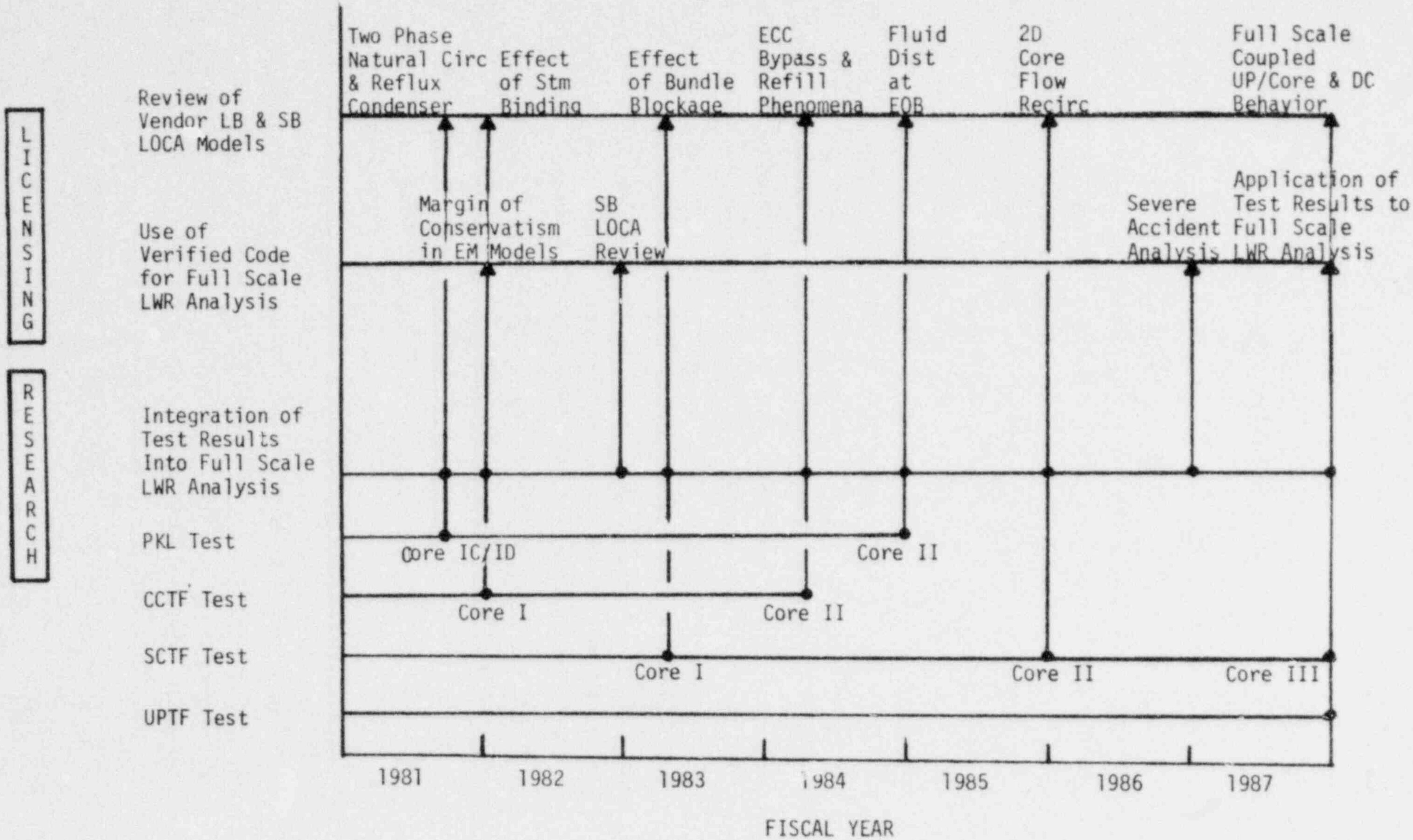
The Office of Nuclear Reactor Regulation and the Commission endorsed the 3D refill and reflood research objectives and program contained in SECY 77-20, 78-409, 78-594, and 79-391. The 3D program was extended by SECY 78-409 to include blocked-bundle tests, and full-scale ECC bypass experiments were added by SECY 78-594. Endorsement of these programs was provided by NRR concurrence in the above SECY memoranda. In addition, the need for confirmatory research on reflood heat transfer in both blocked and unblocked bundle configurations was requested by NRR in NRR-79-20.

Since TMI, the 3D program has been further extended and redirected to include natural circulation (including reflux boiling) and small-break LOCA tests. The objectives and results of these tests will fulfill some of the goals and information needs outlined in the NRC Action Plan (NUREG-0660). The relationship between the NRC licensing considerations and the 3D program is shown in Figure 2-2.

2.3.3 Status of Capabilities

The 3D program involves cooperation among FRG, NRC, and JAERI in a study fulfilling the objectives outlined above. The program was developed in meetings of representatives of the three parties in 1977-1979 and approved by official agreement in 1980. In 1978, work began at laboratories in the three countries in anticipation of formal approval.

Figure 2-2 The Relationship Between Licensing and Research For the 2D/3D Program



In 1978, JAERI completed construction of a 2,000 heater-rod full-length cylindrical core test facility (CCTF) with full-height primary loops and steam generators simulating a PWR. A series of 20 tests, including both the refill and reflood phases of a large-break LOCA, was conducted in 1979-1980.

In 1980, FRG completed the design of a full-scale reactor vessel facility (including a steam-water core simulator) with simulated loops and steam generators to be used for upper plenum tests and ECC downcomer bypass tests during the refill and reflood phases of a large-break LOCA. Construction of this facility, the Upper Plenum Test Facility (UPTF), is scheduled to start in 1981, and testing is to begin early in 1985 and be concluded in 1987.

JAERI is nearing the completion of construction of a 2,000 full-length heater-rod slab core test facility (SCTF) that uses a configuration to represent a radial segment of a PWR with associated simulated loops and steam generator. This facility will be completed by April 1981 and a series of 20 blocked-bundle refill and reflood tests will be started in May 1981. Unblocked slab cores will be used later in testing alternate ECCs in 1983-1985 and to provide for engineering coupling tests with the FRG UPTF facility so that the proper initial and boundary conditions of the steam-water injection core simulator can be established in UPTF.

The NRC is providing key instruments to measure two-phase flow to be installed in all three of the above facilities. In addition, NRC is providing analytical support by performing facility-design calculations and pretest and posttest analyses of experiments using the advanced multidimensional two-fluid transient analysis code, TRAC. The 3D program has also supported FRG in the small-break natural circulation and core uncovery tests conducted in the PKL facility in 1980 by the loan of instruments and experiment analysis. The NRC 3D program contractors and their specific areas of responsibility are listed below:

Contractor	Items Supplied
EG&G	spool pieces, gamma densitometers, drag discs, turbine flowmeters, liquid-level detectors, fluid distribution grids, air/water flow tests, optical probes

ORNL — impedance probes, film probes, string probes, drag bodies and DP transducers, instrument development loop tests

LASL — video probe instruments, TRAC calculations (design support, pretest predictions, posttest analyses, and measurement requirements), model tests

MPR — documentation and instrument specification support

NUS — facility-site resident engineers

The test requirements of the 3D program of research are shown in Table 2-2. The schedule for completing construction of the facilities and performing the experiments is given in Figure 2-3.

2.3.4 Research Program Objective

Specific research program objectives are listed below:

1. Support for evaluating the best estimate (BE) and evaluation model (EM) code calculations used by NRR to assess the vendors' calculations for reflood during a LOCA in a PWR;

2. Assessment of ECC bypass during the refill/reflood phase of a LOCA in a PWR;

3. Collection of data on core cooling by natural circulation and reflux boiling during small-break LOCAs;

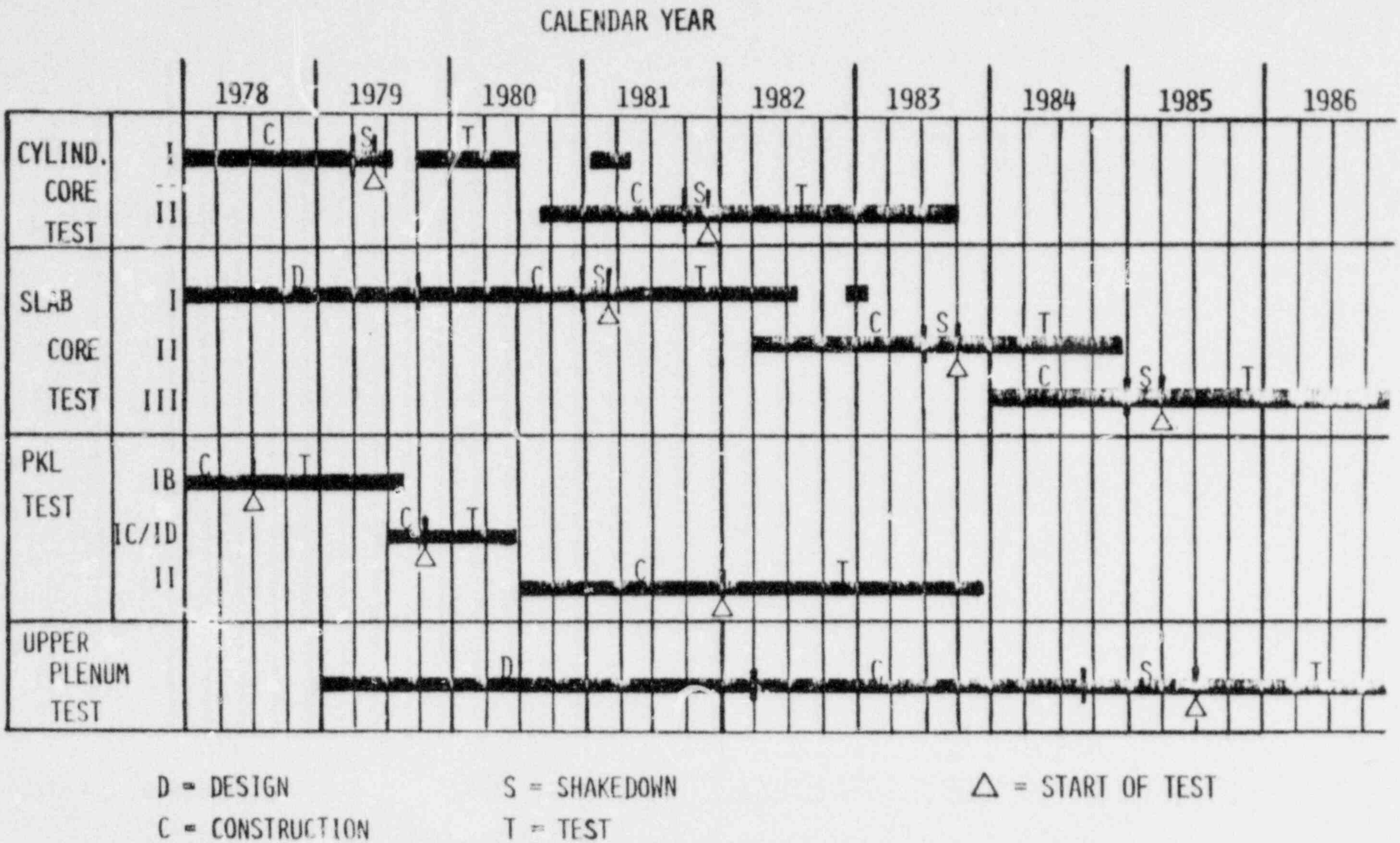
4. Study of core thermal/hydraulic behavior (convective flow and temperature distribution inside a heated core) during reflood for a large-break LOCA during core uncovery and under conditions of core blockage during reflood;

5. Study of events leading to core uncovery during the end of a small-break LOCA; and

Table 2-2 Test Requirements for the 2D/3D Program

Types of Tests To Meet Objectives		Test Facilities: Integral Effects: PKL	Test Facilities: Integral Effects: CCTF	Test Facilities: Separate Effects: SCTF	Test Facilities: Separate Effects: UPTF	Four Objectives To Be Met: I. Effect of Various ECC Injections on Steam Binding for Large-Break LOCA	II. Flow Hydrodynamics in PWR Vessel During Refill for Large-Break LOCA	III. Events Leading to Core Uncovery and/or Recovery for Small-Break LOCA	IV. Flow and Temp. Dist. in a Heated Core
Under Large-Break Refill/Reflood Conditions, Effectiveness of:	Cold-Leg Injection	X	X	X	X				
	Combined Hot-Leg and Cold-Leg Injection	X	X	X	X	X	X		X
	Lower Plenum Injection		X	X	X				
	Vent Valves		X	X	X				
Refill/Bypass in Annular Downcomer Under Cold-Leg Injection During Refill for Large-Break LOCA			X		X		X		
Phase Separation in Upper Plenum and Hot Leg Under Small-Break LOCA Conditions		X	X	X	X			X	
Core Uncovery and Recovery During Small-Break and Natural Circulation Conditions, Including Steam Generator and Vessel Flow Interaction		X	X	X				X	X
Flow Blockage Effects During Reflood				X					X

Figure 2-3 2D/3D Test Facility Schedule



6. Assessment of the effectiveness of alternative ECC system designs (cold-leg injection, combined hot-leg and cold-leg injection, lower plenum injection, and use of vent valves) for limiting peak clad temperatures during reflood for a LOCA.

2.3.5 Research Program Plan

Results obtained so far and the research program plan are shown below by fiscal year. The schedule for experiments in the 2D/3D facilities is given in Figure 2-3.

FY 1978	Initiated the construction of the CCTF. Initiated the design of the SCTF. Completed the assessment of the two-phase flow instrumentation and selected the type and quantity of NRC instruments to be provided to CCTF, SCTF, PKL, and UPTF. Initiated the fabrication of instruments for CCTF I.
FY 1979	Completed the construction of CCTF I, including the NRC instrumentation. Completed the design of SCTF I. Completed the fabrication of PKL IC and ID. Initiated the design of UPTF. Performed the TRAC calculations for SCTF design assistance (steam supply problem).
FY 1980	Completed 20 tests in CCTF I; peak clad temperatures were found to be much less than those assumed previously; the core was quenched within 10 min. Completed small-break LOCA tests at PKL IC/ID, showing that the decay heat can be effectively removed by natural circulation or reflux condenser mode of cooling. Developed film and impedance probes for two-phase flow measurements (void fraction, local velocity, film thickness, and film flow) under high-temperature conditions. Performed the TRAC calculations for UPTF design assistance (flow oscillation in hot legs).

FY 1981	Complete CCTF I test series; complete TRAC analysis for CCTF I tests: analyze the data, and formulate recommendations for NRC licensing consideration. Start fabrication of CCTF Core II, including the NRC instrumentation. Complete the construction of SCTF Core I and initiate blocked-bundle reflood tests. Complete air/water and steam/water tests for the development of upper plenum/core interface instrumentation.
FY 1982	Initiate the fabrication of SCTF Core II. Complete the fabrication of PKL Core II. Complete the design of UPTF. Perform TRAC calculations for pretest and posttest analyses of SCTF I tests.
FY 1983	Complete the SCTF I test series. Complete the fabrication of SCTF Core II. Complete the TRAC analysis of SCTF I tests. Analyze the SCTF I data, and formulate recommendations for NRC licensing consideration. Complete the CCTF II test series.
FY 1984	Complete the TRAC analysis of CCTF II tests. Analyze the CCTF II data, and formulate recommendations for NRC licensing consideration. Complete the PKL II test series. Complete the TRAC analysis of PKL II tests. Analyze the PKL II data, and formulate recommendations for NRC licensing consideration. Complete the construction of UPTF.
FY 1985	Complete the SCTF II test series. Complete the TRAC analysis of SCTF II data. Analyze the SCTF II data, and formulate the recommendations for NRC licensing consideration. Complete the fabrication of SCTF Core III. Perform the TRAC calculations to couple the SCTF Core III and UPTF tests, i.e., defining the initial and boundary conditions consistent to both facilities.

FY 1986 Perform the pretest and posttest analyses using TRAC for SCTF CORE III and UPTF tests.

FY 1987 Complete the SCTF III and UPTF test series. Complete the TRAC analysis for SCTF III and UPTF tests. Analyze the SCTF III and UPTF data, and formulate recommendations for NRC licensing consideration. Review the recommendations formulated earlier based on the data obtained from each facility to check if any revision is needed in view of the fact that the data from the other facilities may improve or alter the earlier interpretation of the data.

2.4 Code Improvement and Maintenance

Computer codes have a major role in both the plant licensing process and investigation of plant transients and LOCAs. These codes are used to investigate system response, control room instrumentation and operator action in plant transients that lead to degraded core conditions. The objective of this research is to provide the analysis tools for both PWR and BWR necessary to support the rulemaking procedures and licensing decisions. These codes will also be used to assist in the planning of experiments and analysis of experimental programs. To accomplish these objectives, the following tasks will be completed:

1. System codes will be developed to analyze plant transients and small breaks for both PWRs and BWRs. These will then be available to perform analyses of plant transients that lead to degraded core conditions. Alternate modes of plant recovery an operator action during these accident conditions will be investigated under other programs, using these codes.

2. Severe core damage analysis package developed under other programs will be included in the transient and small-break analysis code. This will provide the analysis capability to investigate coolability of degraded cores, recovery procedures and accident mitigation schemes.

4. The subchannel analysis code will be completed to provide the capability of performing analysis of local blockage in degraded cores. This code is to be used for determining the flow distribution and coolability of cores having blockage expected in core uncovery transients.

In order of priority, the system codes must be capable of describing (a) operational transients, (b) small-break LOCAs, (c) non-LOCA accident conditions involving undercooling and overcooling, (d) medium and large-break LOCAs. Codes developed will be used to support rulemaking procedures (minimum safeguards and degraded core cooling) and licensing issues.

2.4.1 Regulatory Objective

Adequate computer codes for evaluating the safety of nuclear power plants are needed. The codes that describe the LWR system response fall in two major categories:

1. The so-called Best Estimate computer codes are designed to predict, as realistically as possible, the actual processes and phenomena in LWRs. Such Best Estimate codes are useful to NRR as tools for licensing audit calculations of those accidents and transients for which conservative assumptions are not required. In addition, they are used by the NRC for studies of consequences of multiple failures and of operator actions and for calculations in support of rulemaking activities. Other uses of the Best Estimate codes are given in Section 2.4.

2. The Evaluation Model computer codes are designed, using pessimistic assumptions, to conservatively upper-bound consequences of accidents for which the conservative analysis is prescribed by 10 CFR Part 50. These codes are also used by NRR for licensing audit calculations.

2.4.2 Technical Capabilities Required

The following discussion shows the relationship between the various parts of the program element and the user requirements.

1. NRR requested a more detailed description of the complex interactions taking place during a LOCA in order to assess the degree of conservatism embodied in the licensing calculations. In essence, they require advanced system and component codes.

2. In 1976, NRR requested that the COBRA-4 code be extended to describe the upper-head-injection behavior during LOCA.

3. In 1977, NRR requested that an interim licensing Evaluation Model (EM) code package be developed for evaluation of LOCA consequences in both BWR and PWR plants (in that order) that employ the current versions of the existing codes. In addition, this NRR request also identified a need for user-convenient Best Estimate (BE) codes suitable for repetitive calculations performed as part of the sensitivity and uncertainty studies.

4. In 1978, NRR requested that RES take over the sponsorship at BNL of programs related to upgrading of the reactor physics codes.

5. In 1980, NRR requested implementation in RELAP-4/MOD-7 code of boron insertion and tracking of its transport.

6. In the report to Congress entitled "Review and Evaluation of the NRC Safety-Research Program for FY 1981" (NUREG-0657), ACRS urged implementation of a vigorous program to upgrade reactor simulators.

2.4.3 Status of Capabilities

The geometry and phenomena to be simulated during severe accidents in full-scale LWRs have been considered too complicated to rely on a single computer code or code version for accurate simulation of all transients. The approach has, therefore, been to develop and assess specialized computer simulations of both PWR systems, BWR systems and components.

To meet the objective of analyses of several accident scenarios that are dominated by widely differing physical phenomena, further specialization of code development has been required. Thus some codes, or different versions of the same code, have been designed to give the best simulation for specific accident categories.

The codes being developed and improved under the sponsorship of the NRC are divided into two broad categories, systems codes and component codes. Systems codes must be capable of analyzing thermal/hydraulic transients in PWR and BWR power plants. In contrast, the component codes model in greater detail the behavior of the various individual components of a reactor coolant system.

2.4.3.1 Systems Codes

Historically, the NRC best-estimate (BE) computer code development has proceeded from simple physical modeling to more realistic modeling. Table 2-3 shows completed computer codes that have been upgraded for BE analysis of various categories of accidents and transients. Table 2-4 shows completed computer codes that have been designed with more advanced models for BE analysis. The term "completed code" implies that the scheduled development has been completed; many of these codes are being maintained for current use and are being improved as errors are uncovered or more appropriate models become available. It must be pointed out that, in the case of the small-break LOCA (SBLOCA) and transient categories in Table 2-3, the listed upgraded codes do not address all events (accidents/transients) of interest. RES is currently preparing a report that lists all modeling requirements for each scenario mentioned in WASH-1400 and in Chapter 15 of FSARs, to indicate which currently available and/or future codes can address them.

Table 2-3 Completed "Upgraded" Codes for Best Estimate Analysis of LWR Systems

ACCIDENT SCENARIO	LBLOCA	SBLOCA	TRANSIENTS	ATWS & RIA
PWR CODES	RELAP-4/MOD7 RELAP-UHI	RELAP-4/MOD7	RELAP-4/MOD7, IRT	RELAP-3B
BWR CODES	RELAP-4/MOD7 (limited applicability)	RELAP-4/MOD7	RELAP-3B RAMONA-III	RAMONA-III

Table 2-4 Completed Advanced Code for Best Estimate Analysis of LWR Systems

ACCIDENT SCENARIO	LBLOCA	SBLOCA	TRANSIENTS
PWR CODES	TRAC-PD2 COBRA/TRAC (FOR UHI)	RELAP-5/MOD1	TRAC-PD2, RELAP-5/MOD1 (Both limited applicability)
BWR CODES	TRAC-BD1	TRAC-BD1 (limited applicability)	

Table 2-5 Completed Codes Suitable for Licensing Audit of Conservative Analyses of LWR Systems

SCENARIO	LBLOCA	SBLOCA	TRANSIENTS	ATWS & RIA
PWR CODE	WRAP/PWR RELAP-UHI	RELAP-4/MOD7	IRT, RELAP-3B, RELAP-4/MOD7	RELAP-3B
BWR CODE	WRAP/BWR	WRAP/BWR RELAP-4/MOD7	RELAP-3B RELAP-4/MOD7	RELAP-3B

System codes have also been developed that can provide conservative analyses to audit vendor licensing calculations. These codes are listed in Table 2.5. Two code packages in particular, the WRAP-PWR-EM package for PWRs and the WRAP-BWR-EM package for BWRs, provide the only stylized Evaluation Model (EM) capability that NRC has.

The details of some of the completed codes are described in the already issued Research Information Letters.

Code	RIL#	Date of Issue
RELAP-4/MOD6	39	11/27/78
TRAC-P1A	92	6/18/80
RELAP-4/MOD7	107	12/16/80
WRAP-PWR-EM	108	1/6/81
WRAP-BWR-EM	109	1/6/81

TRAC-P1A has recently been superseded by TRAC-PD2. RILs for TRAC-PD2 and RAMONA-III are currently under preparation, to be followed by RILs for COBRA/TRAC and for TRAC-BDa codes.

TRAC-PD2 and TRAC-BD1 represent advanced, best estimate codes capable of multi diversional analysis of the large and intermediate break LOCAs in PWRs and BWRs, respectively. These codes can also address the main steam line break (without multidimensional reactor kinetics feedback), and a variety of other accidents and transients. They are not, however, optimized for fast and economical calculations of accidents and transients of long duration and are not provided with (a) all the desired controls and trips, and (b) with the non-condensible gas and the boron tracking models. In contrast, the RELAP-5/MOD1 code already contains these capabilities although its controls and trips models may not yet cover all situations of interest. That code views system thermo-hydraulics one-dimensionally and is currently limited to analyses of transient, small break LOCAs, and some of the other accidents in PWRs. The lack of a core reflood mode prevents its applicability to the large break LOCA scoping studies.

COBRA/TRAC is also an advanced, multidimensional system code, specifically aimed at analysis of LOCAs in PWR plants that feature the Upper Head Inspection engineered safeguard. RAMONA-III is the only currently available code capable of addressing the three-dimensional reactor kinetics feedback during transients and non-LOCA accidents in BWRs. It does not yet have the boron tracking capability nor all the desired trips and controls.

2.4.3.2 Component Codes

The component codes allow various reactor components to be nodalized and modeled in a way as complex as desired. The effects of gradual simplifications in nodalization and in modeling can be studied. Thus, for any given component, one can establish the simplest model that still retains the features significantly affecting the behavior of the whole plant system. The model of the component can be tested in various separate effects tests. In addition, component codes have been developed to analyze specialized safety problems that require modeling not needed in systems codes.

Development of all of the component codes has been completed except for the COBRA-TF core subchannel code which may play a significant role in modeling of the severe core damage.

2.4.4 Research Program Objectives

As stated in the preamble, systems codes will be used in support of the Minimum Safeguards and of the Degraded Core Rulemaking for support of the licensing audit work, and for other missions. To reach this objective work is needed to:

1. Complete the development, then provide maintenance, of the fast running BE codes prediction of consequences of the long duration accidents and transients in PWRs and BWRs. Many operational and anticipated transients, as well as the small-break LOCAS, fall into that category.

2. Incorporate, into the existing systems codes, the capability to account for the multidimensional neutronics feedback. As previously mentioned,

this is necessary for analyses of those transients and accidents in which the non-uniform fluid conditions and/or non-uniform control rod insertion could result in the local burnout of fuel rods. Anticipated Transients Without Scram (ATWS), non-symmetrical loop operation, and asymmetric injection of reactivity accidents fall into this category.

3. Continue maintenance of conservative EM codes for PWRs and BWRs. In the case of the large-break LOCAs, this implies maintenance of the WRAP-PWR and WRAP-BWR code packages.

4. Modeling of the severe core damage mechanisms will be completed to the PWR and BWR systems codes.

The additional research program objective is to assist in upgrading the PWR and BWR training simulators so that they are capable of simulating severe accident conditions.

2.4.5 Research Program Plan

2.4.5.1 Systems Codes

The fast-running PWR version of TRAC (TRAC-PF1) is under development at LASL and is scheduled for public release in 1981. The fast-running BWR version of TRAC (TRAC-BD1/MOD1) will be developed at INEL during FY 1981-1982, based on experience gained at LASL with TRAC-PF1.

The aim of the TRAC-PF1 and TRAC-BD1/MOD1 versions is to run BE systems analyses for small-break LOCA and for non-LOCA transients in real time. These code versions will employ a two-fluid (full thermal and mechanical nonequilibrium) model in all parts of the reactor system, accounting for presence of the noncondensible gas and with the capability for one-dimensional, as well as multidimensional representation of the reactor vessel. Extensive modeling of plant trips and controls, boron tracking, and self-initialization will be featured in TRAC-PF1/MOD1 and in TRAC-BD1/MOD1. The final PWR version of TRAC will be TRAC-PD3.

It is attempted to make the PWR and the BWR versions of TRAC as similar (in basic modeling and in numerics) as feasible to reduce the magnitude of the independent assessment.

2.4.5.2 Component Codes

Subchannel codes have always been used by the industry to determine the local "worst condition." The subchannel code must be capable of including the effects of the local flow blockages as induced by the clad swelling or rupture effects, of asymmetric heat transfer within the clad that has been induced by nonuniform heat removal at the clad surface, and of thermal radiation to the adjacent rods, control rods, or fuel bundle cannister (as in BWRs). For this purpose, the work planned for FY 1981 includes (1) incorporation into COBRA-TF of those (verified) fuel-behavior-code modules that calculate fuel rod deformation, clad swelling, and gap conductance, (2) incorporation of one of the existing thermal-radiation models, and (3) compatibility for accepting boundary conditions from the systems-code output tapes. This subchannel code is slated for completion late in FY 1982.

2.4.5.3 Upgrading Plant Simulators

The details of the work scope for achieving this objective are described in the Request for Endorsement letter sent from RES to NRR on December 15, 1980.

2.4.5.4 Results

FY 1981 - Complete the first fast running version of TRAC (TRAC-PF1/) applicable to analysis of small-break LOCAs and of some transients in PWRs.

FY 1982 - Complete the fast running versions of TRAC for analyses of all transients and accidents in PWRs (TRAC PFI/MOD1) and in BWRs (TRAC-BD1/MOD1).

FY 1983 - Complete development of RELAP5/MOD2

- Complete development of TRAC-PD3 and TRAC-BD2 versions that incorporate effects of multidimensional neutronics feedback, in PWRs and BWRs, respectively.

- Complete integration of the severe core damage vessel module, into the selected systems codes for PWRs and BWRs.

FY 1986 - Complete demonstration of the developed software, on the acquired hardware, for upgrading the reactor coolant system representation in BWR simulators.

FY 1987 - Provide maintenance of codes.

2.5 Code Assessment and Application

In this research, test data are used to assess the accuracy of the PWR and BWR TRAC and RELAP-5 systems codes in predictings the phenomena occurring during operational (class II) transients, anticipated transients with and without scram, non-LOCA accidents, and small, intermediate, and large break loss-of-coolant accidents. The test date will come from:

1. incidents and startup tests in nuclear power plants, to the extent feasible,

2. NRC sponsored experiments described in other parts of this 5-year plan,

3. foreign research program, and

4. other U.S. research programs.

Less emphasis will be placed on assessment of the large-break LOCA codes and more on codes designed to address transients, small-break LOCAs, and non-LOCA accidents. Transients that include multifault accidents leading to and including degraded core conditions will be included in the code assessment process.

A major task of this research will be the application of codes developed in the code improvement and maintenance program. Code application efforts will be used to support resolution for Commission decisions, unresolved safety issues, rulemaking procedures and plant licensing issues. This effort involves the end product of code development and code assessment.

2.5.1 Regulatory Objectives

The TMI-2 Lessons Learned Task Force identified the need for short- and long-term analyses. The primary purpose of these analyses are to identify the changes needed in emergency procedures to bring the reactor to cold shutdown, to evaluate the single-failure criterion, and to assess operator errors. The Task Force recommends that RES perform analyses of selected transients, using the best available computer codes, to provide the basis for comparisons with the analytical methods used by the reactor vendors. The Task Force also recommends verification of the analytical methods through comparisons with test data.

2.5.2 Technical Capabilities Required

The measure of code capability is reflected in the degree of uncertainty with which actual events are predicted. Differences between the experimental results and code best-estimate prediction results are comprised of (1) uncertainties in the code input as a result of measurement errors in the initial state of the plant and boundary conditions, (2) the code's modeling inadequacy and weaknesses in the numerical solution technique, and (3) measurement errors in test results.

The appropriate test data base must be carefully selected to make it possible to (1) determine the code accuracy in predicting the measured results for test cases that are relevant to the code mission, and (2) extrapolate the code accuracy to LWR application.

Finally, it is necessary to establish the degree of accuracy acceptable to the NRC to help determine when further code improvements become unnecessary.

2.5.3 Status of Capabilities

Two systems codes have been independently assessed. The Research Information Letter No. 90 on independent assessment of RELAP-4/MOD6 code issued on May 22, 1980 gives details of the assessment findings.

These findings indicate that there are large uncertainties in the RELAP-4/MOD 6 code results. These uncertainties are, or can be made, in a conservative direction by judicial selection of input parameters.

The RELAP-4/MOD 7 code, which superseded the MOD 6 code version, has been used extensively at INEL for predicting the consequences of small-break LOCAs. The experience gained has been variable, which may or may not be related to operational problems in the test facilities.

A topical report on the capability of RELAP-4/MOD 7 to calculate small-break LOCA consequences will be issued after completion of the currently planned analyses of small-break pumps on/off tests in LOFT and Semiscale. There are no plans to perform an independent assessment of RELAP-4/MOD 7 since the advanced codes (TRAC and RELAP-5) have superseded it.

The TRAC-P1A code was independently assessed by BNL, INEL, and LASL Staff.

Results of TRAC-P1A assessment at BNL are reported in BNL-NUREG-27580, -51258, -28290, and NUREG/CR-1651. LASL results are reported in LA-8477 while INEL results appear in EGG-CAAP-5147, -0572, -5189, -5190, and -5191.

All results are summarized in RIL No. 90.

2.5.4 Research Program Objectives

The end goal of code assessment is to ensure that reliable best estimate codes of known and acceptable accuracy are available to the NRC in the licensing audit and safety evaluation activities.

The primary means of reaching this goal involves comparisons of code results against test data from carefully selected experiments that (1) subject the code to the environment present in the majority of accidents and transients of interest in reactor safety evaluation and (2) feature enough variations of geometric scale to examine the code's capability of extrapolating the knowledge gained from test facilities to full-scale LWRs. Code weaknesses uncovered in the course of code assessment that prevent the code from reaching the accuracy goals need to be communicated to code developers in timely fashion to expedite the required code improvements.

The subsidiary and complementary objective of this task is to plan, manage and fund participation in the domestic and international Standard Problem exercises. This activity contributes to the code assessment process.

There are two Standard Problem programs: The first is the U.S. Standard Problems (USSP) program involving only the U.S participants. The second is the International Standard Problems (ISP) program where the participation is international and managed by OECD/CSNI. It is anticipated that the Standard Problem programs will continue as long as important test data are being generated in domestic and foreign test facilities, i.e., at least through FY 1985.

The aim of code application is to (1) per NRR requests, provide plant transient analysis (2) determine the signature of transients or accidents perceivable by the plant operator; and (3) evaluate and recommend the most desirable operator actions that will bring the plant to safe shutdown. Best available codes will be used for these tasks.

2.5.5 Research Program Plan

2.5.5.1 Assessment of Best-Estimate Systems Codes

To assess capabilities of TRAC and RELAP-5 code versions RES has scoped out a comprehensive program involving four national laboratories: BNL, INEL, LASL, and Sandia.

The assessment effort at BNL will emphasize examination of the basic (physical) models in the code through comparisons with test data obtained from a variety of domestic and foreign basic tests and separate effects tests highlighting thermal/hydraulic phenomena that are modeled in the code. This work will also involve testing other correlations/models for the basic processes and appropriate sensitivity studies.

Because of the availability of larger computational resources, the code assessment effort at INEL, LASL, and Sandia will emphasize examination of code performance (Both PWR's and BWR's through comparison of computed results with test data obtained from PWR experimental facilities (LOFT, Semiscale, 2D/3D and Separate Effects) and BWR experimental programs (TLTA and SSTF), as well as applicable foreign data. To prioritize the code assessment effort, emphasis will be place on operational transients, small-break LOCAs, non-LOCA transient, and large-break LOCAs. Activities to achieve the above objectives will be:

1. Assessment of PWR and BWR transient analysis computer code PWR TRAC, BWR TRAC and RELAP 5. In the code assessment process, plant transients that have occurred in operating plants and startup testing will be used to the extent possible. For severe transients where data are not available, operating plants transient data from experimental programs described above will be used to assess codes for transients. Transients that

include multifailures leading to and including degraded core condition, will be included in the code assessment process.

2. Assessment of PWR and BWR small-break computer codes PWR TRAC, BWR TRAC and RELAP5 will be performed, using experimental data from the programs described above. This process will be divided into two major parts: assessment of individual code model using separate effects data and assessment of overall code performance using system experimental data. Acceptance criteria will be developed for individual transients and applied to the code assessment process.

3. A limited assessment of PWR and BWR large-break computer codes PWR TRAC, BWR TRAC and RELAP5 will be performed. This effort will be to assess codes used to perform audit calculations in the licensing process. To the maximum extent possible, the code assessment process will include assessment of analytical models, such as core blockage model, that will also be used in rulemaking procedures.

A comprehensive matrix of test cases against which code results are to be compared will be prepared for the assessment of PWR and BWR systems codes. The listing of the selected cases, the rationale for their selection, and the assessment procedure will be described.

2.5.5.2 Code Applications

Analyses have been and will be performed to assist NRR in resolving licensing issues. Some analyses will also be performed to identify the changes needed in emergency procedures, to evaluate impact of the single-failure criterion, and to assess consequences of multiple failures and operator actions. These analyses will be closely coordinated with NRR to ensure that licensing needs are met.

Past experience has shown that it takes 4 to 6 months to prepare an input deck for a systems code such as RELAP. To avoid long delays in the ability to respond to requests for LWR plant accident analyses, it was decided to compile in advance and store the required plant information in a plant data bank.

The data bank would contain both the basic and the derived information needed for thermal/hydraulic analyses of selected PWR and BWR plants. Information is to be stored in a hierarchic structure that allows access to data that are sequentially more detailed concerning the overall plant, its primary coolant system, secondary coolant system, containment, balance of plant, and all trips and controls.

The plant data bank will be installed and maintained at one or more installations (laboratories) designated by the NRC and will be made accessible via remote terminals. Demonstration of the plant data bank capability and user convenience is scheduled for FY 1982. Complete data for only one plant (Zion) will be entered for this demonstration. Incorporation of data sets for all other plants will be performed by NRR.

2.5.5.3 Results

FY 1982 — Complete independent assessment of TRAC and RELAP-5 codes. Complete plant data bank demonstration. Complete best estimate analyses of overcooling transients, steam line break accidents combined with steam generator tube ruptures and small-break LOCA, pumps on/off consequences, and other accident scenarios specified by NRC.

FY 1983 — Independent assessment schedule will be supplied after detail planning is completed. Code application to analyses requested by NRC. Participate in the domestic and foreign standard problem exercises.

The degree of accuracy of the final LWR systems codes capable of analyzing all accidents and transients, up to initiation and including of severe core damage, established. Because of current uncertainties concerning both the test data availability and code completion schedules for analyses of the severe core damage processes and events, the assessment completion schedules for such analyses are not indicated.

2.6 Fuel Behavior Under Operational Transients

The objective of this program is to provide data and analysis capability to independently assess the reusability of reactor cores after anticipated operational transients (AOT's). The assessment of reusability will be based on the resulting risk to the public and to operating personnel. The objective will be accomplished by:

1. The development and maintenance of computer codes to analyze fuel behavior, predict fuel failure probabilities, and assess the corresponding fission product release after such events.

2. The performance of in-pile and out-of-pile experiments to determine the needed model parameters and material properties for the codes, and to provide the experimental base for code assessment.

The accomplishment of this objective will result in a general reduction of risk both to the public and to plant personnel by providing licensing personnel with the necessary capabilities to properly assess the consequences of such events and, thereby, formulate the appropriate decision on reuse of the core.

Funding for this research will decrease by 60 percent by FY 1983 because of the completion of in-pile testing. Funding for FY 1983 and beyond will be primarily concentrated on inexpensive ex-reactor tests and code assessment and model improvement where required. These codes will also provide basic models required for the Severe Core Damage Analysis Package (SCDAP) defined in Section 2.7.

2.6.1 Regulatory Objective

The regulatory objectives are to analyze the fuel behavior in a licensee's plant design during normal operation and during any postulated accident to determine if fuel and fuel-clad temperatures stay below specified safety and damage limits.

2.6.2 Technical Capabilities Required

Basic studies on clad behavior and pellet/clad interaction (PCI) are a direct result of user (NRR) request numbers RR-NRR-79-28 and RR-NRR-79-17. The FRAPCON code-development effort is a direct result of RES/NRR coordination in defining the kind of model development required by licensing personnel. The NRR staff also plans to use the FRAP-T code more extensively in the near future in their efforts to set licensing limits for anticipated operational transients (AOT).

The data generated from this work and the resulting code refinements will provide licensing personnel with the data and modeling bases required to assess the safety of nuclear reactors during normal and offnormal operation, and provide the necessary modeling bases for analysis of severe fuel damage.

2.6.3 Status of Capabilities

During the emergency core cooling system (ECCS) hearings, it was made very clear that confirmatory research on fuel behavior during LOCA conditions was needed so that the degree of conservatism of the proposed licensing limits could be established. In particular, questions arose on the integrity of the cladding and on the effects of deformed cladding on the core cooling. Because cladding behavior during such postulated events is strongly dependent on temperature, clad oxidation, fuel-stored energy, gap conductance, fuel/clad mechanical interaction, and the overall mechanical properties of the cladding, a comprehensive research program to study these phenomena was begun. All of this work is essentially complete except for some extra fuel/clad interaction studies. The work resulting from the ECCS hearings will be completed in FY 1982. The focus for the program planned for FY 1983-1987 is the result of the NRR request for data on fuel damage resulting from pellet/clad interactions, and the need to maintain advanced fuel behavior analysis capability.

2.6.4 Research Program Objectives

The current objective of this program is to determine experimentally by inpile and expile tests, and subsequently by modeling, fuel behavior in nuclear reactors under anticipated operational transients.

2.6.5 Research Program Plan

The program consists of experimental and analytical studies in three major areas: (1) basic studies of cladding and PCI, (2) inpile tests of fuel rods and the effects of high burnup, and (3) development and assessment of fuel codes.

The basic studies are primarily concerned with pellet/cladding interaction failures induced during power ramps. The studies are conducted both out-of-pile and inpile with modeling performed in parallel.

The current inpile program includes the analysis of the AOT experiments in the NRC-funded Power Burst Facility (PBF) and the steady-state experiments in the Norwegian Halden Reactor to assess inpile fuel rod behavior under both transient and steady-state (for improved stored-energy calculations) conditions.

The best-estimate fuel codes being developed are FRAP-T (fuel rod analysis program - transient) and FRAPCON (steady-state). The FRAP-T code, which incorporates information developed in the experimental program, is used to analyze the behavior of fuel rods during transient events. It is also used for pretest and posttest predictions for the PBF, LOFT, ESSOR, and NRU experimental programs. Moreover, a version of the code has been adapted by NRR for use in licensing calculations. The FRAPCON code, developed for and used by NRR, provides the initial conditions of the fuel (stored energy), based on prior operational conditions before the transient, as input to FRAP-T, which analyzes the actual transient event. Models from both codes will be used in the development of the severe core damage analysis package.

The following is a brief description of the three major programs.

2.6.5.1 Basic Studies of Cladding and Pellet/Cladding Interaction

This work is being done to provide NRR with the technical background to assess the continued use of reactor cores after power transients that could result in a high probability of subsequent fuel rod failures.

Most of this work began in FY 1980 or FY 1981 and is being conducted at three different laboratories: The work will (1) evaluate the out-of-pile stress rupture and low-cycle fatigue behavior of cladding from BWR and PWR plants (begun in FY 1980 at ANL); (2) determine the important parameters required for assessment and completion of the PROFIT model for fuel failures during power ramping (begun by NRR and taken over by RES in FY 1981 at BNWL); and (3) quantify and characterize the PCI failure process via inpile testing (begun in FY 1980 at Studsvik). All this work is in an initial stage and will be completed in FY 1985.

2.6.5.2 Inpile Tests of Fuel Rods

This work is a continuation of basic inpile studies to evaluate fuel behavior under steady-state and transient conditions. The steady-state work in the Halden reactor has been underway since FY 1975. These tests were designed to evaluate the important parameters that determine fuel-stored energy and fission gas release and to define these parameters for FRAPCON, the steady-state fuel code. The program will be essentially completed in FY 1982, except for finishing some post-irradiation examination work that will be completed in FY 1983. Most of the inpile transient experiments are being conducted in the PBF. The test program will continue through FY 1982 and will include tests to simulate such operational transients as BWR turbine trips without bypass. Inpile testing of two operational transient tests will be completed in FY 1981 with final analyses completed in FY 1982.

2.6.5.3 Development and Assessment of Fuel Codes

Work on the development of the transient and steady-state fuel codes (FRAP-T and FRAPCON) began at INEL in FY 1975 under NRC contract. One purpose of the effort is to provide the NRC with up-to-date fuel-behavior-analysis capability for reactor operations. Another purpose is to provide an integrated, easily accessible storage bank of fuel-behavior information in the form of correlation equations and first-principle models derived from past, present, and future fuel-behavior experiments. Independent assessment of these codes is performed using test reactor data and ex-pile separate effects data. In addition, a new assessment program using high-burnup commercial rod data is currently being planned for the steady-state (FRAPCON) code.

All required models are now present in both FRAP-T5 and FRAPCON-1. The final versions of the codes, FRAP-T6 and FRAPCON-2, will contain improved programming techniques and improved links with other codes such as TRAC, RELAP, and COBRA. These versions are scheduled to be completed in FY 1981.

2.6.5.4 Results

Anticipated results, by fiscal year, are:

FY 1981 Time-to-failure versus hoop-stress curves for three lots of cladding in benign environments (ANL). Characterization of Zircaloy tubing in terms of absorbed strain energy versus strain rate, heating rate versus strain rate, and heating rate versus absorbed strain energy (BNWL). Preliminary report from Studsvik on the DEMORAMP PCI study program. Preliminary data from two PBF OPTRAN inpile tests on BWR turbine trips without bypass. Completion and analysis of the LOC-6 test in the PBF (EG&G). Completion and assessment of the FRAPCON-2 and FRAP-T6 steady-state and transient fuel rod codes (EG&G). Completion of the irradiation of IFA's 432, 513, and 527 in the Halden test reactor for use in characterization of fuel stored energy (BNWL).

FY 1982 Time-to-failure versus hoop-stress curves for Zircaloy under iodine, cesium iodide, and zirconium iodide environments (ANL). Completion of data analysis and report on Zircaloy absorbed strain energy to failure for both irradiated and unirradiated cladding (BNWL). Completion of 189 contract B2043 that encompasses steady-state Halden test assemblies; post-irradiation examination and reporting completed (BNWL). Development and documentation of preliminary analysis code for PCI failure predictions (BNWL). Continued maintenance and assessments of FRAP-T6 and FRAPCON-2 (EG&G and BNWL).

FY 1983 Update of PCI analysis model (BNWL). Continued maintenance and assessments of FRAP-T6 and FRAPCON-2 (EG&G and BNWL).

FY 1984 Update of PCI analysis model if needed (PNWL). Completion of commercial rod assessment of FRAPCON-2 using data from high-burnup rods from EPRI and DOE programs. Continued maintenance and improvement, if needed, of FRAP-T6 and FRAPCON-2 (EG&G and BNWL).

FY 1985 Completion of PCI analysis model for use by licensing in prediction of PCI failure probabilities (BNWL). If required

FRAP-T6 and FRAPCON-2 modeling will be improved based on comparison to experimental data (EG&G and BNWL).

FY 1986 Closeout of this program; final updating and assessment of FRAP-T6 and FRAPCON-2 fuel codes.

2.7 Core Damage Beyond LOCA

The accident at TMI-2 has underlined the need for a better understanding of reactor behavior under severe core damage conditions. This research will evaluate (1) the conditions that cause severe fuel damage (2) the formation and coolability of fuel debris beds, and (3) the generation of hydrogen.

The work defined in this program will increase in scope through FY 1985 and is to obtain data for modeling of the behavior of a core that has been damaged (but remains in its location above the support plate) with respect to (1) the coolability of degraded cores, (2) the formation and characterization of debris beds and liquefied fuel within the core, (3) fission-product release and distribution, (4) hydrogen generation and detonability, and (5) postaccident coolant chemistry. Using the above data and models, a computer code for fuel behavior under such conditions will be developed and assessed for coupling with current systems codes such as TRAC and RELAP. Less severe fuel behavior transients are assessed in the research described in Section 2.6. Accidents that involve fuel melting and leaving its original location are discussed in Chapter 5. Data from the research described in this section will be used in the degraded core rulemaking process.

2.7.1 Regulatory Objective

The regulatory objective is to be able to analyze a nuclear reactor core which has lost its original geometry, but is still in its original location, to determine its coolability, physical state, fission-product release potential, hydrogen generation, and effects on coolant chemistry.

2.7.2 Technical Capabilities Required

The planned degraded core cooling rulemaking hearings will require basic data pertaining to the physical and chemical characteristics of severely damaged cores to determine accident mitigation requirements and needed rules to ensure the public safety.

Very little data is currently available on the nature of severely damaged cores. Information is required to determine the coolability of the core, the coolability of various types of fuel/clad debris beds, the nature of the thermochemical reactions that take place at high temperatures, and the extent and transport of the fission products and hydrogen released. Reliable information must be obtained from inpile tests that closely duplicate reactor conditions such as nuclear heat sources (in liquid and solid phases), fission products, and prototypical fuel/clad thermal and chemical reactions.

In addition, the TMI accident demonstrated that in-depth information on the formation, combustion, and handling of hydrogen in accident situations was neither immediately available nor well organized, and in some cases nonexistent. Also, the NRC background in reactor-coolant chemistry applicable to accident situations was shown to be insufficient. In response to these needs, RES began a program of research on hydrogen, and NRR created a Chemical Engineering Branch to provide staff capability in reactor chemistry.

2.7.3 Status of Capabilities

The planned rulemaking activities of the NRC are dependent on an adequate basis for the treatment of fission-product release and transport. The program described in Section 2.6 on fuel behavior under operational transients has produced data that will be used in the development of the Minimum Engineered Safety Features Rule, Phase One. This program addresses the need for such data for Phase Two of that rule wherein more severe accidents are addressed along with the engineered safety features required to render the likelihood of a core-melt accident sufficiently low and to ensure that the fission-product release can be handled efficiently and safely. Mitigation of the consequences of releases accompanying core melt will be dealt with in the Degraded Core Cooling Rule, supported by the severe-accident phenomenology and mitigation program.

During the TMI-2 accident, there was considerable speculation as to the coolability of the core and, therefore, as to what kind of shutdown procedure could be used. There was also considerable speculation about the amount of hydrogen and fission products produced and released, as well as about their physical distribution and transport within the primary, secondary, and containment sections of the reactor building. So that such questions can be answered for possible future events, the information described in Section 2.7.2 is needed as an absolute minimum.

NRR has endorsed the overall FY 1982 program. It is clear from NRR task force reports on TMI-2, from the Kemeny and Rogovin studies, and the NRC Task Action Plan (NUREG-0660) that the planned research is needed. The work described below is based on RES staff understanding of Commission needs and on conclusions reached as a result of meetings with NRR, EPRI, U.S. reactor vendors, and foreign researchers.

2.7.4 Research Program Objectives

The objective of this program is to develop the data and methods necessary to assess the effects of potential core damage in the range beyond that characterized in the current design basis accident but not so severe as that characterized by large amounts of molten fuel. The accident at TMI-2 was within this range. The effects of less severe accidents are assessed by data and methods developed in Section 2.6, on fuel behavior under operational transients. Data and methods for assessment of more severe accidents are developed in Chapter 5 on severe accident phenomenology and mitigation. Since the technical problems are similar in treating fission-product release and transport, this program serves as a link in the consistent development of a regulatory data base spanning the entire range of interest.

2.7.5 Research Program Plan

Most of the work in this program will begin in FY 1981 and will be essentially completed by FY 1987. All programs are currently in the planning and evaluation stage.

So that the projects can be described efficiently, they have been grouped into four general areas: (1) inpile and out-of-pile severe core damage (at least clad melting) experiments, (2) coolant chemistry and hydrogen behavior studies, (3) clad ballooning and flow blockage studies both inpile and out-of-pile, and (4) severe core damage modeling.

2.7.5.1 Severe Core Damage Experiments

This work includes: (1) experiments in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratories (INEL), (2) experiments in the Super Sara loop of the ESSOR reactor in Ispra, Italy, (3) examination of the TMI-2 core, (4) out-of-pile experiments on incipient fuel/clad melting similar to those conducted by Hagen at KfK, West Germany, and, possibly, (5) followup inpile full-length bundle tests in the NRU test reactor in Canada.

The work in the PBF reactor is designed to determine the inpile progression of fuel damage and disintegration in fuel rod bundles for use in the assessment of fuel damage models. The formation, physical characteristics, and behavior of the liquid phase formed between UO_2 and the molten $Zr\text{-}ZrO_2$ will be determined for comparison with data determined in more extensive studies conducted out-of-pile. The physical and chemical characteristics of the resulting debris beds will also be determined and related to the exact details of the quenching procedure and timing. Preliminary test train design and experimental planning began in FY 1980; actual experimental testing is scheduled to begin in FY 1983 and to be completed by late FY 1984 or early FY 1985 for Phase I of the studies. Early results of Phase I data analyses are expected in FY 1983 and program completion is scheduled for FY 1986. Phase II will be planned after a substantial part of the Phase I data is available, and the additional data needed can be determined. Phase II tests should begin in FY 1984 and continue through FY 1987.

The program at the ESSOR reactor will include approximately 15 severe core damage tests some of which will simulate the small-break transient and boiloff that occurred in TMI-2. The program's importance is underlined by two major features: (1) the loop will handle 6-foot-long bundles containing 32 fuel rods, enabling data to be obtained on fairly long, large bundles (PBF can handle only 3-foot-long rods) and (2) an identical parallel loop will be present to simulate the nuclear test with electrically heated rods. The latter capability allows for an accurate determination of the thermal/hydraulic conditions before the nuclear test is performed.

The possible followup inpile full-length bundle tests in NRU in FY 1985-1986 will confirm the extrapolation of the results from the shorter-rod-length tests in PBF and in ESSOR.

The TMI-2 core components will be examined on site and after they are removed from the reactor to characterize the core damage. It is hoped that information can be obtained on the melting temperature, composition gradients, damaged-rod geometry, fission-product distribution, and particle sizes of the damaged fuel. Poolside and hot cell examinations will begin as soon as specimens are available.

The ex-reactor studies of incipient fuel/clad melting will be performed to increase understanding of the kinetics of formation of liquefied fuel, its flow rates and patterns, its chemical composition and composition gradients, and its thermodynamic and physical properties. The tests will use some simulated fuel bundles to investigate bundle effects on disintegration and thermal/hydraulic conditions.

2.7.5.2 Coolant Chemistry and Hydrogen Behavior Studies

The hydrogen program consists of analyses and experiments investigating (1) the generation and transport of hydrogen in reactor accidents, (2) the sampling and analysis of hydrogen, (3) the ignition of hydrogen/steam/ air mixtures and the combustion of hydrogen in containment from deflagrations through detonations, (4) the pressure-time histories of hydrogen combustion events, (5) means of mitigating potential damage from hydrogen combustion, and (6) means of safely handling postaccident hydrogen. A continually updated compendium of hydrogen information which was established in 1980 will incorporate the results of this work.

Hydrogen generation in a variety of reactor accident sequences in various reactors will be studied by calculations with the MARCH code. Transport of hydrogen/steam mixtures in containment will be studied in the program initially using the RALOC computer code when arrangements can be made to obtain it from the German sponsors. Codes to predict the pressure-time histories of combustion events will be developed and tested against the combustion experimental work. Data will also be developed to allow improved prediction of hydrogen generation from the interaction of steam with zinc and organic coating systems in containment.

In the area of reactor chemistry, the new NRR Chemical Engineering Branch is just beginning to establish the role of chemistry in reactor safety, and definition of required chemistry research will be evolving in the near future. There are two areas that will be investigated: (1) the identification of post-accident fission-product species under a variety of postaccident chemical conditions and (2) the possibility of obtaining fission-product "signatures" that can be used to identify different kinds of core damage during accidents.

2.7.5.3 Clad Ballooning and Flow Blockage Studies

This work contains five major experimental programs, all of which focus on the study of clad ballooning and rupture. The ballooning and flow blockage present in a deformed bundle will influence quite strongly the amount and nature of the debris produced in the high-temperature regions of a degraded core in the mid-range of damage-producing accident scenarios. For the same time-temperature scenario, the final damage could range from complete oxidation of all damaged cladding, if the ballooning is extensive and the flow blockage severe, to the formation of the maximum amount of liquefied fuel, if there is little to no ballooning or swelling of the fuel rod.

The MRBT (multi-rod burst test) program, conducted at the Oak Ridge National Laboratory (ORNL), consists of tests of electrically heated fuel rod simulators, 3 feet long, in bundle sizes ranging from single rod to 4 x 4 and 8 x 8. The project was initiated in FY 1974 and is presently planned to be completed in FY 1982.

The Long-Bundle Test (LBT) Program is designed to evaluate the ballooning and burst behavior of full-length, nine-rod, electrically heated, fuel rod bundles inside an array of guard heaters as a function of system pressure, reflood rate, boildown rate, decay-heat simulated power, and thermal/hydraulic parameters. This work is scheduled to begin in FY 1982 at ORNL.

The NRU program, conducted by Battelle Northwest Laboratories in the NRU test loop of the Chalk River, Canada test reactor, consists of five nuclear-heated tests, each containing 32 rod bundles of 12-foot-long PWR-type rods. The tests will be conducted in an environment and for heating rates that simulate the refill and reflood stages of a LOCA. This work will provide the first prototypical inpile confirmation of current licensing practices. Before the ballooning tests, a series of approximately 20 nondestructive, nuclear-heated bundle tests will be conducted to define and determine the thermal/hydraulic conditions and parameters. These tests are scheduled to begin in mid-October

1980. The completion of the NRU ballooning tests, now planned for FY 1982, will probably be delayed until FY 1983 because of budgetary restraints. The final posttest analyses will then be completed some 14 months later, in FY 1984.

The PBF LOCA program, being conducted at INEL, consists of tests of 3-foot-long nuclear-heated single rods under conditions that simulate ballooning and rupture during the blowdown phase of a LOCA. The test matrix has been defined to allow ballooning to occur in the a-Zircaloy, (a+b) Zircaloy, and b-Zircaloy temperature ranges. Five tests of the seven-test program have been completed. The final tests will be conducted in FY 1981 and analyses of the data are scheduled for completion in FY 1982.

The last program in this group consists of six ballooning tests to be conducted in the Super Sara loop as part of the overall ESSOR test program. These tests will be conducted on 6-foot-long bundles containing 32 rods. The tests will complement the PBF and MRBT programs by providing inpile, longer length, and larger bundle data under LOCA blowdown environmental conditions. Because of the parallel loop available in this program, measurements of the actual thermal/ hydraulic conditions and their effect on ballooning will be determined. Initial tests are tentatively scheduled to begin in FY 1983, pending the results of current task force discussions.

2.7.5.4 Modeling of Severe Core Damage

This program is designed to give the NRC the ability to predict and assess the core damage that would result from postulated accidents that produce cladding temperatures exceeding the present design basis of 2200°F up to conditions of large-scale fuel melting. Current plans are to use the results of the severe core damage experiments conducted in the PBF to develop and assess a new code called SCDAP (severe core damage analysis package). It is intended that SCDAP will model such damage on a fuel assembly scale and draw heavily on the current fuel element models developed in Section 2.6 in FRAPCON and FRAP-T, as well as the thermal/ hydraulic models in COBRA, TRAC, and RELAP. Moreover, it was recently learned that a model for fuel/clad melting has been developed at Stuttgart-IKE, West Germany, to model the severe fuel damage incurred in Hagen's experiments at KfK, West Germany. Documentation of this model, called EXMEL, has been obtained and will be implemented where it is deemed necessary. It is also intended that SCDAP be used as a module in modeling corewide damage in codes such as MARCH/CORRAL or an appropriately modified version of the SIMMER-II code. These latter codes are intended to describe gross melting of the core. A program to scope out the needed models and assess the EXMEL code will begin at EG&G in FY 1981 under the current modeling contract, A6050. In FY 1982, new funding will be earmarked for this work. Prior to the completion of a preliminary version of SCDAP in FY 1982, planning for the PBF severe fuel damage tests will be checked by current codes such as RELAP, COBRA, TRUMP, TRAC, and FRAP. These codes can and have supplied accurate predictions of expected fuel temperatures and hydrogen release. Work in this area is expected to continue as long as data are supplied by parallel experimental programs. The program should therefore be completed before or during FY 1987.

2.7.5.5 Results

FY 1981 A report on the "Analysis of Hydrogen Mitigation for Degraded Core Accidents in the Sequoyah Reactor" will be issued in the first half of FY 1981. An experiment plan for hydrogen research will be prepared, relative priorities established, the hydrogen experimental program initiated, and a report of the first test series issued. The second annual compendium of hydrogen information of interest in reactor accidents will be issued. A report on the laboratory hydrogen experiments on fog maintenance and combustion in foams will be issued. Work on a manual to guide operators in strategies for handling hydrogen will be initiated. A survey of iodine aqueous chemistry for accident consequence evaluation will be completed. Laboratory measurements of iodine species present under reactor sump conditions will be completed. A program to investigate the feasibility of using fission-product signatures to distinguish types of fuel failure will be initiated. A planning and design report for the development of the severe core damage computer code (SCDAP) will be issued. The 28 thermal/hydraulic test series in the NRU reactor will be completed. Peak cladding temperatures, turnaround times, and quenching behavior will be determined for unpressurized 12-foot-long fueled bundles during the heatup phase of a LOCA. The initial full-length inpile clad ballooning test in the NRU facility will be conducted and reported on. Flow testing and sectioning of the MRBT 8 x 8 bundle will be completed. The 6 x 6 MRBT bundle tests will be completed and preliminary data analyses reported.

FY 1982 One severe fuel damage test may be conducted, with preliminary data analysis reported. The FITS facility will be modified for hydrogen experiments and test series 1 and 2 will be completed and reported on. Hydrogen tests in subscale FITS facility test series 1 and 2 on dry hydrogen and hydrogen/air/ steam combustion will be reported.

Tests on autoignition and flame characterization will be completed in the steam-hydrogen jet facility. A survey of the aqueous chemistry of tellurium will be completed for accident consequence evaluation. Cells will be developed and tested for high-temperature and high-pressure measurements of iodine species. The feasibility study on fission-product signatures will be completed. Experimental work on hydrogen from galvanized coatings will be completed. A preliminary version of the SCDAP code will be completed. Two NRU clad ballooning tests will be run and reported on. Final data analyses on the MRBT tests will be completed and reported; the MRBT program will be terminated. Program planning and facility design for the Long-Bundle Tests (LBT) will be completed. Final reports will be issued on the ballooning tests conducted in the PBF LOCA series.

FY 1983 One or two severe fuel damage tests will be conducted, with preliminary data analyses performed on one. Fission-product distributions will be determined. Data analyses may be completed and reported on the severe fuel damage test conducted in FY 1982. Preliminary data will be available on the physical, mechanical, and thermodynamic properties of the liquid and solid phases of the reaction products between liquid Zircaloy and UO_2 fuel pellets. Full-length inpile clad ballooning tests in the NRU facility will be completed, and preliminary results will be reported. Two LBT tests will be conducted and flow tested, and preliminary data will be reported. Subscale FITS tests on ignition threshold and reliability and Halon tests will be reported. FITS tests on air/hydrogen/steam burns will be completed and reported, and experiments on detonation will be performed. The VGES trench facility will have been constructed and large-scale testing initiated. Experiments on iodine species present under high-temperature and high-pressure conditions will be completed. Work on instruments and techniques for fission-product signature measurement will be initiated. Experimental work on hydrogen from inorganic zinc coatings will be completed. A production version of the SCDAP severe fuel damage computer code will be completed.

FY 1984 Experiments will be completed on all Phase I severe fuel damage tests conducted in PBF. Fission-product distributions and accident signatures will be reported.

Two Phase II severe fuel damage tests will be conducted in PBF, and preliminary data analyses will be reported. Fission-product distributions will be determined and accident signatures reported.

Initial severe fuel damage tests will be performed in the ESSOR reactor and preliminary data and analysis reports issued. Cleanup experiments will be completed on the properties of the reaction products between liquid Zircaloy and UO_2 fuel pellets. Analyses of results of NRU clad ballooning program will be completed and the program terminated. Three LBT tests will be completed and preliminary test data reported; final data analyses will be reported on the LBT tests conducted in FY 1983. The final advanced version of the SCDAP severe fuel damage code will be completed and assessed. Hydrogen combustion experiments will be performed and reported. A final report on iodine species present at high temperature and pressure will be issued. A final report on hydrogen generation from coatings will be released. Experiments on aqueous chemistry of tellurium will be conducted and reported. Experiments on hydrogen from organic coatings will be completed, final reports issued, and project closed out.

FY 1985 Three Phase II severe fuel damage tests will be conducted in PBF and the preliminary data analyses will be reported. Data analyses will be completed and reported on the Phase I severe fuel damage tests. Three or four severe fuel damage tests will be performed in the ESSOR reactor; data and analysis reports will be issued. Three LBT tests will be completed and preliminary test data reported; final data analyses will be reported on the LBT tests conducted in FY 1984, with comparisons made with the results of the NRU ballooning tests. Hydrogen combustion and detonation experiments will be performed and reported. Studies of the aqueous chemistry of fission products of interest to consequence evaluation will be performed. Fission product signature instrumentation and technique effort will continue. The SCDAP code will be updated, if needed, and kept on maintenance.

FY 1986 Phase II severe fuel damage tests in PBF will be completed and reported. Severe fuel damage testing will continue in the ESSOR reactor. Follow-on full-length severe fuel damage bundle tests in NRU will be performed with available data reported. The examination of the TMI-2 core should be completed with data analyses underway; fission-product distributions will be compared to PBF/ESSOR test data. The final LBT tests will be conducted and the data reported. A final report on the LBT tests will be issued. Hydrogen experiments at large scale will be conducted and reported. Fission-product aqueous chemistry experiments and reports will be completed and closed out. Fission-product signature instrumentation investigations will be completed and the project closed out. The SCDAP code will be updated, if needed, and kept on maintenance.

FY 1987 The ESSOR severe fuel damage program will be completed and final reports issued. Large-scale hydrogen experiments will be conducted. The SCDAP code will be kept on maintenance.

2.8 PBF Operations

This section provides for the operation of the Power Burst Facility (PBF) as a test reactor. The reactor is used to test samples of nuclear fuel under accident conditions as specified by regulatory needs. The end products are test fuel assemblies that have been subjected to these severe conditions and are ready for examination in hot cells to determine the effects on the fuel and the clad material.

2.8.1 Regulatory Objective

This program provides resources for the operation, maintenance, and modification of the PBF. In addition to operational transient (OPTRAN) experiments, a new series of experiments investigating severe fuel damage in accident situations will be initiated and performed during the period through FY 1987. These experiments will involve the release of substantially higher quantities of fission products and quantities of hydrogen than the facility has had to deal with in the past and will entail additional efforts to safely deal with such situations.

2.8.2 Technical Capabilities Required

The PBF is a very flexible facility for the testing of reactor fuels under normal operating and accident conditions. It incorporates a high-temperature, high-pressure loop with an in-pile test section, controlled blowdown system for LOCA investigations, transient power capabilities, and an extensive data-acquisition and recording system. The in-pile test section can accept severe damage experimental bundles consisting of 32 three-foot long fuel rods, safely perform the test, and be decontaminated afterward. The facility performs the experiments described in detail in other programs.

2.8.3 Status of Capabilities

The thermal fuels behavior program (TFBP) experiments in power-cooling mismatch, reactivity-accident insertion, gap conductance, loss-of-coolant accident, irradiation effects, thermocouple effects, LOFT-lead rod, and in other test series have been performed in the PBF. The facility includes a pool-type driver core with a central inpile tube connected to a high-pressure, high-temperature circulating loop that can simulate normal operating conditions and accident/transient conditions that can exist in PWR and BWR nuclear power plants. The inpile loop has undergone major modification to provide rapid depressurization capability for LOCA testing.

2.8.4 Research Program Objectives

This program provides for:

1. Design, procurement, installation, testing, and safety analysis of modifications to the PBF facility;
2. Facility operating staff;
3. Preventive and corrective maintenance of the PBF systems and components;
4. Utilities, supplies, and spare parts;
5. Health physics, safety, and quality assurance services;
6. Data system operation, data reduction, and reporting;
7. Preparation of the instrumentation systems for use in the experiments;
8. Documentation of operating procedures, standard practices, and manuals;
9. Plant and reactor engineering activities for the facility; and
10. Instrument, electrical, and control engineering for plant systems.

2.8.5 Research Program Plan

The facility has been serving the thermal fuels behavior program for about 6 years. It will be mainly devoted to support of the severe core damage test program in the coming years. Consideration is now being given to safety upgrading of PBF to comply with TMI lessons learned.

3. LOFT

The LOFT (Loss of Fluid Test) Reactor is the only reactor facility able to perform loss of coolant accident (LOCA) system transients and anticipated transients without scram, using nuclear heating. A revised test matrix is being developed, addressing the highest priority concerns of the LOFT Special Review Group, RES and NRR, while allowing the possibility of preplanned contingency tests. Based on RES estimates that two of the four contingency tests will be performed, completion of experiments is estimated to occur by January 1983. If all four of the contingency tests are required, the completion of the experimental program is estimated to occur by June 1983. Once the test matrix has been agreed to by NRR and RES Office Directors, changes in the test description or LOFT program will occur only under extraordinary circumstances via the authority of the office director. The LOFT program plans include maintaining a minimum technical support staff for 2 years after the last test is completed to keep the ability to restore the plant to operation quickly. Plans to decontaminate and decommission LOFT are scheduled to be carried out after the 2-year standby period, that is, in FY 1985. Final plans for LOFT are awaiting Commission approval. The remaining LOFT test program will emphasize three areas of concern.

1. Operational transients will provide:

    a. Data for assessment of computer codes. The behavior of large PWRs under certain transient conditions has been questioned by NRR. The data from the LOFT operational transient tests will be used to evaluate the ability of vendors to predict plant performance and necessary operator intervention.

    b. Conditions that must be dealt with correctly by the reactor operators to prevent an accident escalation leading to degraded core conditions.

c. A test bed for human-machine interface program advances in improved operator displays and accident mitigation techniques to the extent that LOFT is the most cost effective place to do the work.

2. The LOCA data base will be completed by running an intermediate break size test and by defining the lower bounds of the fuel clad ballooning response with pressurized fuel. A large-break LOCA test run with fuel pressurized to best estimate conditions is not expected to result in significant ballooning or bursting of the cladding.

3. Prepressurized fuel tests will provide data for the NRC's degraded core cooling rulemaking and 10 CFR Appendix K. LOFT is in a unique position to provide supporting information on the behavior of damaged fuel due to the typicality of its 15x15 rod fuel assemblies.

Following the completion of the test program, the facility will be placed in standby mode for 2 years during which a technical staff cadre will be maintained. At the end of this time, the facility will be recommended for decommissioning if it has not been taken over by another funding source.

3.1 Regulatory Objective

The safety analysis codes used by the reactor licensing staff should be compared to actual physical phenomena that are expected to occur in licensed nuclear plants to the greatest degree possible. Whenever possible, data will be obtained from accidents which occur in operating reactors. However lack of instrumentation and the limited types of accidents experienced, requires that some accident situations be performed in test facilities. Full advantage is being taken in the final 2 years of LOFT's planned operation to perform tests in which system effects are expected to predominate with the objective of providing to the NRR staff test data against which their safety codes can be tested and improved.

3.2 Technical Capabilities Required

The LOFT test program was formulated during FY 1970-74 and represented a skeletal structure of various test series. Subsequently, as testing has been done, commercial experience acquired, and licensing issues have arisen, the definition of each test series has been filled in. Throughout this procedure, NRR has been deeply involved. In fact, the entire program through FY 1982 essentially consists of tests specified by NRR, with input from the nuclear industry.

Furthermore, as new needs have been identified--especially following the TMI-2 accident--the LOFT program has been extended to include the assessment and development of instrumentation for identifying accident conditions and the assistance of operators in plant recovery. In addition, an advanced operator diagnostic and display system is now being tested and improved using actual accident conditions encountered during LOFT tests.

The NRR office has prepared a Three Mile Island task action plan, NUREG-0660, in which the LOFT results relate to 22 different action items. These action items and the manner in which LOFT relates to them are described in Appendix A. They may be categorized into the following six areas:

1. training and qualifications of operating personnel,
2. simulator use and development,
3. operating procedures,
4. a vent path for noncondensibles,
5. fault-diagnostics system to train operators in the control and mitigation of accidents which may lead to severe core damage, and
6. steps in the evolution of a severely damaged core.

The LOFT program is beginning to provide results which contribute significantly to areas 1, 2, 3, and 5. Before terminating in mid-FY 1983, it will contribute considerably to area 6, steps in the evaluation of a severely damaged core.

3.3 Status of Capability

In 1962, the AEC initiated a test program at the INEL to study the consequences of a commercial light water reactor (LWR) LOCA that was caused by a major primary pipe rupture. This was principally a study of the release, transport, and deposition of fission products resulting from a reactor core meltdown that was called the LOFT-U program. It was limited to a few nonnuclear blowdowns simulating pipe rupture and a reactor-core meltdown. The facility included a pressurized water reactor (PWR) experimental test facility.

In the mid-1960s, the size of commercial nuclear power plants increased to about 1000 MWe (3000 MWt). As a result, additional engineered safety features were developed to prevent releases of radioactivity to the environment should a LOCA and a subsequent reactor core meltdown occur. In 1967, reactor plant designs began to include new safety features, such as emergency core cooling systems (ECCS), for core flooding and prevention of core meltdown in the event of a LOCA. The AEC redirected the LOFT program to include studies of the new safety design features for large reactor systems and cores, rather than a reactor core meltdown. In 1969, redesign of LOFT was begun so that the system would model as closely as possible the conditions present in the primary coolant system, core, and ECCS of a large PWR. Since LOFT was an essentially complete design, and a partially completed facility, significant design changes and almost a new start were necessary to satisfy the new requirements. Special care was taken in scaling LOFT flow areas, including the break area, pumps, and system volumes to insure that the chronology of events in a LOCA and the duration of those events would be representative of a commerical PWR. In addition to insuring thermo-hydraulic similarity between LOFT and a commercial PWR, the fuel used in LOFT was designed to be typical of the commercial PWR fuel of the early 1970s. Since that time, the fuel pellet specifications have been updated to reflect the changes in requirements resulting from NRC fuel densification studies.

When ERDA and NRC were formed in 1975, the responsibilities for completion of the LOFT facility and operations were assigned to ERDA, and the responsibility for the experimental program was assigned to NRC. The first test series, which in some respects represented commissioning tests, was nonnuclear and was run from 1976 to 1978.

The first nuclear test in LOFT as a fully functional nuclear plant was run on December 9, 1978. Since then, the facility was considered complete and the NRC has been responsible for funding the program.

3.3.1 LOFT Facility Description

The major component of LOFT is the mobile test assembly (MTA), which consists of a 55-thermal-megawatt reactor system and reactor cooling system mounted on a double-width rail-transportable test dolly. The MTA is installed in a high-pressure containment building that has auxiliary systems for reactor plant support and a contiguous underground control room.

The reactor core is 5-1/2 feet long, and 2 feet in diameter, and contains 1300 PWR-type fuel rods. The core is instrumented with high-temperature thermocouples and other specially developed instrumentation to measure temperatures, flow, pressures, and coolant levels inside the reactor vessel.

The reactor coolant system has one active, heat-dissipating, operating loop that models the three unbroken loops of a four-loop plant, as well as a special blowdown loop that can simulate a ruptured loop in a large PWR. The blowdown loop contains special quick opening valves that simulate rupture of a large reactor coolant pipe in a commercial nuclear steam supply system. This blowdown loop discharges reactor coolant into a suppression tank which is pressure and temperature controlled to simulate the containment pressure conditions calculated to exist following a large PWR LOCA. The suppression tank is connected to equipment for processing any fission products released during the experiments.

The LOFT facility's ECCS has the same basic components a large PWR ECCS has and is designed to provide a similar performance. Three systems for emergency-coolant injection are provided: (1) gas pressurized water-filled accumulators that can inject a large volume of water into the reactor system quickly; (2) high-pressure injection pumps that can provide a small flow of high-pressure coolant for small breaks; and (3) low-pressure injection pumps that can provide large volumes of water for core cooling after an experiment or for core cooling in the highly unlikely event of a major primary system rupture. The primary coolant system and ECCS are extensively instrumented, and ECCS injection points and flow rates are easily varied for experimental purposes.

Since early 1980, an operator diagnostic and display system has operated in the LOFT control room. This system is being used to develop fault diagnostics systems, improved procedures for accident situations, and improved display and interpretive techniques, while using the advice from reactor operators and the experience of actual accident conditions.

3.3.2 The Program

FY 1976, FY 1977 and FY 1978

From 1976 to 1978, a four-test series of large break loss-of-coolant tests was conducted before a nuclear core was installed. These tests provided (1) important information on the delivery of emergency core cooling water to the core region, (2) operating experience needed prior to beginning operation in the nuclear mode, and (3) important experience with the specialized test instrumentation. The nuclear core was then installed, a power-range test program completed, and a final zero-power test was run, with the core shut down, to measure the magnitude of the blowdown forces on the core.

FY 1979, FY 1980, and FY 1981

Due to the fact that the prime regulatory concern was with the large double-ended cold-leg break, the first nuclear series in the LOFT test program involved the double-end cold-leg break with power as the main parameter. Small break and anticipated transient series were scheduled for a later time. However, by May 1979, the first two large-break tests had been run - the first at two-thirds, the second at full commercial core power density - and the results were more favorable than expected (both in LOFT and when extrapolated to a commercial plant). Since the TMI-2 accident had just occurred, the NRC decided that the small break and transient series needed immediate attention, and the LOFT program was therefore rearranged.

Since May 1979, LOFT has run:

7 small breaks to study:

- effect of break size (1-inch and 4-inch pipes)
- effect of break location (broken loop, intact loop, stuck open PORV)
- effectiveness of each heat sink (steam generator, break, ECC injection, primary make-up and let-down)
- effect on liquid inventory of primary pump operation (on or off),

4 operational transients involving

- loss of feedwater
- loss of primary coolant flow
- loss of steam load
- excessive steam load increase,

and 1 core uncovery test which followed a small break and allowed the fuel clad temperature to rise to 700°F. This provided data on core cooling in an uncovered situation, needed for planning future core damage tests.

The program plan from September 1980 on is shown in Table 3-1 and is subject to minor changes based on final negotiations between RES and NRR. The expected achievements and their applicability are discussed in Section 3.5. By the end of FY 1981, the following additional tests will be done.

1. A loss of feedwater with delayed scram (L9-1) leading to an empty steam generator secondary side (as occurred at TMI-2) (L3-3). The Westinghouse recovery procedure (locked open PORV) will first be tried. Then, return of the secondary side feedwater will test a different recovery procedure. Will provide data on reflux cooling and its transition to steam-water, then water, natural circulation, needed for code assessment, and will provide a test of different procedures for recovery from a lost secondary side.

Table 3-1 Planned LOFT Test Sequence and Target Dates as of September 1980

(The current schedule is being reassessed using criteria developed for the Long Term Planning Strategy. While the Short Term Test Schedule is not expected to change, long term test plans may be modified as a result of this agreement.)

TEST ID	TARGET DATE	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
L3-5	9-27-80	50	35	Small break (2.5%) intact loop cold leg--pumps off.
L3-5A	9-27-80	Add on to L3-5		Investigate primary system recovery utilizing steam generator.
L6-2	10-10-80	37	25	Loss-of-power to primary coolant pumps.
L6-1	10-12-80	37	25	Loss-of-steam load (closure of MSIV's).
L6-3	10-14-80	37	25	Excess load increase (cooldown transient).
L3-6	12-1-80	50	35	Small break (2.5%) intact loop cold leg--pumps on. Pumps tripped at end of experiment to measure water remaining.
L8-1	12-1-80	Add on to L3-6		Core uncovery without ECC at low decay heat level.
L9-1	4-8-81	50	35	Loss of all feedwater (multiple failures) with delayed scram; PPS setpoints representative of LPWR (PORV challenged). Mild ATWS. S/G dries out.
L3-3	4-8-81	Add on to L9-1		PORV locked open (W recovery procedure). Final recovery by flooding S/G.
CV Leak Test	7/81			Required test of containment leak integrity.
L6-7	7/81	50	65	Turbine trip, stuck pressurizer spray, stuck atmosphere dump spray, simulates Arkansas Nuclear One startup incident causing rapid cooldown.
L9-2	7/81	Add on to L6-7		Rapid cold water accident, upper plenum voiding, simulating St. Lucie-1 incident.

Table 3-1 Planned LOFT Test Sequence and Target Date as of September 1980 (continued)

TEST ID	TARGET DATE	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
L5-1	10-81	50	65	Intermediate size break (accumulator line). Determine if large break and small break models continue to predict intermediate break results. Also check out liquid level device.
L8-2	10/81	Add on to L5-1		Core uncovery at high decay heat level. Reflood with degraded ECC capability. May be the same as L5-1.
LA-10	12/81	50	65	Operational transient contingency test.
Whole core changeout				Fl center bundle at 350 psi (BOL). Large peaking factor if only CB changed.
L2-5/LA-1	4/82	50	65	Worst prototypical hydraulic conditions in core. Off-normal conditions. Investigate fuel behavior at BOL fuel pressure (no fuel damage expected).
L5-2	6/82	50	65	Intermediate size break contingent upon need.
L9-3	9/82	50	65	ATWS with loss-of-feed water.
L3-9	12/82	50	65	Small break LOCA contingent upon needs for code assessment.
L9-4	3/83	50	65	ATWS
L2-6	6/83	50	65	Same as L2-5 with 700 psi fuel pressure (EOL). Fuel damage and fission product release expected.

2. A piggy-back cool down experiment, the first portion to simulate the Arkansas Nuclear One cool down incident (turbine trip, stuck open pressurizer spray, stuck open atmospheric dump), the second to simulate the rapid cool down with upper head voiding reported in St. Lucie-1. Results will permit a scaling comparison between LOFT and commercial plants, and will provide data needed to assess codes used to predict cold water accidents.

3. A small break (4-inch) with the coolant pumps left running, as in the L3-6 test. The break location and other initial conditions will be significantly changed from L3-6 to provide confirmation of the validity of code improvements made in the wake of the L3-6 results.

4. Piggy-backed on to the L3-3 test and the intermediate break will be two tests in which the core will be uncoverd and which are intended to provide experience and data needed to plan subsequent fuel and core damage tests, while not at this time damaging the fuel.

5. During the entire period, studies will continue of procedures to stabilize and recover a plant in accident conditions, the interpretation and improvement of instrumentation needed to do this, and the advanced operator display and diagnostic systems.

Results from the FY 1980 and FY 1981 test are currently being reported to the licensing staff in the form of RILs as listed in Table 3-2. The subject matter in these RILs has been organized to provide insight into phenomena observed, and later to provide an interpretation of the results in terms of commercial PWRs.

3.4 Research Program Objectives

The objective of LOFT is to establish conditions in a nuclear reactor which are characteristic of accidents postulated for a large PWR so that methods can be developed and tested for analytical description, for accident recognition, and manual and automatic plant stabilization and recovery.

Table 3-2

LOFT RILs Schedule

Content	Date
1. L1 Series of nonnuclear LOCEs	Completed 7/78
2. LOCEs L2-2 and L2-3 with Zion Relationship	Completed 8/79
3. Anticipated Transient L6-5, L6-1, L6-2, L6-3	3/81
4. LOCEs, L3-0, L3-1, L3-2, and L3-7 phenomena evaluation	4/81
5. LOCEs, L3-1, L3-2, and L3-7 and Zion Relationship with RELAP4/MOD7	6/81
6. Small break prediction capabilities	5/81
7. LOCEs L3-5 and L3-6 pumps off-on results	6/81
8. Plant recovery from a complete loss of feedwater and steam generator dryout	11/81
9. Loss-of-Feedwater with and without delayed scram (L6-5, L9-1) extrapolated to large PWR behavior	11/81
10. Experience with the LOFT technical support center	3/81
11. The simulation of nuclear fuel-rod behavior with electically-heated rods	5/81
12. A comparison of information normally available to reactor operators with information from measurements during LOFT accident simulations	6/81
13. Achievements and capabilities of the LOFT Man-Machine Program	6/81

The specific goals of this program are:

1. acquiring of data for the assessment and improvement of computer codes intended to predict the behavior of PWRs under a wide variety of accident conditions;

2. understanding of the behavior of PWRs under accident conditions and of the operator actions needed to stabilize and recover the plant;

3. interpreting and improving plant instrumentation needed to identify accident conditions and to assist the operator in recovering the plant; and

4. developing an advanced operator display system and testing at a PWR under actual accident conditions.

3.5 Research Program Plan

During FY 1982, one of the final two large-break LOCAs will be performed along with two intermediate break tests, an operational transient test, and an ATWS test with loss of feedwater as the initiating event. The large-break LOCA will demonstrate the effects of fuel prepressurization and loss of offsite power, which are the two major outstanding questions in the double-ended cold-leg break addressed by Appendix K of 10 CFR 50 and will support NRC's degraded core cooling rulemaking activities.

These tests are not expected to generate failed fuel since the fuel conditions are to be "best estimate" values. Testing under these conditions will provide data on the margins inherent in the ECCS evaluation rule.

FY 1983

During FY 1983, the final small-break test, ATWS test, and large-break test will be conducted. The final large-break test will be performed with pressurized fuel run at off-normal conditions to produce clad ballooning and burst. The ballooning and burst pattern will be examined to study the amount of flow blockage that occurs and to assess the needs for cooling a degraded core. The information is to support the degraded core cooling rule due FY 1983.

Two tests each are being planned for in FY 1982 and in FY 1983 as contingency tests. These tests will be planned and preanalyzed in case they are needed. If some or all are not needed, the schedule will be compressed accordingly.

Operation of LOFT as a test facility will be terminated around mid-FY 1983, after which the facility will be placed in a standby mode for 2 years with a minimum technical support staff. If another funding source has not been obtained by the end of this period, the facility will be recommended for decommissioning.

Analysis of the previous year's test results will continue through FY 1984. Funding through FY 1986 will cover (1) standby maintenance and decommissioning of the facility, (2) completing analyses and reporting of results, (3) disposing of all fuel, (4) reimbursing DOE for all test requirements, and (5) the costs of destructive examination of the fuel.

The test program described above is shown in Table 3-1.

3.5.1 Results

The test program at the LOFT facility has consisted of four distinct types of tests which have yielded results of significance to the licensing process. (1) large pipe break loss-of-coolant experiments without nuclear power, (2) large pipe break loss-of-coolant experiments with nuclear power, (3) small pipe break experiments with nuclear power, and, (4) anticipated transients with nuclear power.

The large-break nonnuclear tests were initiated at approximately full reactor coolant pressure and temperature. They were designed to study the effectiveness of the emergency core cooling (ECC) systems in delivering coolant to the core and thus confirm certain conservatisms in the NRC's ECC rule. The results showed that the ECC water is delivered more quickly to the core, more reactor coolant remains in the core region, and less ECC water flows from the break than is predicted by codes based on the ECC rule. These tests also demonstrated that ECC systems, which are identical to those in commercial reactors, work as expected, and the tests provided invaluable experience in handling a nuclear reactor under accident conditions.

The large break nuclear tests showed that early in the accident, even before the ECC system is actuated, the core receives a flow of water that significantly lowers the temperature of the fuel leaving it more readily coolable than expected when the ECC water arrives. When the models of the predictive procedures were corrected to properly represent the observed thermal hydraulics, this unexpected behavior was then predicted. Computer codes, which were so modified, then predicted that the same cooling phenomenon would occur in a commercial reactor operating under the same conditions, subjected to the same large-break accident.

The small-break test series is designed to study

1. plant behavior during pressure variations: increasing, decreasing and held up at some intermediate value;

2. effectiveness of different available heat sinks;

3. different break sizes and locations; and

4. primary coolant pump: on and off during small breaks.

The first test was done only 2 months after the TMI-2 accident. It was initiated at full pressure and temperature but, in order to obtain much needed data as early as possible, the core was not generating power. This test was a study of a stuck open PORV and, therefore, demonstrated some of the phenomena that occurred early in the TMI-2 accident. The second test, simulating a

4-inch break, caused a slow, continuous depressurization and eventual activation of the ECC systems to refill the plant before core uncovery. The third test, simulating a 1-inch break, caused a very slow pressure reduction with stabilization at an intermediate value. Operator intervention then brought the pressure down sufficiently to actuate the ECC systems, and the plant was brought under control without uncovering the core. Of special interest to NRC was the indication that the steam generator transitioned from liquid natural circulation to liquid-vapor natural circulation, and possibly reflux condensation with no evidence of instability. Another important discovery was that flow paths, which bypass the core and which exist in all PWRs, can have an important influence over the course of a small-break accident.

The fourth small-break test examined the effectiveness of various heat sinks available to PWRs. Preliminary study of these results suggest that for larger small breaks (4-inch pipe and above), the break flow is sufficient to carry away all decay heat, while for smaller breaks (1-inch pipe), the steam generator is the dominant heat sink and, therefore, the pressure of the primary system closely follows that of the secondary system.

The next two small break tests were identical except for primary coolant pump operation. In the first, L3-5, the pumps were shut down; in the second, L3-6, they were left running. Compa on of liquid inventories showed that the pumps should be shut down early in a small-break accident to reduce the rate of loss of water through the break. In a piggy-back test with L3-5, the liquid level was lowered far below the hot leg and the flow conditions in the hot leg were monitored. All flow measurements showed that even then flow continued in the positive direction through the intact loops, and no significant reflux cooling could be established.

In a piggy-back test with L3-6, the core was entirely uncovered, and the clad temperature allowed to rise by 200°F before the ECCS was initiated. This provided important data for estimating heatup rates for uncovered cores. The remaining program is now planned to combine tests in this way.

APPENDIX A

ACTION PLAN TASKS ADDRESSED BY LOFT PROGRAM

When the TMI Action Plan was drafted, LOFT was committed to two action items. These are:

1. Task I. D. 5 Control Room Design, Decision Group A, Priority 2

Objective: Improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them.

Schedule: Improved display and diagnostics will be installed in LOFT* by May 1980 .... Adequacy of disturbance analysis methods will be verified by December 1982.

Status: On schedule. Operating experience on the display system has been acquired during three SBLOCAs and one transient. Initial diagnostic experience is expected in late 1980.

2. Task II. E. 2 Emergency Core Cooling System, Decision Group A, Priority 1

Objective: Decrease reliance on the emergency core cooling system (ECCS) for other than loss-of-coolant accidents; ensure that the ECCS design-basis reliability and performance are consistent with operational experience; reach better technical understanding of ECCS performance; and ensure that the uncertainties associated with the prediction of ECCS performance are properly treated in small-break evaluations.

Schedule: In the LOFT facility, six transient and small break tests will be conducted in FY 1980 and in FY 1981.

Status: In FY 1980, two small breaks and one transient have been completed, and before the end of this FY two more small breaks and one more transient are scheduled. In FY 1981, the schedule includes three small breaks, (including the pumps on and pumps off cases urgently needed by NRC) three transients and, an intermediate break. The latter has been scheduled in anticipation that the results of the small break tests and analysis will raise questions about the behavior of somewhat larger break LOCAs.

Already, the tests have provided important information on natural circulation, transitions between heat transfer regimes in the steam generator, and the importance of core flow bypass on the course of a small break LOCA.

The augmented operator display equipment referred to above in Task I.D.5 was operated during a small break test and a transient test.

Although not specifically mentioned in other parts of the Action Plan, LOFT results could be useful to NRR in completing several other tasks. These are listed below.

1. Task I. A. 1.3 Control Room Staffing Requirements

Objective: The objective is to increase the capability of the shift crews in the control room to operate the facility in a safe and competent manner by assuring that a proper number of individuals with the proper qualifications and fitness are on shift at all times.

LOFT Program Contribution

LOFT is studying the requirements of the number and qualifications of operators to be present in the control room.

2. Task I. A. 2 Training and Qualifications of Operating Personnel

Objective: Improve the capability of operators and supervisors to understand and control complex reactor transients and accidents, and improve the general capability of an operations organization to respond rapidly and effectively to upset conditions. Increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capability to perform their duties.

LOFT Program Contribution

In order to help operators understand reactor transients and accidents and to teach them rapid and effective responses to upset conditions, industry needs good information on these subjects. LOFT is engaged in a series of LOCAs, transients, and so forth, which provide experience in reactor behavior and data to develop and assess analytical tools. The LOFT results should, therefore, be useful in meeting the objective of this task.

3. Task I. A. 4 Simulator Use and Development

Objective: The objective is to establish and sustain a high level of realism in the training and retraining of operators, including dealing with complex transients involving multiple permutations and combinations of failures and errors. Another overall objective is to improve operators' diagnostic capability and general knowledge of nuclear power plant systems.

LOFT Program Contribution

Referring to the objective of improving operators' diagnostic capability, the LOFT man-machine program will test this capability using an integral nuclear system designed to undergo a wide range of accident scenarios. This will provide a proving ground for diagnostic techniques and for determining the usefulness of such techniques to operators.

4. Task I. C. Operating Procedures

Objective: Improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration maintenance, testing, and surveillance will be included. A principal part of this work is to improve procedures for dealing with abnormal conditions and emergencies by improving the delineation of symptoms, events, and plant conditions that identify emergency or off-normal situations that confront the operators and, once identified, to assure consistency with operator training.

LOFT Program Contribution

The LOFT Program supports this objective and specifically NRC actions I.C.1, I.C.5, I.C.6, I.C.7, I.C.8 and I.C.9 in three different ways:

a. LOFT research in the area of Augmented Operator Capability includes a subtask designed to study and improve the procedures for dealing with abnormal conditions. Operators and psychologists work together (a) to determine the incidents for which procedures are actually needed, (b) to improve the logic of these procedures, (c) to design clear display arrangements for these procedures which incorporate a fault diagnostics system (d) to test out and improve upon these displays and diagnostic system under actual accident conditions. This task was initiated in late 1979 and is now beginning to provide results which are being made available to utilities, vendors, the NRC, and the foreign nuclear industry.

b. At the request of NRR, in late 1980, LOFT will run small break LOCA simulations (a) with the primary coolant pumps off and (b) with the pumps on. The results are to be used in determining the procedures which utilities must follow in the event of a small-break accident. These and other LOFT small break tests are to be used in the assessment of small break audit calculations which NRR has required the vendors to perform. LOFT shall study NSSS vendor review of low-power and power-ascension test and emergency procedures.

c. The series of small breaks, transient and steam generator tube rupture tests being performed through FY 1984 will help to define the emergency procedures for such events in commercial plants. They will confirm or improve our understanding of the effectiveness of heat sinks such as the steam generator, the make-up system and break flow, and the synergetic relations between these components as system conditions change and as operators take various actions - an understanding which is required by NRC if it is to responsibly perform many of the actions described in Action I.C.

5. Task I. D. Control Room Design

Objective: Improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them.

LOFT Program Contribution

Item I.D.1 A developmental model of an Operational Diagnostics and Display System (ODDS) was installed in the LOFT Main Control Room (MCR) to explore the potentials of an intelligent man-machine interface, and to help define the research and development needed to transfer this concept to the MCRs of licensed LWRs. This ODDS system consists of a computer which receives instrument data from the LOFT data acquisition computers. The ODDS computer can exercise computational programs in real time and display results rapidly enough to be of direct use to the reactor operator. A set of color-graphic CRT terminals which display the computer output is the most visible part of the ADC program. These displays reduce the necessity for the operator to sample many channels to form an opinion about a particular piece of information. Currently a key board provides for reactor operator input and more flexible input techniques are being studied, such as touch panel, voice, and light pen.

6. Task II. B. Consideration of Degraded or Melted Cores in Safety Review

Objective: Enhance public safety and reduce individual and societal risk by developing and implementing a phased program to include, in safety reviews, consideration of core degradation and melting beyond the design basis. Included in the four program phases are phase (1) short-and medium-term actions for scoping and implementation and phase (3) research programs and design studies to develop additional needed information.

LOFT Program Contribution

Item II.B.1 In line with NRC's action on reactor coolant system vents to deal with noncondensibles, LOFT is studying a design for a vent path from the upper head. In combination, the pressurizer connections to the primary system would be changed to permit positive level indication in the pressurizer.

Item II.B.4 NRC is requiring operator training in the use of safety and non-safety systems to control and mitigate accidents which may lead to severe core damage. The fault diagnostics system referred to in I.C. as part of the man-machine task is well under way. By late 1981, we expect to have an on-line system, tested out under actual accident conditions, available to support this type of training.

Item II.B.5 NRC is designing a research program to study the behavior of severely damaged fuel. In a recent meeting called to help define this program, it was recognized that eventually large scale experiments would be required to evaluate overall severe core damage behavior. LOFT is ideally suited for this purpose. Since the smaller scale tests are scheduled for the FY 1982-84 period, LOFT could complete its current test program and prepare for severe core damage tests to follow on directly from the smaller scale tests in FY 1984-85, prior to terminating the project.

7. Task II. F. Instrumentation and Controls

Objective: Provide instrumentation to monitor plant variables and systems during and following an accident.

LOFT Program Contribution

Item II.F.1 Additional accident monitoring instrumentation. In preparation for fuel damage tests, LOFT is installing a primary coolant fission product monitoring system.

Item II.F.2 Identification of and recovery from conditions leading to inadequate core cooling. The need for reactor vessel liquid level measurement was clearly illustrated at TMI. LOFT uses discrete conductivity probes to evaluate the liquid level in the core, upper plenum, lower plenum and the downcomer. These measurements have been extremely valuable in evaluating LOFT tests. Other studies performed at LOFT to measure liquid level in the reactor vessel include (a) cladding and fluid thermocouples, (b) self-powered neutron detectors, (c) differential pressure transducers, and (d) intermediate range ion chambers. In cooperation with EPRI, a nonintrusive neutron detection device is being installed on the LOFT reactor vessel. In addition, LOFT is collaborating with an NRC-industry group to install and test other promising techniques.

8. Task II.F.3 Instruments for Monitoring Accident Conditions

Since the LOFT instrument system was designed to monitor system conditions during a variety of accident situations, it is capable of measurements under the full range of possibilities of coolant qualities, pressures, flows, and temperatures. The experience with these instruments is readily available to NRC and industry.

Task III.A.1.2 Upgrade License Emergency Support Facilities

The Technical Support Center, recommended to be established in all nuclear plants, was established in LOFT. It has been functioning during the last five accident simulations, and the experience gained is now being prepared for a report to NRC.

4. PLANT OPERATIONAL SAFETY

The primary objective of plant operational safety is to provide information to the staff on how to prevent or lessen the occurrence of mechanical, structural, human, and electrical failures in nuclear plant systems, as well as how to evaluate the consequences of accidents as related to these areas. The issues addressed include human-factors engineering, plant system behavior during transients, instrumentation and control, mechanical, structural and materials behavior, seismic effects, and design verification of important safety equipment. This research program will also support rulemaking underway for degraded cores.

Many of the problems in nuclear power plants that the Commission and the regulatory staff have had to deal with on a day-to-day basis have been in the area of plant operations. These include:

1. human-machine interface,
2. inability to determine the reactor vessel liquid level during an accident,
3. the ability of the containment to withstand overloads,
4. improper calculations of piping stresses due to earthquakes which caused the shutdown of five plants,
5. pipe cracking and leaking,
6. steam generator tube failures,
7. challenges to safety systems caused by operator or maintenance errors,
8. abnormal thermal and mechanical stresses on the reactor pressure vessel, and
9. validity and applicability of the ASME Boiler and Pressure Vessel Code.
10. the ability of inservice inspection techniques to reliably detected and evaluate flaws.

This program unit is subdivided to address the following issues:

1. Human Factors Engineering

   The research in this area explores:

   a. what should the role of the operating crew be to ensure safe operation;
   b. to what extent should computer technology be introduced into reactor operations to supplement the crew functions,

c. whether licensee proposed modifications may really improve the performance and safety of the whole system, and

d. how management affects the safety of plant operations.

2. Instrumentation and Electrical Systems

This research will determine:

a. if qualification practices used by industry for electrical equipment used in nuclear plants are an adequate test of the ability to withstand accidents;

b. if fire protection standards are adequate;

c. if instrumentation and electrical design will function as expected; and

d. if new reactor vessel liquid level instrumentation will function under accident conditions as claimed.

3. Plant Systems Behavior

The effort described in this section evaluates

a. the design basis for engineered safety safely features and plant systems,

b. control system failure modes and system interactions, and

c. evaluates diagnostic methods and techniques for the inspection staff.

4. Behavior of Mechanical Components

a. Addresses the performance of piping, vessels, pumps, valves, snubbers and mechanical equipment under operating and accident conditions;

b. verifies computer codes used to predict mechanical behavior;

c. assesses ASME Pressure Vessel Code (sections III and XI; and

d. determines the behavior of mechanical systems under earthquake and other accident conditions.

5. Behavior of Structural Components - Structural safety research is aimed at developing methods and procedures verified by test results that are suitable for NRC staff use to verify structural calculations, estimate structural failure probabilities, and determine margins of safety. The goal is to establish improved structural design criteria to reduce risk and enhance safety.

6. Behavior of Materials - This research area develops analytical methods for prediction of crack initiation and arrest in vessels under thermal shock, and evaluates leaking or breaking or pipes under a variety of loading cases, as well as stress corrosion cracking in normal operation. The integrity of steam generator tubing is studied. More reliable detection and faster and more accurate ways to evaluate flaws are being developed for vessels, piping and steam generators.

Most of these areas will support several rulemaking efforts underway for degraded cores and minimum engineered safety feature. The individual programs are described in more detail in the eight sections that follow. Most of them are new programs started in the last 1-3 years in response to vital licensing needs.

4.1 Human-Machine interface

The research conducted on the human-machine interface improves NRC's basic understanding of the impact of humans on reactor safety and of the factors that affect the performance of the human-machine system. Among the current and anticipated licensing issues to which this information applies are these:

1. What is the role of the reactor operating crew and what should it be to ensure safe operation?
2. What are the corresponding psychological, educational, and training requirements?
3. To what extent and how soon must recognized human engineering deficiencies be corrected?
4. To what extent and at what rate should computer technology be introduced into reactor operations?
5. Do proposed modifications really improve the performance and safety of the total system? How much improvement and at what cost?
6. How does management affect the safety of plant operations? What can be done if improvements are deemed necessary?

Through analyses and controlled experiments, this program attempts to provide an objective technical basis for clarifying and resolving these issues.

4.1.1 Regulatory Objective

The objective of this program is to generate and communicate research results supporting the development of regulatory positions on improvements in the human-machine interface. The ultimate regulatory objective toward which this research aims is the reduction of the human contribution to risk to an acceptably low level.

4.1.2 Technical Capabilities Required

The technical capabilities required have been generally stated as follows:

1. The TMI-2 Lessons Learned Task Force Report (NUREG-0585) recommended that RES:

   a. Establish a program to evaluate the safety effectiveness of designs of disturbance-analysis systems, and
   b. Formulate a program to establish a technical basis for distinctive licensing criteria for manual and automatic operations, including a specific examination of the role of the operator.

2. The Task Force also recommended many other actions for the NRC staff which indicated the need for additional research (e.g., instrumentation requirements, safety system monitoring requirements, revisions to procedures).

3. The TMI Action Plan (NUREG-0660) describes needed research on:

   a. Improved control room instrumentation (operator-process communication and disturbance-analysis systems), and
   b. Improvements in simulator capabilities.

Specific research requests and endorsements are as follows:

1. In Research Request SD-78-2, the Office of Standards Development asked RES to initiate research to identify the time required for operators to perform safety-related actions during an accident.

2. In Request for Endorsement of Research RES-80-15, RES requested and subsequently received endorsement for programs on operator role definition, plant status monitoring, and computerized display and diagnostic systems.

3. In Request for Endorsement of Research RES-80-13, RES requested and subsequently received endorsement for programs to analyze the operator's impact on initiating, aggravating, and mitigating severe-accident sequences.

4. In Research Request NRR-80-7, the Office of Nuclear Reactor Regulation asked RES to perform a detailed task analysis for control room operating crews.

In its "Comments on the NRC Safety Research Program Budget for Fiscal Year 1982" (NUREG-0699), the Advisory Committee on Reactor Safeguards characterized the program described below as one that "will provide data and information which will assist NRR in strengthening and revising license requirements to improve safety and reduce risks. These programs are considered important to plant operational safety and should be continued and expanded within reasonable manpower and equipment resources."

4.1.3 Status of Capabilities

Designers of nuclear power plants take special care to develop inherently safe machines exhibiting stable and reliable operation. Beyond that, the first line of reliance for safely operating a nuclear power plant is the reactor operating crew. Their proper performance helps reduce the number of challenges to plant safety systems and helps maximize effectiveness in mitigating the challenges that inevitably occur.

For many events, the machine is designed so that certain critical functions (e.g., reactor shutdown) are performed automatically when unsafe conditions are approached. Functions requiring manual intervention or cognizance are prescribed by written procedures covering a spectrum of stylized accidents. In the event that safety systems fail and procedures do not apply, the operators are also the last line of reliance; i.e., they are the key components in contingency decisions and accident-mitigation strategy if the design basis for the plant is exceeded.

The TMI-2 accident and subsequent investigations identified important deficiencies in the ways humans were being factored into the performance of the total system. The NRC required immediate action on many correctable deficiencies of particular concern. Other deficiencies need additional consideration before longer-term policies are developed and implemented. It is especially clear that much greater attention must be paid to integrating human factors into all aspects of reactor technology, from initial conception through final implementation and operation. Research can augment the technical basis for clarifying and deciding these issues in the future.

4.1.4 Research Program Objectives

The objectives of this program are to generate and communicate a sound technical basis upon which to decide regulatory issues related to the human-machine interface. In this context, results include:

1. Generation of new concepts,
2. Suggestions for system functional requirements,
3. Assessments of technical feasibility,
4. Confirmations of the adequacy of technical approaches used to develop and demonstrate new concepts,
5. Estimates of risk reduction potential,
6. Proposals for regulatory requirements,
7. Recommendations on implementing improved systems and procedures,
8. Analytical models and analyses of human performance within the human-machine system, and

9. Literature and technology surveys.

4.1.5 Research Program Plan

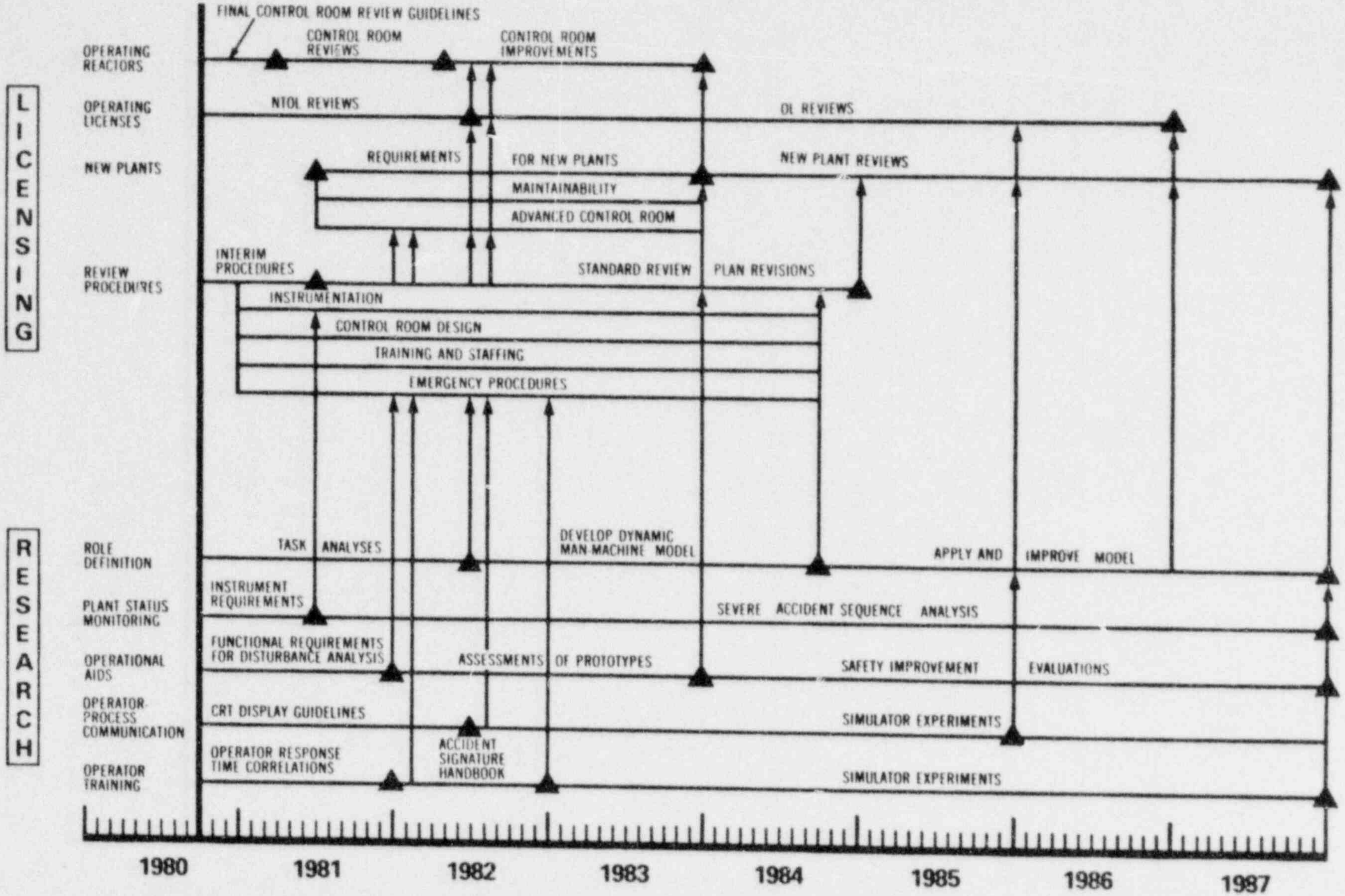
The following describes those research tasks intended to achieve the overall objectives of the human-machine interface program. The approximate schedule for accomplishing these tasks and their relationship to licensing are shown in Figure 4-1.

4.1.5.1 Role Definition

Studies will be performed to better understand the roles assigned to humans in currently operating systems and the impact of humans on safe operation. These studies will be expanded to cover future designs. For example, studies will be performed to determine optimum and acceptable allocations of human and automatic functions within the overall system. The results will be used to identify the necessity, feasibility, and effectiveness of potential improvements in staffing, training, procedures, equipment, and interface design. The projects planned are:

1. Task analysis: The goal is to define more clearly the roles of operating, maintenance, operating support, and management personnel.

   Tasks: Task analyses will be performed to delineate clearly the roles of individuals as a function of plant operating mode and to identify the staffing and information requirements necessary to complete each defined task. Sources of operating and maintenance errors will be identified and methods evaluated for reducing such errors. The staff will use the results as a benchmark against which to judge the adequacy of similar efforts required of the industry.



The relationships among licensing and research. Proper sequencing of research allows timely introduction of technical results into regulatory decisionmaking.

FIGURE 4-1

2. Manual vs automatic action: The goal is to develop guidance and criteria for allocating functions between the operator and the machine in support of draft standard ANSI N-660.

Tasks: Current studies analyzing the time required for operators to take prescribed safety-related actions will be completed. This will include a calibration of data from simulator experiments against operational experience. Recommendations as to the degree of automation that should accompany the activation and operation of engineered safety features and to improving operator training and information displays will be developed.

Later efforts will critically evaluate the advantages and disadvantages that varying degrees of automation have had in crew performance in nuclear power plants and other process industries. Factors to be addressed include:

a. How automation affects operator motivation, vigilance, and attitude,
b. How automation, with the operator serving as monitor and backup, affects qualifications and training requirements and other job-related factors; and
c. The degree to which it is desirable, from an operator performance standpoint, to preserve the role of control manipulation for the operator and use automation as a backup.

The effects on crew performance of specific proposals for altering the manual-automatic functional allocation will be evaluated as such proposals become available.

3. Human-machine dynamic modeling: The goal is to provide and apply analytical techniques for assessing the performance of the integral human-machine system.

Tasks: Early efforts will investigate the feasibility and utility of dynamic, quantitative models in nuclear applications (e.g., modeling the control room operator as an information processor and as an active component in a control loop.) If initial results are promising, an analytical model of the human-machine system will be developed, verified, and used to identify and evaluate tradeoffs in performance and design.

4. Accident sequence analysis: The goal is to analyze the operator's impact on safety during a broad spectrum of accident sequences.

   Tasks: Best-estimate state-of-the-art codes will be applied to analyze the physical response of specific plants to a spectrum of accident sequences, including those having a relatively high probability of core damage. Particular emphasis will be placed on assessing the perceptions of the operator, his needs for information, the alternative actions he might take given various combinations of component failures, and the consequences of those actions. The analyses will consider the various recovery options available and the necessary timing of such actions. Results provide input to tasks on plant status monitoring, operator training, and simulator capabilities.

4.1.5.2 Plant Status Monitoring

Current studies to develop and analyze systematically the requirements for plant monitoring will be completed. The starting point was the definition and description of accident sequences that have a high probability of leading to core damage. These efforts supplement activities by the regulatory staff to develop and implement positions related to status monitoring (e.g., Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"; Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"; definition of the plant safety parameter display system; and capabilities at onsite and offsite technical support centers). Specific projects are:

1. Instrumentation to follow the course of an accident: The goal is to provide the operating crew with the necessary information to indicate unambiguously the status of the plant.

Tasks: Results of accident sequence analyses will be used to determine the need, range, and justification for instrumentation to monitor physical parameters (e.g., temperatures, flows, radiation levels) during an accident. Accident signatures uniquely characteristic of the plant condition will be developed for diagnostic and corrective-action aids. The means to effectively present this information to the operator will be evaluated and recommendations made.

2. Status monitoring of engineered safety features: The goal is to provide the operating crew with timely warnings of potentially unsafe conditions.

   Tasks: Important accident sequences are analyzed to identify the need and justification for improved monitoring of the operability status of engineered safety features on standby (e.g., valve alignments, power supplies). The safety significance of system, subsystem, and component outages and of combinations of such outages will be reviewed to suggest monitoring requirements. The analysis will indicate the relative significance of these outages. The means by which the approach to limiting conditions for operation may be monitored and annunciated automatically will be evaluated and recommendations made.

4.1.5.3 Operational Aids

Studies will be performed to identify and evaluate the validity of pertinent methodologies used in computerized display and diagnostic systems. Other operational aids such as safety displays and operating procedures are to be examined. The findings help the regulatory staff determine the need for and nature of requirements for such systems. These include:

1. Disturbance analysis systems: The goal is to recommend functional requirements for computerized systems capable of detecting and diagnosing the cause of a disturbance. Also, the adequacy of technical approaches used by the industry in developing and demonstrating such systems will be confirmed.

Tasks: The technical adequacy of the various approaches used to diagnose disturbances will be evaluated. These approaches include alarm pattern recognition, noise monitoring and diagnostics, parameter monitoring and comparison to setpoints, plant system logic models, and dynamic modeling. Logic models of plant systems and their interactions in response to abnormal conditions will be evaluated. The relationship of these systems to be operator's monitoring, control, and decision-making functions will be examined.

Preliminary functional requirements will be proposed, followed by refinements based on a more in-depth review of information needs and the state of the art. Of particular interest are the feasibility and effectiveness of applying diagnostic systems to the whole plant and their potential for detecting adverse interactions among systems. The effectiveness of prototype systems installed in operating power plants will be assessed.

2. Advanced disturbance analysis systems: This task calls for recommendations regarding functional requirements for computerized systems capable of recommending actions to be taken by the operator in the event of a disturbance. The adequacy of technical approaches used by the industry in developing and demonstrating such systems will be confirmed.

   Tasks: The feasibility of such systems has been demonstrated by the Halden Reactor Project with reactor simulators. As with diagnosis, the concept may rely strongly on the completeness and quality of a priori systems analysis. Research will examine the methods to generate and verify these analyses and, in particular, examine the consequences of errors in the analysis. Functional requirements for such systems will be recommended. Reliability goals for computer hardware and software will be assessed and recommendations made. The need for the operator to be able to independently verify computer output and to have an independent backup capability will be reviewed. The effectiveness of prototype systems installed in operating power plants will be assessed. Participation in the Halden Reactor Project will be continued.

3. Accident response procedures: The goal of this effort is to enhance the quality and utility of written procedures.

   Tasks: Analyze important accident sequences to identify the need for revisions in content of existing procedures and guidelines. Human factors experts will review the guidelines and recommend changes where necessary. In addition, areas in which research will support improvements in the guidelines will be identified. Research will identify and test methods for evaluating procedures in regard to operator comprehension, operator acceptability, and procedure readability. This effort will support regulatory staff reviews of plant procedures. Simulator experiments will be performed and operating experience reviewed to identify the manner and extent to which written procedures are actually used under emergency conditions. Finally, experiments will be designed and conducted to assess the effectiveness of changes (e.g., symptom-oriented procedures) resulting from earlier programs.

4.1.5.4 Operator-Process Communication

Studies are being performed to define improvements that might be needed so that the operator may better communicate with the plant process. Particular emphasis is on control room design. The findings will help the regulatory staff determine the most appropriate staffing characteristics and assess the value of operational aids and improvements in control room design.

1. Data presentation: The goal is to improve the ways that data are collected, stored, and presented to the operator.

   Tasks: A computerized graphic display system has been installed at the LOFT facility. Recent emphasis has been on using risk insights and human engineering principles to generate improved video formats and other displays of safety-related information. Future efforts will emphasize controlled experiments that gather data on improvements in operator performance achieved through various display schemes. Participation in the Halden Reactor Project will be continued. Recommendations will be made regarding better use of computer technology, graphics, and audio displays.

2. Control room modifications: The goal is to validate the effectiveness of proposed modifications in instrumentation and control panels.

   Tasks: Experiments will be designed and conducted under simulated conditions to test the performance of operators subjected to changes in instrumentation and control panels. Of particular interest is the potential for introducing additional confusion or negative transfer of training under stress. The results should yield insights as to the best ways to correct recognized human engineering deficiencies.

3. Performance under stress: The goal is to lessen the stress on plant personnel during abnormal conditions and to improve their performance under unavoidable stress.

   Tasks: Control room operations are to be reviewed to identify potential problems (e.g., tracking) encountered by operators and shift technical advisors in stressful situations, and improvements for alleviating these problems are to be recommended (e.g., in data display, training, assignment of responsibility).

4. Operator acceptance: The goal is to identify and alleviate potential problems related to the use of computerized aids by reactor operators.

   Tasks: Related experience in nuclear and nonnuclear applications will be identified and reviewed, and the relative merits of various design and implementation approaches will be assessed. Methods for optimizing operator acceptance will be recommended.

4.1.5.5 Operator Training

Studies aimed at improving the quality of training, including more effective use of simulators, will be performed. The findings will help the regulatory staff assess the adequacy of training and requalification programs.

1. Simulator capabilities: The goal is to improve the utility of simulators in training operators.

   Tasks: Accident sequences in WASH-1400 and subsequent risk analyses will be reviewed to identify those combinations of equipment failure and operator error that should be reproducible by simulators. Advanced codes will be used to calculate the physical response of plant systems during these conditions. A handbook of accident signatures will be prepared to be used in extending the capabilities of reactor simulators. The degree of accuracy with which abnormal conditions such as two-phase flow and core degradation must be simulated in order to achieve training objectives will be assessed. Besides the safety significance of accident sequences, other factors that affect the selection of training exercises will be reviewed systematically and recommendations made.

2. Education and training requirements: The goal is to develop validated education and training requirements for members of the operating crew.

   Tasks: The results of task analyses for operating crew members will be related to existing and proposed requirements. In particular, the extent to which such education and training optimize the crew's ability to control offnormal conditions will be considered. A review of what constitutes equivalency for various levels of formal education will be performed.

3. Operator examinations: The goals are to evaluate the current examination process, to find more efficient and valid examination techniques, and to explore what alternatives should be considered in requalification examinations.

   Tasks: The examination process will be critically evaluated for the purpose of validating that the techniques employed are achieving the objectives sought. Studies will attempt to establish the degree to which the process correlates with and predicts on-the-job performance. Alternatives to the current process will be identified and evaluated.

4.1.5.6 Interfacing Tasks

The following programs provide important interfaces with the human-machine interface program.

1. LOCA and Transient Research: Within this program are developed the best-estimate thermal/hydraulic system codes used to analyze accident sequences. This program also aids improvement in simulator capabilities to the extent that such improvements rely on advanced codes.

2. LOFT: The computerized graphic display system installed in LOFT in 1980 has been the central facility used to develop and evaluate CRT displays. The proposed program assumes the continued availability of this or comparable equipment after the anticipated shutdown of the LOFT reactor in FY 1983. As time progresses, the display system's dependence on real-time data from the LOFT reactor is decreasing in favor of input from stored data and simulation. Should these capabilities be lost, the ability to meet the objectives described on schedule could be adversely affected.

3. Severe Accident Phenomena and Mitigation Research: Results from the accident sequence analyses generated as part of the human-machine interface program will be combined with phenomenological insights to augment the technical basis for rulemaking proceedings dealing with events beyond the current design basis.

4. Systems and Reliability Analysis: Research in this program emphasizes the quantification of human performance and the impact of human error on risk. Operational experience is reviewed to determine human error rates in operations and maintenance. The factors that shape human performance, such as stress and fatigue, are identified and incorporated into quantitative, predictive models used in risk analyses. Results from this program therefore provide guidance to the Human-Machine Interface program by indicating which accident sequences merit more detailed analysis and which improvements in human performance are effective contributors to risk reduction.

4.1.5.7 Results

Anticipated results by fiscal year are as follows:

FY 1983 Provide a handbook of accident signatures for simulator upgrade and operator training; validate education and training requirements for control room personnel; complete task analyses for operations support and maintenance personnel; document assessment of safety parameter display systems, and incorporate results into standard review plans.

FY 1984 Validate education and training requirements for operations support and maintenance personnel; develop dynamic model of the human-machine system using available data; identify and evaluate the consequences of increased automation; provide recommendations on the effectiveness of the operator examination process and alternatives.

FY 1985 Provide definitive criteria for manual and automatic functional allocation; document assessments of prototype disturbance analysis and surveillance systems at domestic and foreign reactors; provide a review of the impacts of symptom-oriented procedures on plant safety.

FY 1986 Incorporate data from prototype assessments into standard review plans for computerized diagnostic systems.

FY 1987 Complete the verification of dynamic human-machine performance model and bring it into routine use.

4.2 Instrumentation and Electrical Systems

The research described in this section covers instrumentation and electrical systems having a direct effect on plant safety system functions and also a strong influence on the safe operation of the plant by providing data to the plant operators that will influence actions and decisions bearing on safety. Current efforts on the development of equipment qualification procedures will develop testing procedures that can be used for evaluating Loss of Coolant Accidents (LOCA) and main steam line break (MSLB) equipment qualification tests. NRR, SD and IE will be provided with information on how radiation levels and environmental testing signature requirements influence test results. This will provide NRR with information to determine whether existing vendor testing requirements for the qualification testing of nuclear power plant safety components and systems provide the desired level of assurance that they will operate correctly in the event of a serious accident. This information will also support NRR's current review of licensees qualification of safety related equipment and IE's independent verification testing program.

The fire protection research program has already provided data for evaluating the need for cable tray separation and use of fire barriers to prevent the spread of cable fires. This work is being extended by Commission Order CLI-80-21 dated 5/27/80 to provide a fire replication test of full-scale cable tray areas from plants selected on the basis of having licensed configurations. These tests will assess the adequacy of licensing requirements. Other research on fire protection will provide NRR with licensing requirements for fire detector selection and location and the effectiveness of various fire suppression agents in controlling and extinguishing fires. A balanced and sound fire protection requirement requires early detection of fires in essential cable tray areas, sound design of the cables, tray location and separation to preclude the loss of redundant safety circuits and provision for automatically and effectively extinguishing fires when they occur. The fire protection research program will provide licensing with data to prepare licensee requirements covering these issues.

Current efforts in instrumentation are concerned with evaluating liquid level measurement concepts being proposed for determining liquid levels and the presence of steam void in the reactor vessel, pressurizer and primary loop of reactors. The TMI accident and St. Lucie steam bubble problem all point to the need for better liquid level instrumentation to provide the operator with a clear picture of what is going on in the primary system of a nuclear power plant during a loss-of-coolant accident. The research program will assess proposed liquid level instrument concepts, provide NRR with an assessment of vendor and licensee proposed systems and identify concepts for appropriate development by DOE or the nuclear industry.

Other research efforts in the area of instrumentation planned for the future relate to the behavior of instrumentation essential to the safe shutdown and operation of systems required to function following an accident. Many of the instruments at TMI failed to operate at the time of the accident and were inadequate in range for the operator to know what was going on. Both the issue of instrument performance and environmental qualification will be addressed in the planned reasearch program to develop and provide NRR with data to be used to assess safety requirements for instruments to follow the course of an accident.

4.2.1 Regulatory Objective

The objectives of the instrumentation and electrical program include (1) evaluating safety criteria and requirements for safety-related instrumentation and electrical equipment and (2) assessing the vulnerability of the equipment to failures. The category of safety-related instrumentation is currently considered to be broader than it was before the TMI accident. This section complements Section 4.3 on plant systems behavior.

4.2.2 Technical Capabilities Required

Research is needed to support upgrading the requirements that provide the various levels of defense. This can be done by studying qualification guidelines and equipment failures and their effect on the performance of safety functions, as well as the effect of design deficiencies on the system performance. The results of the research can be used to revise licensing requirements in those areas where the defense in depth of a safety function is insufficient. Previous research following the Browns Ferry Fire focused on Regulatory Guides 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," and 1.120, "Fire Protection Guidelines for Nuclear Power Plants." It is expected that the expanded research effort planned in this area in the 5-year plan will focus on regulatory guides such as the latest version of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environ Conditions During and Following an Accident," and new regulatory requirements.

There is a need for research on new and improved instrumentation in response to new regulatory requirements and improved systems that are being proposed by the nuclear industry, as well as for the development and evaluation of concepts not yet a part of nuclear power plant instrumentation and control systems. For example, tests are to be conducted in FY 1981 at the LOFT and Semiscale facilities on both vendor and NRC-developed instruments for measuring liquid level in the reactor pressure vessel and pressurizer of a PWR. These tests include liquid level measurement systems which measure the differential pressure head of the liquid, heated thermocouples devices, external-vessel neutron-flux measurements, and ultrasonic liquid-level sensors. This work is being performed to support the NRR Core Performance Branch in its assessment of licensee-submitted instrumentation plans called for by the NRC Action Plan (NUREG-0660).

4.2.3 Status of Capabilities

NRC licensing requirements are based on the premise that sufficient defense in depth will ensure that the required safety of a nuclear power plant is maintained despite all credible accidents and equipment failures. This premise has resulted in specific design requirements, qualification tests, and protective measures for safety-related instrumentation and electrical systems.

Nonetheless, the results of operating experience and research have shown that there are certain weaknesses in some of the levels of defense specified for safety systems. Also, the TMI-2 accident has shown that there is a need for unambiguous instrumentation for the plant operators. This need for improved instrumentation was explicitly identified in TMI study reports.

The Fire Protection Evaluation program was initiated in FY 1975 to answer specific questions with regard to the cable-tray-separation criteria used in regulatory guides. The program was later expanded to include other aspects of fire protection. The results to date include the following:

1. Cable-tray-separation criteria were evaluated and recommended changes were made.

2. Cable-tray barriers and cable tray fire-retardant coatings were evaluated as they relate to typical design-basis fires.

3. A cable-tray fire propagation model was developed and experimentally verified.

4. Preliminary data were obtained on the effectiveness of ceramic-blanket barriers and water sprinklers to protect vertical cable trays exposed to a typical design-basis fire.

5. Preliminary data were obtained on the effectiveness of Halon 1301 by establishing the concentration and soak time required to suppress a deep-seated cable-tray fire.

The qualification testing evaluation program has provided the following inputs to NRC's regulatory program:

1. A preliminary determination was made of the synergistic effects of simultaneous radiation and steam testing for specific safety-related materials.

2. An aging methodology relating single and combined temperature and radiation environments has been developed and verified with a limited number of cable-insulation materials.

3. A preliminary determination of the adequacy of currently used radiation-test simulators was made.

4. A preliminary list of TMI-2 instrumentation and electrical equipment that NRC suggests be removed for examination and test has been compiled.

4.2.4 Research Program Objectives

4.2.4.1 Qualification-Testing Evaluation

The specific objectives for the qualification-testing evaluation portion of the research program are:

1. To provide an independent assessment of LOCA and main stream line break qualification (MSLB) testing procedures and the design verification of Class IE systems potentially sensitive to LOCA-caused, nonrandom multiple failures;

2. To provide a model that can be used to simulate the natural aging process of representative Class IE materials by accelerated aging methods; and

3. To determine the nuclear source term composition of the environment resulting from a design basis LOCA and to evaluate the effects on Class IE components.

4.2.4.2 Fire Protection Evaluation

The specific objectives for the fire protection system evaluation portion of the research program are to provide independent data needed in support of NRC fire protection requirements, or to modify them as appropriate concerning:

1. Cable and cable tray configuration designs;
2. Fire-suppression and fire-detection systems; and
3. Other fire protection equipment.

4.2.4.3 Improved Plant Instrumentation

The specific objective of the improved plant instrumentation program is to assess plant instrumentation concepts to determine whether they will perform adequately in nuclear operation and in the environment accompanying a nuclear accident so that NRR is provided with sufficient information for licensing them. This development may require, in the case of new unproven instrumentation, development and evaluation up to a point where the safety benefits and feasibility have been demonstrated so that future licensing needs for information can be anticipated.

4.2.4.4 Design Verification Evaluation

The specific objective of the design verification evaluation portion of the research program is to assess the adequacy of currently used safety-related instrumentation and electrical components to perform their safety-related function during accidents. Generic deficiencies will be identified and recommendations that will lead to improved equipment designs will be provided to NRR and SD.

4.2.4.5 Plant Electrical Systems Evaluation

The specific objective of the plant electrical system evaluation portion of the research program is to provide an assessment of the plant electrical system performance with credible component failures, loss of offsite power and offnormal load and electrical powerline conditions.

4.2.5 Research Program Plan

The qualification-testing evaluation and fire protection evaluation programs were undertaken by the Division of Reactor Safety Research in 1975. These programs have been periodically redirected since then to accommodate specific needs identified by the Offices of SD, IE, and NRR. The fire protection program is expected to run through FY 1987 as new user needs are identified in connection with the implementation of NRC fire protection guidance (such as that given in the NRR Branch Technical Position 9.5-1) and the new fire protection rule (Appendix R) issued by the Commission (effective October 1980). The ongoing NRR/IE evaluation of licensee qualification of safety related equipment will generate a need for continuing research on qualification-testing evaluation procedures and requirements. Details of these research programs are given below. The new liquid level instrumentation evaluation program began in FY 1979 immediately after TMI by redirecting NRC contractors working on LOCA experimental research instrumentation programs that were near completion.

4.2.5.1 Qualification-Testing Evaluation

The following specific task areas comprise the qualification-testing evaluation program:

1. A LOCA QTE test facility that is capable of conducting simultaneous steam-and-radiation LOCA qualification tests was constructed in FY 1979-1980 under this program. This facility will be maintained and upgraded throughout the 5-year program to reflect changing requirements (such as providing a full superheat capability).

2. LOCA test methods are being evaluated to ensure that the qualification tests conducted by the suppliers of electrical equipment to licensees adequately simulate the accident environment and that the failure criteria used relate to the ability of equipment to perform the required safety function. It is also important to ensure that the test specification is complete enough to guarantee that results will be repeatable at different facilities.

3. A method for accelerated aging of safety-related materials and components is being developed. This method will result in guidelines for the preconditioning of equipment that is to be tested under design basis accidents such as a LOCA. Current NRC specifications require that qualification testing of safety-related equipment be conducted on equipment that has been aged to simulate any deterioration that might result over the expected life of the equipment. Testing of aged equipment makes it more likely that nonrandom failures resulting from design basis accidents can be prevented.

4. A preliminary study was completed in 1980 to evaluate radiation simulator adequacy for qualification testing of class 1 safety-related electrical

equipment. The LOCA radiation source, in terms of the magnitude, the rate, the spectra, and particle type, based on fission product release fractions specified in the NRC Regulatory Guide 1.89 was evaluated. It was found that the source "hypothesized" in the regulatory guide more nearly represents conservative unterminated LOCA conditions. More realistic "best-estimate" LOCA radiation signatures will be developed in the future, based on accident assumptions being developed in the degraded core research effort and the probabilistic accident sequence analysis studies now in progress. Realistic fission product release assumptions as a function of time and containment geometry for the accidents studied should lead to more realistic source assumptions in the qualification of equipment.

Another part of this study compared the damage to class IE equipment from commonly used radiation simulators (cobalt and cesium gamma radiation emitters) to that anticipated from a LOCA source made up of fission product gamma and beta emitters. Here too conservative assumptions are made in the qualification testing, based on treating beta radiation on a gamma equivalent basis. Since the beta irradiation tends to be stopped in the outer layers of electrical insulators, its treatment on a uniform volumetric basis will exaggerate the damage. This work is to be continued with experiments using beta irradiators to develop a more realistic equivalence for beta and gamma radiation damage to commonly used safety-related materials. Future work should consider the equivalence of beta/gamma and neutron/gamma ratios on bulk degradation, charge breakdown transients and permanent effects to electrical insulators.

5. Selected TMI-2 instrumentation and electrical equipment will be removed and examined in FY 1983-1985 to provide feedback to the current equipment qualification program. In close cooperation with the Department of Energy (DOE) and industry, a limited set of safety-related instrumentation and electrical equipment will be chosen for removal from TMI-2 if adequate historical and performance data are available. The equipment will be examined and tested, if possible, to obtain a better understanding of the performance of equipment that has been qualified to existing standards and guides, used in normal operation, and followed by exposure to the

specific accident environment of TMI-2. Damage and failure modes will be identified by these examinations and by tests with additional theoretical and statistical support, where appropriate. This assessment will provide another measure of the adequacy of existing standards and guides.

4.2.5.2 Fire Protection Evaluation

The following specific task areas make up the fire protection evaluation program:

1. Full-scale replication tests mandated by the Commissioners in Order CLI-80-21 will be conducted using exact mockups of portions of operating power plants to verify that fire protection systems--designed to NRC regulations and approved by the licensing staff--are effective in guaranteeing plant safety during a design basis fire. These tests are to start in FY 1981, with a final report scheduled for FY 1983.

2. Penetration fire test methods will be evaluated in FY 1980-1982 to ensure that fire stops that are qualified--using current testing practices--guarantee a 3-hour fire separation between redundant safety divisions when exposed to a design basis fire as required by NRR Branch Technical Position 9.5-1. Preliminary small-scale testing has identified furnace pressure and simulation of excess fuel on the fire side of the penetration fire stop as important test parameters that may require modification. Additional larger-scale tests will be conducted in FY 1982-1984 to verify these conclusions, as well as to check other aspects of the fire stop test that could not be studied with small-scale testing.

3. Fire-suppression systems are being evaluated to determine their effectiveness in suppressing a deep-seated cable tray fire. Testing with halon, carbon dioxide, and water will be completed in FY 1982.

4. Previous cable tray test data are being analyzed in an attempt to characterize the development of the fire as a function of time. These data will be used to develop a predictive model for use in performing and reviewing

a fire hazards analysis (which is currently a regulatory requirement). It is intended that these data, along with a limited amount of full-scale test data, will provide a basis for a design of fire confinement and suppression systems. These data will also be used in research to support the development of fire detection system guidelines.

5. A test method that can be used for installed fire and smoke detection systems will be developed. Although considerable data exist regarding fire and smoke detectors under laboratory conditions, there are only limited test or analytical data for these detectors when they are used in typical nuclear power plant configurations and in responding to fires anticipated in nuclear power plants. This program consists of two parts: (1) detector performance in response to anticipated design basis fires will be obtained in typical nuclear power plant configurations and (2) a tracer-gas test method will be developed for inplace testing, including the required correlation between the detector response to a design basis fire and the tracer-gas concentration at the location of the installed detector. This work will be conducted in FY 1981-1983.

4.2.5.3 Improved Plant Instrumentation

A program on improved plant instrumentation began in FY 1980, following the TMI-2 accident, in response to the need for better information to guide the operator during abnormal occurrences and accidents. Initial research is directed toward assessing and improving techniques to measure the two-phase level in the reactor pressure vessel (RPV), pressurizer, and components of the primary loop.

The following specific task areas make up the improved plant instrumentation program:

1. Improved PWR invessel liquid-level detectors are being evaluated;

2. Instrumentation systems that have the capability for self-verification and self-calibration will be evaluated; and

3. Instrumentation requirements and qualification thereof for following the course of an accident will be assessed (Regulatory Guide 1.97).

Work on testing and evaluating improved liquid-level sensors for measuring the level under two-phase conditions in the RPV and pressurizer of PWRs will be completed in FY 1982. The testing of various liquid-level sensor concepts in BDHT, LOFT, and Semiscale under small- and large-break LOCA conditions will be completed in FY 1982.

A program to assess the capability and qualification of the instrumentation proposed by licensees in response to Regulatory Guide 1.97 to follow the course of a nuclear plant accident will begin in FY 1981. Survival and performance will be tested over appropriate ranges and containment atmospheric conditions in FY 1982-1984.

4.2.5.4 Other new areas of research to be initiated in the period FY 1981-FY 1983 include:

1. Design verification evaluation,

2. Plant electrical systems evaluation, and

3. Contribution of high electrical disturbances to the failure modes of safety grade electrical equipment and protection requirements.

Details of these proposed programs follow:

1. <u>Design Verification Evaluation</u>

   The following specific task areas comprise the design verification evaluation:

   a. Lists of categories of generic instrumentation and electrical components designs covered by Regulatory Guides 1.89, 1.97, and 1.120 are being developed from plants in operation and under construction.

b. A priority list of these generic designs will be identified by:

(1) Examination of licensee event reports (LERs) and operating experience using maintenance records for large failure rates and problems;

(2) Evaluation of data from qualification test reports;

(3) Identification of equipment that must function in the most hostile environments; and

(4) Review of accident-sequence studies to determine which component failures following a design basis accident are most serious.

c. A design adequacy assessment will be performed on the above generic designs by analyzing important design features such as:

(1) Design deficiencies,

(2) Material-compatibility problems,

(3) Fabrication problems,

(4) Problems caused by ambient and accident environments,

(5) Interface problems for installed equipment,

(6) Maintainability problems,

(7) Wear and aging problems, and

(8) Quality control problems.

d. Tests will be conducted on a subset of the generic designs identified above to verify areas of deficiencies and failure modes, as well as design margins that could not be conclusively identified by analysis.

e. A list of acceptable instrumentation and electrical generic designs will be developed, including appropriate specifications and recommendations for desired safety performance.

2. Plant Electrical System Evaluation

The following specific task areas comprise the plant electrical system evaluation program:

a. Generic electrical system components used in plant electrical systems will be identified.

b. A design review of generic electrical system components will be performed to identify failure modes in vital electrical systems and the resulting conditions the failures impose on the plant availability of electrical systems. A similar review will be performed on the plant electrical systems to identify failures that could affect essential electrical supply requirements. LERs and other operating records will be reviewed to determine what offnormal load, powerline, and electrical system interruptions and disturbances might be anticipated. Using the results of these reviews, a list of possible challenges to the plant electrical system will be compiled in the form of voltage, current, frequency, and power factor disturbances.

c. Computer models will be developed for generic plant electrical systems--including auxiliary supplies and interconnections--that are capable of simulating the disturbances to the plant electrical system identified above. The plant electrical system performance under these offnormal conditions and its capability for meeting the electrical requirements of affected safety systems will be evaluated.

d. Using LERs and accident sequence studies, high probability multiple events such as failures and disturbances that could affect the plant electrical system will be identified. The plant electrical system performance with the postulated multiple events will be determined using the computer models developed.

3. High Frequency Electrical Disturbances and Protection Requirements

Lightning has been responsible for instances of voltage surges and damage at several nuclear power plants. Some of these incidents have produced conditions that would have limited the availability of redundant protective systems. Existing plant designs will be studied to determine whether acceptable industry standards for such events exist. The study proposes using a simple experimental approach utilizing a reflector to send a small (well below damage threshold) beam of high frequency electromagnetic pulse radiation at potential power plant collectors and measure the pulse level at plant locations within containment where essential electrical safety equipment is located. The results of this study will be translated into functional requirements that could be implemented in a regulatory guide. It has been proposed that this work start in FY 1981.

4.2.5.5 Results

The results of the instrumentation and electrical research program have been used to verify and augment NRC regulatory guides and industry standards. Detailed data obtained have also aided the regulatory review staff in making licensing decisions. The work planned to start in FY 1982 will have, as an additional goal, the evaluation of safety-related instrumentation and electrical systems to identify generic deficiencies.

FY 1981 Evaluate CE heated thermocouple level system at ORNL; report and evaluate test data. Evaluate Westinghouse liquid level water column differential pressure (dP) system at Semiscale; report and evaluate results after test. Refine ORNL heated thermocouple prototype. Support NRR to evaluate, through testing, licensee-submitted instruments. Test EPRI neutron flux level system at LOFT. Initiate vulnerability assessment of pilot nuclear power plants to voltage surges.

FY 1982 Complete Westinghouse differential pressure test at semiscale. Refine ORNL torsional ultrasonic level system. Test new level instruments submitted to NRR for licensing. Develop hardening (i.e., less susceptible to damage from outside forces) recommendations for lightning protection.

FY 1983 Conduct additional full-scale fire replication tests. Complete CO_2 fire suppression tests. Provide data for use by licensing and standards in developing and codifying design criteria (GDCs, regulatory guides, etc.). Continue aging methodology development for design basis accident testing. Continue LOCA and high energy line break qualification testing evaluation. Complete TMI-2 instrument evaluation. Continue design verification review for safety-related instrumentation.

FY 1984 Conduct full-scale separate effects fire tests. Complete inplace smoke detector test development.

FY 1984-1987 Continue assessment of aging methodology. Continue qualification testing evaluation. Continue design verification reviews for safety-related instrumentation. Conduct additional fire tests as requested. Characterize cable tray fires, and use in preparing and evaluating fire hazards analyses.

4.3 Plant Systems Behavior

This program provides for research on the design requirements and operational behavior of plant systems important to safety in use in currently licensed nuclear power plants and proposals for new improved systems in future plant designs. Emphasis is to be placed in the research studies on those plant systems and components essential to the safe operation, shutdown and post-accident cooling of the nuclear plants. Early emphasis will be given in the program to studying control system failure modes and effects and an assessment of the potential for failures to lead to severe plant operational degradation and possible accidents. The control sytems studies will consider both the single failure criteria, common mode failures and multiple or cascade failures. Consideration will be given to whether failures in non-safety grade sytems and equipment can cascade from one system to another causing failures in essential safety systems. This information and data will be used by NRR in assessing control system designs in currently licensed and proposed nuclear plants and provide, along with the SASA and integrated reliability program results, a sound basis for assessing the risk to the public from control systems failure. The data will also assist NRR in developing possible revisions to the General Design Criteria (GDC) where needed to improve the level of safety.

Studies will also be performed considering plant system behavior as a function of the design criteria employed. For example, many support systems are shared by multiple essential plant safety systems. The consequences of failure in the support system and possible system interaction effects will be considered. The need for more reliable, redundant or separate support systems and the need for plant systems dedicated to a single safety function will be evaluated. Other studies to evaluate the ability of safe shutdown systems and systems to function following an accident will be performed. These studies will provide criteria and requirements to be used by licensing in achieving greater assurance that nuclear plants can be safely shut down and that shutdown cooling requirements can be met after accidents.

Included also in this activity are system-related activities such as providing independent field measurements using diagnostics and monitoring to support NRR and IE in independent assessment of the cause of nuclear plant system failures and in verifying the adequacy of subsequent licensee corrective actions. Techniques, methods and equipment for diagnostic and surveillance measurements are to be evaluated to provide NRC with an independent assessment of licensee proposed diagnostic and monitoring equipment to be proposed as part of their response to the TMI Action Plan and provide sufficient data to develop regulatory guides and criteria for requirements to enhance safety through the introduction of such systems where the level of safety of nuclear plant operation can be significantly enhanced by their use.

4.3.1 Regulatory Objective

The main objective of the plant system behavior program is to assess the adequacy of nuclear plant system designs to provide for safe operation, to evaluate their vulnerability to system failure and to determine whether they include adequate design provisions for dealing with accidents or abnormal events. For example, studies to identify the causes of failure in control systems are to be performed. Criteria and guidelines for reducing the vulnerability of control systems to failure and associated accident causing potential will be proposed based on the study results. This program complements the instrumentation and electrical program described in Section 4.2, the SASA program described in Section 4.1 and the integrated reliability program in Chapter 10.

4.3.2 Technical Capabilities Required

The research provided by this program is needed to:

1. Provide assessments to be used by the NRC licensing staff to evaluate the design basis for engineered safety features and plant systems currently in use and planned for future nuclear plants, as well as to assist in the development of criteria, guides, and standards for plant safety systems requirements and safe operation;

2. Evaluate control system failure modes and effects and system interactions for both safety and interrelated nonsafety grade systems to determine plant vulnerability to severe accidents. The results from this study will contribute to the reassessment of the single failure criteria.

3. Support the NRC inspection of nuclear power plants by developing measurement techniques for evaluating the adequacy of plant safety features during construction and operation;

4. Aid the NRC licensing and inspection staffs in independent diagnostic investigations of abnormal occurrences or failures at operating nuclear power plants;

5. Assess and develop criteria for safety system requirements to increase the level of safety in plant systems design; and

6. Evaluate techniques, systems, and instruments for improved safety in operation through surveillance and monitoring for early detection and diagnostic of plant assessment abnormal behavior.

The three current active programs in this section were initiated by research requests from NRR: the noise diagnostics program at ORNL began in FY 1976 in response to user request NRR-76-19; the continuous online surveillance system program was started at NRR's request (NRR-79-16) in January 1980; the program to assess the feasibility of LWR subcritical reactivity monitoring by the californium-252 source-driven neutron noise analysis method was requested by NRR-79-23 in FY 1980.

4.3.3 Status of Capabilities

Most of the research to be performed on plant system behavior will consist of new programs to be initiated in FY 1981-1983 as part of the increased attention being given to plant operational safety.

The noise diagnostics program at ORNL has provided technical support to NRC by:

1. Independent measurements of the Palisades nuclear plant core barrel vibration problem and diagnostic analysis to identify the cause, using noise analysis followed by additional measurements to verify the correction;

2. Independent measurements at the Calvert Cliffs plant of core barrel vibration and diagnostic assessment of the flange-clamping force required to correct the problem;

3. Independent measurements of the BWR-4 instrument tube vibration problem and verification of the cause and correction at the Browns Ferry and Hatch nuclear power stations;

4. The development of criteria and guides for core vibration measurements in CE PWRs and instrument tube vibrations in BWR-4s. These criteria were used by the Office of Standards Development in preparing branch review plans and proposed regulatory guides;

5. Field measurements and diagnostics during and after the TMI accident; and

6. Independent assessment of core movement and temperature fluctuations in the Ft. St. Vrain gas-cooled reactor project.

The current ORNL effort is concerned with completing the assessment of criteria for loose parts monitoring (LPM) systems (Regulatory Guide 1.133). In particular, sensor mounting methods, calibration procedures, part size identification, and part location techniques have been studied in large-scale vessel experiments and are to be reported in FY 1981. This information will support the NRC assessment of LPM systems installed at nuclear plants and future revisions to Regulatory Guide 1.133.

The collection of baseline primary system signal signatures characterizing both core and system behavior of PWRs and BWRs was initiated in FY 1979. This work is being carried on at a modest level to provide essential data which is needed to evaluate plant status such as that at TMI-2 following the accident and is expected to continue at that level. Other tasks in the noise diagnostic program include (1) the development of basic analytical methodology and measurement techniques to support the use of noise analysis in diagnosing reactor vessel vibrations or fuel blockage that might result from inadequate design or system failure and (2) an assessment of the use of noise analysis measurements for determining BWR reactor stability without perturbing plant operations. Analytical calculations using the transient reactor kinetics LAPUR code have been made to study the feasibility of measuring stability by noise analysis. They showed close agreement between stability parameters experimentally measured at the Peach Bottom BWR nuclear station using noise techniques and the results of detailed transfer function analytical calculations using LAPUR. The conclusions from this study need to be confirmed by experimental measurements at operating BWRs. The stability of a BWR core decreases toward the end of a fuel cycle.

A self-learning, automated, continuous online surveillance system was installed at the Sequoyah PWR during the initial startup of the plant in September 1980. This system will be used to determine whether a computer-based system using plant signal spectrum analysis diagnostics and pattern recognition techniques can continuously monitor nuclear power plant operation to anticipate and identify the occurrence of any unanticipated abnormality in the plant systems behavior which might represent a significant accident precursor. Early identification of abnormalities offers the opportunity of preventing a later accident. This program was begun in FY 1980 and is expected to be completed in FY 1982.

4.3.4 Research Program Objectives

The research provided by this program is directly related to NRC regulatory program needs through:

1. The assessment of plant vulnerability to control systems failures and systems interactions;

2. The development of criteria for assessing plant systems designs being reviewed by NRR such as the operability of and access to essential shutdown cooling systems following an accident involving radioactive releases and the possible need for additional alternate or dedicated shutdown cooling systems;

3. The independent verification by field diagnostic measurements of licensed nuclear plant system malfunctions and confirmation that the problem has been adequately corrected by the licensee's plant revisions;

4. The assessment of techniques and systems proposed for performing diagnostics or surveillance of system behavior at operating plants;

5. The assessment of system design criteria and evaluation of plant operational behavior to provide information needed to prepare regulatory guides and standards to be used in licensing plants; and

6. The provision of technical support and development of equipment for use by the Office of Inspection and Enforcement in making independent inspection measurements.

4.3.5. Research Program Plan

A research plan for specific activities under this section follows:

4.3.5.1 Subcritical Reactivity Monitoring

A program to evaluate the feasibility of LWR subcritical reactivity monitoring using the californium-252 source-driven neutron noise analysis method was started in FY 1980. The initial phase will determine the feasibility of using this method with large LWR cores in which spatial harmonics and core coupling must be evaluated. The initial analytical phase of the study will employ the JPR reactor kinetics code to determine coherence factors between detectors in a large LWR core. Calculations are to be completed in FY 1981. Further work on this task pertaining to fabricating a suitable source chamber would occur in FY 1982, and final experimental verification is planned for in FY 1983 if the concept is found to be feasible in the analytical studies now in progress and it is determined that an experimental test is needed. This program fills an outstanding NRR-identified need for monitoring LWR subcriticality during refueling, core maintenance and following core accidents. Potential application to TMI core recovery will be considered.

4.3.5.2 Plant Systems Evaluations

Analytical and design review studies of plant safety and safety related system behavior will be started in FY 1981. The level of this effort will be increased in FY 1983, FY 1984, and FY 1985. These studies will support licensing actions and the development of criteria and guides for assessing the safety of nuclear plant operations through the identification of system interactions which could lead to unsafe operation, design deficiencies, and failure modes. These studies include:

1. The assessment of criteria required in the design of systems and facilities to adequately consider the need for operational use and access during and following accidents;

2. The impact on plant safety associated with system interactions and failures when components and portions of systems are connected to the same or shared among several engineered safety systems;

3. The safety consequences of and criteria for nuclear plant automation as a means of decreasing system vulnerability;

4. The consequences of individual component failures on plant safety system behavior and availability; and

5. The influence of transients on plant systems behavior.

The exact scope and program tasks to be initiated in FY 1982-FY 1984 will depend on the needs of NRR. Decisions on priority and level of effort will be guided by the results of LERs, abnormal occurrence reports, and plant accident studies now being performed by NRR and AEOD and the accident sequence studies being conducted in the SASA and reliability research programs.

4.3.5.3 Independent Field Measurements

Activities that support NRR and IE licensee reviews will continue to be provided by field measurements to verify malfunction and failures. Also provided will be support in analyzing data from operating reactors to identify the cause of malfunctions.

4.3.5.4 Control System Studies

A control system evaluation study is to be started in FY 1981. The study will include a failure modes and effects analysis (FMEA) for generic plant systems, and the effects of electrical, control, and mechanical interactions on control system behavior and plant systems behavior. The studies will cover the tasks below which will be initiated in FY 1981-1982. All of the tasks are to be completed by FY 1986:

1. Failure modes and effects analysis of generic plant instrument and control systems for single failure, and common-mode or common-cause failures. This study will ultimately cover a generic plant typical of each of the major vendor's current designs;

2. Systems interaction studies, including mechanical and electrical effects;

3. Development of general criteria for evaluating control systems;

4. The effect of cascading failure and multiple (noncommon mode) failures on the safety of plant systems; and

5. Development of criteria and plans for classification of safety-related systems based on results from the above, primarily the FMEA.

An assessment of the need for a dedicated hybrid analog system for performing specific control system evaluations will be made in FY 1982. The hybrid analog would be assembled in FY 1982, and generic control systems response studies would be made from FY 1983-FY 1987 for plants identified by NRR for which an independent safety assessment is desired. This task will be closely coordinated with the Severe Accident Sequence Analysis Task.

4.3.5.5 Results

The schedule for current and future research accomplishments on plant systems behavior is given in the sections below.

1. Noise Diagnostics

FY 1981 Loose parts monitoring research is to be completed.

FY 1982 Collection, categorization, analysi , and data bank file for PWR and BWR neutron signatures are to be completed.

FY 1983 BWR stability measurements using noise diagnostics are to be verified experimentally.

FY 1984 Collection, categorization, analysis, and data bank file for PWR and BWR system process signal signatures are to be completed. Noise analysis methodology for detecting core internals vibration, blocked fuel assembly flow disturbance, and other malfunctions for both PWRs and BWRs are to be completed.

FY 1985 Studies of systems such as computer verification of plant status and liquid inventory tracking will be completed.

During the entire 5-year period of the long-range plan, NRR and IE will be given support in assessing malfunctions and failures in nuclear plants through field diagnostic measurements and analyses.

2. Subcriticality Monitoring

FY 1981 Analytical determination of the feasibility of applying this technique to large LWR cores is to be established.

FY 1982 If feasibility is established and a experimental test is to be made, a source chamber fabrication would be complete.

FY 1983 If experimental test measurements with a large LWR core are required, they would be started.

3. Continuous Online Surveillance

FY 1981 Development and fabrication of an improved NRC-dedicated surveillance system with expanded power spectral density (PSD) and signal recording capability will be completed.

FY 1982 Pattern recognition, signature identification, and surveillance system operation will be investigated by measurements at the Sequoyah PWR plant. Continuous online surveillance system development and demonstration will be completed at Sequoyah.

FY 1983-1984 Demonstration of the continuous online surveillance system at a BWR plant is planned and a study of BWR signature characteristics as accident precursors will be conducted.

4. Control System Evaluation

FY 1982 The development of general criteria for evaluating control systems will be completed.

FY 1983 The initial assessment of the failure modes and effects analyses of generic plants will be completed.

FY 1984 Systems-interaction studies that will begin in FY 1982 will be completed. Cascading control system failures and multiple failures will be completed.

FY 1985 The need for a hybrid analog will be evaluated in FY 1982 and if determined to be needed will be assembled in FY 1983. Control system analysis using the hybrid analog will be performed in support of NRR licensing assessment and identified safety concerns and will be completed by this time.

5. Criteria for System Design for Accidents

FY 1984 PWR and BWR safety systems that are required to function during accidents will be identified, operational needs and access requirements will be studied, and design criteria will be completed. This study will consider both currently operating and proposed nuclear power stations on a generic basis to identify design weaknesses. Suggested revisions to existing plants and the optimization of plant systems for increased safety during accidents will be considered and information for criteria and design standards will be provided.

6. Safety of Shared Systems

FY 1983 Criteria and guides for improving the safety of those plants where systems or components are shared would be prepared.

FY 1984 Existing PWR and BWR plant designs will be studied to identify the components and support systems shared among various engineered safety features and interconnections among different safety systems. An assessment of the resulting vulnerablility and reliability will be made. This study to identify serious weaknesses as a result of safety system interactions in current generic plant designs, started in FY 1982, will be completed.

7. Nuclear Plant Automation

FY 1982 An investigation of the use of computer control and micro-processors and the automation of existing plant protection and safety systems (including both engineered safeguards and reactor protection systems) will be started.

FY 1984 The question of using digital control systems to automate all plant systems and operations in nuclear plants will be considered in a study to be started.

4.4 Mechanical Components

Research on mechanical components and systems is conducted to assist the Commission and licensing staff in making their regulatory and policy decisions to resolve problems concerning the behavior and reliability of mechanical systems and components related to safety.

Licensing problems and safety issues impacted by this program relate to the following areas:

1. performance of pressure vessels, piping, pumps, valves, snubbers, and similar components and equipment under postulated accident conditions including earthquakes;

2. response of mechanical systems to operating transients such as safety relief valve discharge;

3. assessment of how loads are combined;

4. evaluation of industry-wide computer codes used to predict the behavior of mechanical components and systems; and

5. assessment of the validity and applicability of the ASME Boiler and Pressure Vessel Code, Sections III and XI.

6. Information to assist the licensing staff in the Systematic Evaluation Program and response to Public Law 96-295 Section 110 (Bingham Amendment).

The emphasis of this program is on strengths and weaknesses in present licensing practice and qualification criteria. Where possible, these strengths and weaknesses are expressed in a probabilistic, risk-oriented format that aids in the licensing decisionmaking by more accurately describing the behavior of mechanical components and systems. The focus is on both operating reactor (including the Systematic Evaluation Program) and future reactor designs (including those under construction). Some research has generic licensing applicability such as the work on combinations of loads and margins of safety of components subjected to earthquakes; other investigations are more narrow in scope such as the testing of BWR-6 Mark III equipment to map out the response characteristics for later use in determining behavior of similar equipment under normal and accident conditions. This program comprises experimental and theoretical aspects and is becoming increasingly involved in international jointly funded research.

4.4.1 Regulatory Objective

The mechanical engineering research program is an analytical and experimental program designed to provide information on the engineering and structural integrity of mechanical systems, components, and equipment of light-water reactors under anticipated operational, environmental, and postulated accident conditions. This information must include system and component response characteristics, definition of failure modes, failure probabilities, and the functional limits of active and passive safety-related components. The primary regulatory objective of the program is to provide the NRC licensing staff with methods and techniques for evaluating operability and structural integrity in terms of margins of safety and failure probabilities.

The mechanical engineering research program provides research support for the development and application of criteria for equipment qualification and testing methodology. Supporting the data base for standards development related to the functionability of mechanical systems and components for improved plant safety is a continuing responsibility.

There is also a need to determine if safety requirements will be maintained as systems that have been designed to the ASME code, Sections III and XI (required by NRC regulations) reach the end of their design life.

The NRC licensing staff needs the capability to accurately predict the mechanical behavior of systems and components under operating and accident conditions. Decisions must be made regarding such things as the placement and reliability of piping supports and snubbers, the dynamic response of pipes and vessels, and the reliability of pumps and valves that are required to function under severe accident conditions such as large earthquakes Thus, to provide this capability, it is necessary to develop a methodology that will predict not only individual component behavior, but system behavior as well, and integrate the uncertainties associated with the various parameters of the methodology chain. If the real behavior of systems and components can be predicted within the limits of calculated uncertainties for operating and accident condition, then an evaluation can be made of the design conditions resulting in a better understanding and estimate of the margins of safety.

An essential part of mechanical engineering research planning and management is maintaining an awareness of related research being conducted elsewhere. Consequently, liaison with foreign and domestic organizations, including government agencies, must continue. Interaction with other NRC offices in order to have a firm understanding of their needs will also continue.

The mechanical engineering research program is divided into four major areas: the seismic safety margins research program (SSMRP), mechanical reliability of systems and components, structural reliability of systems and components, and general reliability of systems and components.

4.4.2 Technical Capabilities Required

Technical capabilities required are classified in the four major subdivisions as described below:

4.4.2.1 Seismic Safety Margins Research Program

Needs exist for improved seismic risk evaluations, better assessments of the dominant contributors to seismic risk, and improved standards for the seismic design of nuclear power plants.

4.4.2.2 Mechanical Reliability of Systems and Components

A better understanding of the performance of pumps, valves, snubbers, and similar equipment and the functional limits and failure modes of these same items is needed. Satisfaction of this need will lead to more reliable mechanical operation through changes in design, testing and operating specifications.

4.4.2.3 Structural Reliability of Systems and Components

An assessment of aspects of the ASME code relating to buckling and fatigue is needed. A better understanding of load combination probabilities and methods for combining loads is also needed. Structural computer codes to calculate the effects of dynamic environments are unvalidated and need verification.

4.4.2.4 General Reliability of Systems and Components

Improved dynamic qualification-testing standards for mechanical and electrical components are needed. Procedures for assessing damage subsequent to severe environmental and accidental events must be developed.

4.4.3 Status of Capabilities

4.4.3.1 Seismic Safety Margins Research Program

The occurrence of large earthquakes affecting nuclear power plants is a major concern in ensuring the health and safety of the public. These concerns include:

1. insufficient data to predict the maximum earthquake likely to occur at a plant site;
2. lack of experimental data that demonstrates the integrity and functionability of components and systems under earthquake loads;
3. the need for critical safety related components to operate during or shortly after a large earthquake;
4. the potential for large earthquakes to cause simultaneous failures in diverse safety systems. Thus, the concept of safety by redundancy may be jeopardized.

Seismic design of nuclear power plants includes the definition of the earthquake, or seismic hazard, in terms of the size of the earthquake and characteristics of vibratory motion. The definition of the seismic input involves consideration of the geology and seismology of the region, and the local history of earthquakes and their influence on structures. Mathematical models are then constructed to determine the response of structures, components, and equipment to the defined seismic input.

Over the past decade, seismic design criteria for nuclear power plants have undergone a very rapid change as a result of advancement in the state of the art.

In order to adequately assess these new design criteria a probabilistic approach needs to be developed for the following reasons:

1. The seismic contribution to the overal risk is not known. The Reactor Safety Study (WASH-1400) did not adequately consider seismic risk.
2. A probabilistic approach will provide the means to assess the adequacy of factors of safety and other conservatisms used to assure the safety of nuclear plants. Thus higher or lower factors may be applied to the individual components or systems.
3. Designs based only on seismic considerations may be evaluated on a systems basis to see if the overall plant safety has been increased or decreased. Placement of seismic restraints and snubbers are a good example of areas to be investigated.
4. Uncertainties may be identified and evaluated on the basis of their contribution to the overall risk.
5. Older plants designed to less stringent seismic design criteria can be evaluated using a probabilistic systems approach to determine if there is sufficient inherent margins present to allow the plants to remain in operation.

This evolution in design criteria, in combination with the uncertainty in the seismic risk and the lower-allowable damage levels at nuclear power plants, indicates a need to develop a methodology that will allow a quantitative determination of the safety significance of one set of criteria against another.

This probabilistic approach can ultimately provide practical tools for the assessment of nuclear power plant safety. It can lend stability to the seismic safety licensing process by serving as a means for producing probabilistic estimates of risk and by pointing to the most important areas of possible improvement. To function in either of these roles, the probabilistic analysis must explicitly acknowledge and evaluate sources of uncertainty. The SSMRP methodological chain thus considers uncertainties in each of its links, couples them, and propagates those uncertainties throughout the calculation.

The probabilistic seismic analysis methodology has been developed and the calculational capability demonstrated. Models for seismic input, soil-structure interaction, dynamic response of structures and subsystems, and fragility have been developed and combined using a probabilistic computational procedure. Sensitivity studies to gain engineering insight on seismic safety requirements have been started. Numerous side studies were undertaken to examine modeling alternatives, sources of error, and available analysis techniques. The SSMRP methodology is being used to assist the licensing staff in their evaluation of the seismic design of PWR auxiliary feedwater systems.

4.4.3.2 Mechanical Reliability of Systems and Components

The licensing process is tailored to ensure that components function satisfactorily when they are relied on for plant shutdown and are needed to mitigate the consequences of a postulated accident. Current codes and standards generally do not address component operability and component qualification under operating and accident conditions in sufficient depth for NRC needs.

In some cases, the equipment is not designed or fabricated to standards commensurate with the expected accident service conditions. Consequently, increased attention has been directed to the reliability estimates and the operability requirements and performance of mechanical components and systems. In addition, as new equipment and applications are introduced, it is important to ensure that safety is not compromised; therefore, effective design criteria of a generic nature must be established.

4.4.3.3 Structural Reliability of Systems and Components

Mechanical components and systems are often load-carrying members whose structural integrity must be maintained under normal operating and accident conditions. An important aspect of calculating the structural integrity of a component is determiring the proper combination of applied loads and the appropriate methodology of combining the dynamic responses. Current load-combination criteria may be overly conservative. The ASME Boiler and Pressure Vessel Code is used to ensure the integrity of components. The Code, in many cases, does not adequately address areas of critical importance and must be thoroughly evaluated.

In addition, design methodologies and techniques such as computer programs have never been satisfactorily validated. Finally, it is important that component test criteria are sufficient to ensure the struct l integrity of components under normal and accident conditions.

4.4.3.4 General Reliability of Systems and Components

Dynamic qualification testing of mechanical and electrical components used in nuclear power plants in a dynamic environment is desirable to ensure their ability to function during and after seismic occurrences. The current methods of seismic qualifying of mechanical and electrical components have evolved over a period of years. The methods used in the past are significantly different from those in use today. C ents presently operating in nuclear power plants were tested by past m and, in light of present knowledge, there is concern about the adequacy o ir qualification as well as anomalies in existing methods. Studies in this area must include definition of the vibratory motion experienced by the component to determine if the test conditions adequately simulate operating and accident conditions.

One of the principal shortcomings in system risk assessments is the lack of fragility data on system components. One possible source of fragility data is the results of component-qualification tests.

Essential mechanical safety systems in nuclear power plants are designed and evaluated for a variety of severe accidental and environmental events such as earthquakes and loss-of-coolant accidents. After such an event does occur, the mechanical system must be evaluated to determine if the system has degraded to the point where continued operation of the system will pose unacceptable risks to the public.

4.4.4 Research Program Objectives

4.4.4.1 Seismic Safety Margins Research Program

The objectives of the SSMRP are to estimate the conservatisms (or lack of conservatisms) in the NRC standard Review Plan seismic safety requirements and to develop improved requirements.

The SSMRP is a comprehensive, multidiscipline research program, the purpose of which is to provide a methodology for determining the inherent margins of safety in a nuclear power plant that is subjected to a large earthquake. The program will also estimate the seismic risk, in terms of probabilities of failure and radioactive release, and identify which areas of the seismic design and their uncertainties are the most important in terms of safety and risk.

4.4.4.2 Mechanical Reliability of Systems and Components

This program defines the tests and analytical methods, or their combinations, to ensure that components and systems will function under postulated plant conditions to safely shut down the plant, maintain the integrity of the reactor coolant boundary, or mitigate the consequences of an accident. In addition, the program addresses selected mechanical engineering needs for the inspection and enforcement programs, such as quality assurance and the development of qualification test methodologies. A definition of equipment safety margins and assessments of risk associated with equipment and system performance is included as an integral part of those studies that relate system and component interactions.

4.4.4.3 Structural Reliability of Systems and Components

The technical basis to ensure that safety-related systems, components, and equipment satisfy the applicable portions of the General Design Criteria (specifically those dealing with strength, stability, and deformation requirements) is to be provided. These systems and components are primarily ASME Class 1, 2, and 3 components, component supports, and core-support structures. In addition, bases should be provided to determine that acceptability of the stresses and deformation of active components and systems are compatible with their operability requirements under all postulated conditions.

4.4.4.4 General Reliability of Systems and Components

The objective of the general reliability program is to provide technical support to both the structural and mechanical safety programs and the performance of systems and components program.

4.4.5 Research Program Plan

4.4.5.1 Seismic Safety Margins Research Program

The approach for achieving the objectives stated in Section 4.4.4.1 is to develop a probabilistic approach that more realistically estimates the behavior of nuclear power plants during an earthquake. Techniques developed will be tested against experimental data wherever possible. There will be three phases in the program:

1. In Phase I, these techniques will be used to perform sensitivity studies to gain engineering insights into seismic safety requirements. The probability of failure of structures, systems, and components and the probability of radioactive releases over a range of earthquake levels will be used to help determine priorities for Phase II research. Studies of the individual steps in the seismic methodology will be documented and used to provide preliminary results for issuing licenses. Phase I will be completed in FY 1982.

2. In Phase II, the methodology from Phase I will be used to estimate the conservatisms (or lack of conservatisms) in the SRP seismic safety requirements. Research will be performed in directions defined in Phase I to develop an improved methodology. Finally, this improved methodology will be used to refine estimates of the conservatisms and to define the seismic contribution to reactor risk, including impact on other safety considerations. Phase II will be completed in FY 1984.

3. In Phase III, the improved methodology from Phase II will be used to recommend changes in the SRP deterministic seismic safety requirements. Phase III will be completed in FY 1985.

The probabilistic seismic systems methodology developed by the SSMRP will provide a means to describe the behavior and assess current and future seismic designs and the impact of those designs on the overall safety of the plant. An example would be the evaluation of the need or desirability of increasing or decreasing the number of seismic supports, such as snubbers and anchors, or the desirability of stiffer systems that may induce larger thermal loads.

The SSMRP is composed of eight separate projects carefully integrated to produce a useable tool for licensing. They are:

1. Plant/site selection,
2. Seismic input,
3. Soil-structure interaction,
4. Structural building response,
5. Structural/mechanical subsystem response,
6. Fragility,
7. System analysis, and
8. Development of the SMACS seismic code and BE-EM comparison.

4.4.5.2 Mechanical Reliability of Systems and Components

The following paragraphs provide a brief description of the projects supporting the mechanical reliability program:

1. Piping System Restraints and Snubbers: An analytical and experimental characterization of snubber performance will be established to help ensure plant piping system reliability. In addition, this project will assess the impact of pipe restraint devices on the performance of a piping system during normal operation and will provide a measure of risk to overall plant safety that the use of these devices introduces. This project was started in FY 1980 and will be completed in FY 1984.

2. Pump and Valve Qualification: Acceptance criteria and methods for qualification (supported by a technical parametric data base) of safety-related pump and valve operability will be developed. Work will be started in FY 1982 and completed in FY 1985.

3. Non-Safety-Related Systems: The methodology will be developed to identify and examine those plant systems that are currently classified as non-safety-related but that contribute directly to plant safety. This methodology will involve establishing the kinds of interactions that will occur between the non-safety-related and safety-related systems, the resultant system failure modes, and the resultant risk to plant safety. This project will begin in FY 1982 and will be completed in FY 1984.

4. Containment Leak Testing and Vent Systems: Reduced-pressure leak-rate testing will be evaluated, and time duration for testing and the value of supplemental leak-rate testing will be assessed. Methods for ensuring containment-liner-weld integrity will also be developed. An additional objective is to provide a safety assessment on the use and performance of purge valves and venting systems. Work will be started in FY 1982 and completed in FY 1985.

5. Hydrogen Management: The equipment and mechanical system designs required for the management of hydrogen released inside LWR containment will be determined. This project will begin in FY 1983 and will be completed in FY 1986.

6. Safety and Relief Valve Qualification: The industry safety and relief valve testing program required by the TMI Lessons Learned Task Force will

be monitored to verify that the PWR pressurizer safety and relief valves and the safety-relief valves on the BWRs will perform adequately with two phase and liquid flow. It also involves review of data developed from these tests to provide methods for analysis of these valves and associated piping to confirm the adequacy of existing and future installations. The project was started in FY 1980 and will be completed in FY 1984.

4.4.5.3 Structural Reliability of Systems and Components

The following projects will be conducted as part of the structural reliability program:

1. Load Combinations: Probabilities will be developed for the simultaneous occurrence of an earthquake and a double ended guillotine break of the primary system piping. A reliability-based procedure for combining responses to dynamic loads applicable to mechanical systems and components using load factors will also be developed. This project was started in FY 1980 and will be completed in FY 1985.

2. Participation in the HDR Program: There will be collaboration with the West Germans in research at the HDR facility related to LOCA events as they affect reactor vessels and their internals, the reactor cavity, the containment, and safety valves. Present methods of estimating soil-structure interaction, building response, and piping system response under dynamic environments similar to earthquakes will be evaluated by comparing blind posttest and pretest calculations with actual measurements. Work was started in FY 1980 and will be completed in FY 1985.

3. Assessment of the ASME Boiler and Pressure Vessel Code: Those portions of the ASME Boiler and Pressure Vessel Code used in the NRC licensing process for mechanical components will be reviewed and assessed to determine whether they require modifications. Examples include dynamic versus static allowables, class 2 fatigue analyses and stress corrosion. The project will start in FY 1983 and end in FY 1986.

4. Verification of Applied Mechanics Computer Codes: Existing computer programs that predict the response of mechanical systems and components in nuclear power plants will be verified, and benchmarks for demonstrating the acceptability of all applied mechanics computer codes used in the design and analysis of safety-related systems in nuclear power plants will be developed. Work was started in FY 1979 and will be completed in FY 1986.

5. Functional Capacity of Piping Systems: Realistic functional criteria for the design of Class 1, 2, and 3 piping system components subjected to dynamic environments (i.e., seismic, impact, and flowshock loads) will be developed. The project will start in FY 1983 and end in FY 1986.

6. Anchoring of Component Supports: Tests will be conducted to investigate the adequacy of design rules and the potential degradation and sensitivity to construction and installation errors of expansion anchor bolts. This project will start in FY 1983 and end in FY 1985.

7. Large-Scale Testing Facility: The need for a facility geared towards structural and mechanical testing of nuclear power plant systems, components, and structures will be assessed. This work will be started and completed in FY 1983.

4.4.5.4 General Reliability of Systems and Components

The general reliability program consists of:

1. Foreign Research Coordination: Ways will be developed of obtaining in a timely manner foreign results, of exploring areas in which the NRC and foreign governments can conduct joint venture research, and of evaluating the bases for differences in national licensing practices between the U.S. and other countries. This project was started in FY 1981 and will be completed in FY 1986.

2. Design Qualification Testing and Assessment: Many methods of qualifying the design function of components subjected to operating environments

have been developed. The adequacy of these methods must be assessed. This two-phase program will develop the technologies necessary for this assessment. The first and more immediate phase is concerned with the assessments under dynamic environments. The second and long-term phase is the assessment under environmental conditions; i.e., temperature, pressure, radiation, and chemical environment.

The need and methods appropriate for testing safety-related piping systems at nuclear power plants will be evaluated, focusing on verification of dynamic parameters and mathematical models and on identification of installation and construction errors.

Past and present methods of qualifying the operability of mechanical and electrical nuclear plant components will be evaluated and correlated. A program and procedures for assessing the degree of damage incurred by mechanical systems and components following severe accidental and environmental events will be developed. This program will include:

a. A testing program to study the effects of various inputs to determine which wave forms are the best simulation of earthquake excitation,

b. An evaluation of the influence of component aging and environmental degradation effects on the dynamic qualification of equipment,

c. The performance of tests to determine the probability of failure (fragility) of selected critical equipment identified by the SSMRP,

d. The development of scale-modeling guidelines for the dynamic testing of equipment, and

e. An assessment of the reliability and uncertainty of dynamic qualification methods.

A program will be developed to identify and quantify all the vibrations and accident-inducted dynamic loads that may have an effect on the functional operability of safety-related mechanical and electrical equipment. The program will be started in FY 1982 and completed in FY 1986.

By testing, damping values and fragility levels will be determined for structures and equipment in older operating reactors as an aid to the seismic portion of the Systematic Evaluation Program. This project will be started in FY 1982 and be completed in FY 1986.

3. Generic Study of Fasteners Used in Mechanical Components: The characteristic failure and loosening modes of bolt and screw fasteners in a static and vibrating environment will be examined. A second objective is to establish thread-gauging and vibration-test requirements related to fastener loosening and strength. Work will begin in FY 1986 and end in FY 1987.

4.4.5.5 Results

1. Seismic Safety Margins Research Program

The licensing staff will be able to use the fragility (probability of failure) data to evaluate the adequacy of components for various dynamic input levels. The probabilistic estimates can be used to assess various proposed design changes submitted by the applicants. The analysis techniques and computer software can be used by NRC staff to conduct independent assessments and audits.

FY 1983 Document results of fragility test conducted in the U.S. and in foreign countries; provide initial estimates of the contribution to risk from a seismic event; complete assessment of impact of design and construction errors on the probability of a failure in the primary system; complete sensitivity studies to identify major contributors to seismic risk; complete Phase II of the SSMRP.

FY 1984 Complete Phase III of the SSMRP; provide an estimate of the contribution to risk from large earthquakes; provide probability-of-failure data for mechanical components; provide software for probabilistic response and systems analysis for seismic design; provide a reliability-based procedure for design of mechanical and electrical components under simultaneous dynamic environments using load factors.

2. Mechanical Reliability of Systems and Components

FY 1983 Complete evaluation of ATWS testing of pressurizer safety and relief valves; complete analysis and design of a comprehensive pump and valve qualification program; provide recommendations for regulatory guidelines on the design, application, and testing of snubbers; identify acceptable duration and frequency for conducting containment leak tests; provide qualification test methodology and criteria for containment purge valves; establish a data base and engineering requirements for systems and components for the management of hydrogen.

FY 1984 Develop a program to verify and qualify passive-restraint devices; complete studies evaluating the performance of leak-protection and containment-isolation systems; provide criteria and evaluation guidelines for containment-venting system performance.

FY 1985 Complete methodology development for correlation of test results with respect to performance of pumps and valves; assess seismic performance of control rod drives evaluated; provide recommendations for qualification and inservice inspection of new generation of hydraulic and mechanical snubbers.

FY 1986 Provide improved standards for all types of piping system restraints and snubbers.

3. Structural Reliability of Systems and Components

FY 1983 Complete development of analytical and physical benchmarks for inelastic analysis of piping systems; complete qualification of computer models to estimate structural response as a result of blowdown and check valve transients at HDR; document damping of piping systems as a function of amplitude using HDR snapback tests; provide recommended modifications to the ASME Code, Section III.

FY 1984 Evaluate small-break LOCA probability; recommend procedures for assessing structural integrity after severe environmental and accident conditions; provide benchmarks for three-dimensional models of piping configurations.

FY 1985 Complete elastic benchmark tests of mechanical components using ongoing test programs.

FY 1986 Complete benchmark tests of selected high-priority topics from SRP for verification of applied-mechanics computer codes; complete development of improved standards for the design of expansion anchor bolts and other component supports.

FY 1987 Document verified computer codes in applied-mechanics analysis; complete fluid-structure interaction studies of core-barrel response to blowdown loads.

4. General Reliability of Systems and Components

FY 1983 Provide guidance for optimum schedules for inservice inspection; document a procedure for preoperational testing of safety-related piping systems; recommend new facility conceptual design for seismic and environmental testing of components.

FY 1984 Provide a procedure for assessing accident damage to mechanical components; provide assessment, correlation, and recommendations for dynamic qualification of mechanical and electrical components; provide evaluation of aging effects on dynamic qualification testing; provide evaluation of modeling techniques for qualification methods; recommend vibration test requirements for fastener loosening and strength; provide methods for preoperational testing of piping systems; develop priority for selected fragility tests; identify uncertainties associated with dynamic qualification testing.

FY 1985 Integrate the results of Japanese seismic tests intc U.S. licensing practice; provide test procedures to identify pipe system installation and construction errors; provide assessment, correlation, and recommendations for environmental qualification of mechanical components.

FY 1987 Formulate preoperational dynamic test plans for nuclear reactors.

4.5 Structural Safety

The capacity of nuclear plant structures, which include concrete and steel containment shells, fuel buildings, and service buildings to withstand accident and operational loading must be known to ensure that such events will not cause failure or loss of function during accident. Research on structural safety is conducted to assist the NRC in making regulatory and policy decisions to resolve problems concerning the behavior and reliability of structures related to safety. The emphasis of the structural safety program is on the strengths and weaknesses in present licensing practice and evaluation criteria.

Design requirements for nuclear plant structures have undergone rapid changes over the past decade. Some of the major changes are:

1. increase in seismic acceleration values, three components of seismic input, and increase in the magnitude of seismic loading due to changes in the seismic response spectra.

2. new hydrodynamic loads in pressure suppression containments.

3. asymmetric dynamic loads and quasi-static overpressure loading from hydrogen burning in low volume containments.

These changed requirements and accident scenarios, now considered in new plants, were not considered in the original design of certain operating plants.

The Systematic Evaluation Program (evaluation of the 11 oldest plants) and P.L. 96-295, Section 110 (mandated review of the adequacy of all operating facilities to meet current requirements) are among the significant licensing activities to address these concerns.

The structural safety program was recently initiated to provide research support (failure modes and safety margins of structures) for the resolution of these issues. Considerable care and resources are necessary to develop this program from its current formulation stage to produce much needed answers.

The objective of the research activity in this area is to develop analytical methods verified by well documented test results to determine failure modes and safety margins of structures available to resist combinations of static and dynamic loads, to estimate probabilities of failure of structures and to establish simplified techniques and procedures that will enhance the capacity of NRC staff to independently verify structural calculations.

The long-term goal for this program area is to better understand how safety-related structures behave under various accident scenarios so that improved structural design requirements can be developed to reduce risk and enhance safety.

4.5.1 Regulatory Objective

The overall objective of this program is to provide information necessary for the safety analyses of structures at nuclear power plants under anticipated operational, upset, and postulated accident conditions, including the consideration of such environmental phenomena as earthquakes, floods, and tornadoes. This information must include response characteristics, definition of failure modes, and failure probabilities, as well as limits of deformation necessary to ensure the functioning of active and passive components related to safety.

4.5.2 Technical Capabilities Required

Structural safety research is organized into three major activities that are concerned with the choice of design loads, prediction of structural response, and structural performance.

In regard to design loads, the requirement that nuclear plant facilities be able to withstand extreme environmental and accidental loadings placed new demands on the structural engineering profession. Some of the loadings postulated for nuclear plants had not previously been considered quantitatively in structural engineering practice. Either the occurrence of the event causing the load was considered too rare to be included in the design basis or the consequences of the rare failures to be expected were considered tolerable. Today areas of particular interest include loadings as a result of earthquakes, floods, and hydrogen explosions. Another area of interest is the way in which the effects of loads should be combined for design purposes.

In the structural response prediction activity, interest centers on the analytical methods used to predict the behavior of structures under design loadings. For conventional buildings, the analytical procedures in common use have been developed through years of experience. In general, the procedures have been calibrated against laboratory measurements on scaled models and, in some cases, on full-scale buildings. There is a general confidence about the ability of those procedures to predict the performance of buildings under design loadings. The degree of confidence is related to the amount of relevant experimental data and field experience. In regard to common building types for which many tests have been run and for loadings that can be easily duplicated in the laboratory, there is a great deal of confidence. There is less confidence where data are sparse, as in nuclear power plant structures, or where loadings are difficult to simulate, as in earthquakes. Research is needed to assess the accuracy of methods and procedures used to predict structural response and to ensure that the analytical models represent the as-built structures.

In the structural performance activity, interest centers on how well the codes and standards used for design reflect the actual performance of nuclear plant structures. These codes and standards evolved from historical practice in the design and engineering of pressure vessels, concrete buildings, and steel structures. The practices had to be modified to reflect not only the more stringent loads in nuclear plant structures, but also the differences between conventional structures and nuclear plant configurations. In many instances, the bases for the requirements of these codes and standards have been more engineering judgment than definitive test data. The extrapolation to cover nuclear structures has led to concerns about the appropriateness of the rules. Questions concerning the degree of adequacy and/or overconservatism of certain requirements have been difficult to resolve in the absence of the relevant test data.

Specific user requests are:

1. Quantification of inherent safety margins in seismic design (expanded in June 1977), NRR-76-8 Rusche to Kouts (6/7/76);

2. Evaluation of margins available in flood protection of nuclear power plants, NRR-77-16 Denton to Levine (10/26/77);

3. Computer codes for soil-structure interaction (SSI), NRR-79-11 Denton to Levine (4/13/79); and

4. Containment systems for mitigation of degraded core/core melt accidents, NRR-80-4 Denton to Budnitz (7/14/80).

4.5.3 Status of Capabilities

In the past several years, the engineering research needs related to structural behavior were addressed as a part of water reactor safety research or as a limited part of technical assistance work. With a few exceptions, the problems in structural design, behavior, and testing were not formally addressed in sufficient depth. Increased attention, however, has been drawn to the need to resolve the large number of engineering problems reported in operating plants in the areas of mechanical, structural, and materials engineering. This program is addressing the increased research needs to quantify margins of safety in structural design. Research has also been undertaken to better understand (1) problems related to plant behavior during and after earthquakes and other events and (2) other major issues related to design and construction requirements. In addition, attention will be directed to improving plant safety at both operating plants and future plants.

4.5.4 Research Program Objectives

The primary objective of the program is to provide the NRC licensing staff with methods, techniques, and criteria for evaluating structural adequacy in terms of margins of safety and failure probabilities for accident conditions. In addition, the staff must be able to ensure that the engineering behavior of safety-related systems and components meets licensing requirements.

4.5.5 Research Program Plan

4.5.5.1 Design-Load Activity

1. Engineering Characteristics of Seismic Input: It is well known that the peak ground acceleration in the nearfield is not correlated to the magnitude of the event. According to prevailing views, it is safe to design nuclear power plants for less than the expected peak ground acceleration if the duration of the peak motion is sufficiently small. There is, however, no consensus on how an appropriate acceleration value should be chosen.

   The research effort is aimed at defining the input motion characteristics to produce equivalent structural response. Recommendations for choosing effective peak accelerations and considering the effects of rotational motion will be completed in FY 1983. Additional effort concentrated on incorporating improved seismological perceptions of earthquake hazards is anticipated through FY 1984.

Characterization of seismic input by site-specific spectra and the use of a critical excitation technique for independent evaluation by the NRC staff are also long-term goals of this research activity. Additional work in this area will be planned in FY 1983 and initiated in FY 1984.

2. Flood Effects: Current methods and criteria for analysis of flood levels and flood protection are based on deterministic requirements. No assessment is made of the probability of flood conditions postulated. However, the Advisory Committee on Reactor Safeguards and the licensing staff have been interested for some years in assessing the potential for floods to cause severe accidents at nuclear plants. In order to estimate flood risks at nuclear plants it is necessary to characterize the flood elevations at plant sites on a probabilistic basis. Methods are necessary to estimate the probability of exceedance of design basis floods. This resulted in a request for research (NRR-77-16) entitled, "Evaluation of Margins Available in Flood Protection." This project was developed in response to that request.

   This study is currently (FY 1981) being initiated through the competitive procurement process. The primary objective is to identify and develop methodologies to estimate probabilities of exceedance of flood levels and probabilities of radioactive release from nuclear power plants as a consequence of floods, including an assessment of the uncertainties in the estimates. Component probabilities to be investigated include probabilities of floods that may potentially damage plants including the effect of flood related to equipment malfunction and plant internal flooding, probabilities of plant damage given such floods, and probabilities of radioactive release given such damage. Results on the probabilistic methods of flood-level estimation should be available in FY 1983. The next phase of this study will involve specific plants, including the consideration of sites subjected to seiches and tsunami. The latter phase of this study is scheduled for completion in FY 1984. Additional studies in this area beyond FY 1984 will involve using the results and applying them to selected sites in order to determine the margins of flood protection available at these sites.

Additional work on hydrology and flood effects research is also underway in Siting and Environment Research (Chapter 6) and in the Methodology Development (Chapter 11) program under Systems and Reliability Research.

3. Effects of Hydrogen Burning: Deflagration or detonation of hydrogen inside containments has been postulated as an accident scenario. Because rapid hydrogen releases in large quantities were considered unlikely, containment structures were not designed for their possible effects. Since degraded core cooling conditions that generate large quantities of hydrogen are now being considered as accident scenarios it is necessary to develop a thorough understanding of containment behavior.

   This project, initiated in FY 1981 is scheduled to be completed by FY 1983. The objective is to develop a containment behavior model to allow probabilistic assessments of capacities under hydrogen explosions.

4. Load Combinations for Design of Structures: Nuclear plant structures must be able to withstand extreme environmental and postulated plant accident loads, and combinations thereof with normal plant operating loads. The current practice of using load factors to combine loads was based on judgment not on a rational assessment of risks. Therefore, the degree of conservatism is not consistent for different postulated local combinations. Also, it is not clear that the relatively large number of combination are necessary, since some may envelope others. A more rational basis for choosing load combinations and associated load factors is desirable.

   This project was initiated in late FY 1980. The current efforts, aimed at developing load combination criteria for static and dynamic loads, will be completed in FY 1983. Additional effort such as incorporating results from tests and other programs to determine available safety margins will take place through FY 1984.

4.5.5.2 Structural Response Prediction Activity

1. Soil-Structure Interaction (SSI): Recent surveys of the state of the art in SSI analysis indicate that although much work has been done and a number of calculational techniques exist, there is no great confidence in the ability of current modeling techniques to predict the effects of SSI.

   A definitive assessment of the significance of uncertainties about current ability to model SSI phenomena will be made. It will include comparisons of the predictions made by different modeling techniques. Probabilistic models of nuclear plant response will form the bases for the assessment. This effort will be completed in FY 1982. Improvements in seismic design techniques will be developed during FY 1982-1984. Recommendations for improved methods and approaches to be used in licensing reviews will be provided in FY 1985.

   This project is a part of the Seismic Safety Margins Research Program described in Section 4.4 of this chapter.

2. Building Response to Earthquakes: The contribution to the overall seismic risk to a nuclear plant is believed to be dominated by earthquakes of the order of 2 to 3 times the safe shutdown earthquake (SSE). Response of plant structures at that level will be substantially inelastic. Thus, inelastic behavior will be studied. Sensitivity studies that vary the uncertainties associated with the analysis techniques (including elastic and inelastic behavior) will also be performed over a wide range of earthquakes. These studies will provide insights regarding the importance of various predictive methods to the overall risk to the plants. Improvements in seismic design techniques will be developed during FY 1982-1984. Recommendations will be provided for improved methods, and approaches to be used in licensing reviews.

The project is a part of the Seismic Safety Margins Research Program described in Section 4.4 of this chapter.

3. Dynamic Testing tructures: It is necessary to establish that the responses predi...u by analysis bound the response of the as-built plant. Dynamic testing of structures is a potential tool to verify the degree of accuracy of analytical models. Major questions that must be answered include, what tests are feasible, what are the limitations of methods that will utilize the test data, and what confidence levels can be assigned to data obtained from various tests?

   The applicability of testing methods as adjuncts to, or possibly substitutes for, analytical methods to predict structural response or assess structural damage must be examined. The project will develop a critical evaluation of alternative methods of dynamic tests, both preoperational and post accident, and will recommend an experimental methodology that is consistent with available analytical methods. Detailed recommendations will include guidance for the selection of damping values and other sensitive parameters for the analysis of as-built structures. Verification of recommended methods will use existing experimental sources such as data obtained from explosive testing at the Heissdampfreaktor (HDR), preoperational test data obtained from other foreign sources, and relevant nonreactor experiments appearing in the open literature. This effort began in FY 1980 and will be completed in FY 1984.

4. Benchmarking of Computer Codes: Instances of errors in computer calculations used for seismic analysis of piping and structural systems have been reported. In order to ensure safety, the licensing staff deemed it necessary to verify structural design calculations on a selected basis.

   Verification of calculations include (a) verification of the algorithm utilized to perform calculations, and (b) verification of the model used to represent the behavior of safety related structural systems. This requires the use of well verified computer codes. No research effort has been devoted to this area so far.

This project will provide the licensing staff computer codes to (a) check the reliability of codes used by applicants or (b) perform independent safety analysis. No new computer codes will be developed. Existing codes will be adopted for application by the staff. Benchmark problems will be devised to permit an evaluation of code performance. Only existing data will be utilized to formulate the problems. Methods will be included to simplify code input (coordinate and mesh generators, etc.) and evaluate output (graphical displays, plots, etc.) to increase staff efficiency. Work will begin in FY 1981. Operating computer codes and benchmark problems will be available in FY 1983.

4.5.5.3 Structural Performance Activity

1. Safety Margins for Containments: The design of containment structures is generally governed by the pressure and thermal loads assumed to be a part of a loss-of-coolant accident (LOCA). The design procedure is based on elastic methods and, while reliable for the assumed loading conditions, cannot be relied [illegible] for loads in excess of the design basis. A containment building can be [illegible]ected to perform satisfactorily at pressure and temperature combinations more demanding than those for which it was designed; however, no reliable estimate of the actual loading associated with unsatisfactory performance can be made because of a lack of experimental results. Strictly analytical approaches cannot reliably predict ability to retain pressure.

   This project will emphasize the development of an analytical or semiempirical method to predict the ultimate capacities of concrete (reinforced and prestressed) and steel containments subject to accident and extreme environmental loadings. Separate effects types of experiments will be used to verify these methods by investigating specific and identified concerns (i.e., certain penetrations, critical containment portions, etc.) using controlled and measured boundary conditions. Differences in design and construction practices from changes in percentage of reinforcement, material yield strength, and detailing of large penetrations are among many factors that will be considered in planning these experiments.

A test plan will be developed in FY 1981. Analytical development will also begin in FY 1981. Small-scale (1/15 to 1/25) tests will begin in FY 1981 and continue into FY 1982. Larger scale tests (1/4 to 1/6) will begin in FY 1982 and continue through FY 1986.

2. Safety Margins for Category I Structures: The buildings, other than containments, that house safety-related equipment at nuclear power plants are often heavy, concrete shear-wall structures. These buildings utilize heavy exterior walls, and the internal load distribution in these heavy Category I buildings differs from that encountered in framed structures. Consequently, the analytical and experimental evidence developed over the years for framed structures is not directly applicable.

   This project is aimed toward assessing the margin of failure for common classes of nuclear Category I structures. The main method used for this assessment will be an analytically supported and carefully planned experimental program. Accounting for energy dissipation by means of equivalent viscous damping or other improved criteria will receive special attention. The objective is a better understanding of: (a) damping behavior, (b) characterization of damping values due to cracks and loose joints, and (c) nonlinear and cyclic response from large dynamic loads. This project will result in methods that can be used to verify predictions of Category I structures under design loadings and to estimate the margins of safety available to accommodate loadings outside the design basis.

   Initiated in late FY 1980, this project is expected to continue through FY 1986, with experimental testing of small-scale structures being completed in FY 1984.

3. Buckling of Steel Containments: The current standard methods for determining the buckling loads of steel containment vessels that are subjected to unsymmetrical dynamic pressure loads have not been verified by testing or accurate analysis. A testing program has been established to check the accuracy and applicability of methods proposed to predict buckling loads for containment shells. Coordination with ongoing analytical prediction programs will be established to ensure that the experimental design will be able to encompass all the variables thought to be significant. A comparison of analytical and experimental results for shell-buckling predictions will be provided. The adequacy of the predictive methods for ranges of variables significant in containment design will be summarized. This project was initiated in late FY 1980 and is scheduled to be completed in FY 1985. Scaled model testing is expected to be completed in early 1984.

4. Adequacy of Codes and Standards: The design of concrete containments is governed by Division 2, Section III, of the ASME Boiler and Pressure Vessel Code. The design of other Category I concrete structures is governed by American Concrete Institute (ACI) Standard 349-76, "Code Requirements for Nuclear Safety-Related Structures." Provisions in these codes and standards are based on a consensus judgment; they are not based on specific experimental evidence. Areas of particular concern are those related to requirements for (a) resisting tangential and peripheral shear in reinforced concrete containments and (b) ensuring adequate ductility in reinforced concrete structures under dynamic loading.

   Experimental evidence is needed to assess the adequacy or degree of conservatism in code requirements. Tests will be conducted on large and small specimens. Large specimens will be used to simulate full-size reinforcing bars; small specimens will be used to evaluate the effects of different variables.

   This project is scheduled to be initiated in FY 1981 and completed in FY 1984. Results of this study will include recommended values of permissible shear and ductility limits.

5. Effectiveness of Quality Assurance and Inspection Procedures: The current practice relies mainly on quality assurance and control procedures to ensure adequate construction. It is an outgrowth of the practice used in conventional buildings where the emphasis is on assuring a serviceable structure. However, quality control and assurance procedures for nuclear power plants must be oriented to accomplishing the safety objectives associated with a building or a structure.

This project will focus on several tasks. The NRC staff must be able to state that the plant as-built--or in the case of an operating plant, as existing--is structurally adequate for public health and safety. An assessment will be made of the contribution of nuclear quality assurance and quality control in civil/structural construction. Recommendations should be made regarding nondestructive examination techniques and any additional procedures that may be necessary to ensure safety. Finally, specific recommendations will be made for achieving improvements in design criteria, material specifications, construction tolerances, acceptance standards, and reinforcement and structural steel fabrication details. This project will result in recommendations for independent inspections by the NRC staff that can markedly increase the assurance of plant safety.

This project is scheduled to be initiated in FY 1982 and to be completed in FY 1986.

6. Safety Evaluation Methods and Acceptance Criteria: There is currently no guidance or procedure for the inspection of plants following a severe earthquake. Heavy structures such as those in a nuclear facility may sustain damages that are not visible, but will significantly reduce margins of safety against future earthquakes and other plant operational and accident loadings. A systematic procedure for plant inspection and associated acceptance criteria will be developed under this activity. Other elements under this project will include development of simplified evaluation tools in terms of interactive computer codes; preparation of evaluation and acceptance criteria for the review of spent fuel pools, relevant load combinations, and leak tightness requirements; and development

of manuals for safety evaluation of nuclear structures. Development of criteria for plant inspection following a severe earthquake will be initiated in FY 1983. Other activities associated with this project will begin in FY 1982 and continue through FY 1986.

7. International Cooperative Program: Research conducted in the U.S. on the structural safety of nuclear plant facilities has concentrated on analytical predictions of structural behavior or testing of structural elements on a modest scale. Design tools used for nuclear plant structures have, for the most part, never been verified by comparison to actual test data. The most promising sources of relevant data are the tests being performed in Europe and Japan. Engineering practice there, especially in Japan, tends to put more reliance on large-scale test data than does U.S. practice.

   Current activities are limited to participation in the HDR project being conducted in the Federal Republic of Germany. Low-level dynamic tests, intended to simulate response to earthquake loading, were carried out in 1975 and 1979. Current effort is being devoted to determining how well the response of the reactor building could be predicted using current U.S. computer codes. It is anticipated that a cooperative program with Japan will permit the U.S. computer codes to be checked against data generated at the large-scale shake table that is to be completed in 1982. The possibilities of cooperative ventures with other Japanese and European test programs are being explored. This project is scheduled to begin in FY 1981 and continue through FY 1986.

4.5.5.4 Results

1. Design Loads Acitivity

FY 1982 Preliminary recommendations for adjustment of free-field instrumental spectra to obtain design spectra as a function of significant ground motion characteristics such as energy content; analytical assessment of containment capacities and identification of failure modes under internal explosions.

FY 1983 Develop concrete containment model to characterize behavior under internal explosions; develop methodology for various event combinations.

FY 1984 Define seismic input at foundation level to produce equivalent structural response; assess probabilistic methods of flood-level analysis.

FY 1985 Assess probabilities of radioactive release from a selected nuclear power plant as a consequence of flooding; recommend load-combination criteria for selected structures.

FY 1986 Assess uncertainties associated with radioactive release probabilities as a result of flooding; develop acceptance criteria associated with specific load combinations for selected structures.

2. Structural Response Prediction Activity

FY 1982 Assess uncertainties due to earthquakes in SSI effect and structural response prediction; develop benchmark problems for SSI analysis; evaluate dynamic testing methods for structures; develop benchmark problems for selected structural computer codes.

FY 1983 Analyze the sensitivity to SSI-effect and building response uncertainty of the radioactive release probability from earthquakes; develop selected structural computer codes for independent verification by the staff.

FY 1984 Recommend risk-based methods of structural safety evaluation under earthquake effects; develop acceptance criteria for dynamic testing of structures; verify selected benchmark solutions for structural computer codes.

FY 1985 Recommend extrapolation of results of dynamic testing of structures from low to higher level motion; recommend benchmark problems and solutions for verification of structural computer codes used for seismic analysis of selected structural types.

FY 1986 Complete recommendations for dynamic testing of structures and acceptance criteria; complete recommendations for benchmarking of structural computer codes, including development of computer codes suitable for use by the staff.

3. Structural Performance Activity

FY 1982 Complete test plan for structural safety margins programs; complete selected tests for benchmark problems for containment buckling.

FY 1983 Perform tests on containment and other Category I structures; assess nonlinear methods of analysis for prediction of buckling loads through correlation with tests; assess methods for nondestructive examination of concrete.

FY 1984 Assess methods of analysis for their ability to predict structural behavior at limit states through correlation with test data; recommend methods of analysis for predicting buckling loads; recommend acceptance criteria for seismic and peripheral shear under combined biaxial tension, including deformation limits for liner plate integrity; recommend nondestructive examination methods for concrete suitable for use by the NRR staff.

FY 1985 Assess safety margins available in containment and other Category I structures to accommodate loadings outside the design basis; recommend reduction factors applicable to analytical prediction of buckling loads; develop simplified input-output techniques for computer analysis of safety-related structures; publish a catalogue of test data developed through the International Cooperative Program.

FY 1986 Verify predictions of safety-related structures under design loadings by means of test data; recommend independent inspections by the NRC staff that can significantly improve assurance of plant safety; complete manuals for review and evaluation of selected structural components.

4.6 Fracture Mechanics

For the licensing staff to decide on the present and continuing safety of critical structures such as reactor pressure vessels and piping systems in LWRs under normal operation and under abnormal accident conditions, they must have verified analysis methods and a solid background of experimental data upon which to base those judgments. It is the role of this program to provide both the analytical methods and the experimental data to back up pertinent sections of the ASME Code (Section III and XI) as well as licensing decisions on the safety of pressure vessels under accidents such as thermal shock and for piping systems to establish the likelihood of either leak or break and the consequences of such breaks. The importance of this work is that fracture of the pressure vessel or a significantly large break in a major pipe could result in the inability to keep the fuel core covered with coolant, thus leading to a core melt accident. In the staff's attempt to determine specific plant compliance with the General Design Criteria, the ASME Code and various federal regulations and regulatory guides related to vessels and piping systems, they require information on material properties, crack growth rate, stress and load distributions and the influence of flaws of different sizes on integrity under normal and accident loadings. The end product of this research is a series of analytical procedures incorporating these data, which support or modify the requirements in the ASME Code, for use by the staff in evaluating the continuing safety of the nuclear plant primary system pressure boundary, and to be used for guidance in the design of new systems.

4.6.1 Regulatory Objective

The regulatory objective of this unit of work is to provide information that will confirm or deny the assumption that pressure vessel and primary system piping failure are very low probability events and that there will be some evidence such as a leak that will give a warning prior to a break. It is also the objective of this work to expand the state of the art dealing with fracture mechanics and structural integrity and to see that this information is introduced into the applicable government regulations and ASME codes dealing with the design and analysis of nuclear components.

4.6.2 Technical Capabilities Required

The required technical capabilities in the area of fracture mechanics, now as in the past, are being developed through cooperation among staffs of the user offices, primarily NRR and SD, and the staff of the RES Metallurgy and Materials Research Branch. The needs so determined include:

1. Development of a test procedure for establishing the fracture material properties for steel on the upper shelf of the material toughness curve;

2. Validation of linear-elastic fracture mechanics as an accurate methodology for analyzing pressure vessel fracture behavior when the material of the vessel is on the lower-transition fracture toughness zone;

3. Development of an upper-shelf fracture toughness data base for irradiated and unirradiated low-upper-shelf weld and plate material;

4. Development of a methodology and the experimental verification of such a methodology for the application of elastic-plastic fracture mechanics to thick-walled pressure vessels whose material is in the upper-transition or in the upper-energy-shelf fracture toughness zones;

5. Experimental validation of the "J" integral and "Tearing Modulus" methods for elastic-plastic analysis of degraded piping and pressure vessels;

6. Development of a larger data base for inclusion in or modification of the K_{IR} curve of the ASME Boiler and Pressure Vessel Code;

7. Development of a methodology for accurately assessing the consequences of pipe rupture; and

8. Development of a probabilistic model of piping reliability to help redefine the criteria for the location and number of pipe restraints.

Specific user requests are:

1. Toughness data on irradiated low-upper-shelf weld metal, NRR-1975;
2. Protection against pipe rupture, NRR-77-21;
3. Experimental verification of the tearing stability concept, NRR-77-24; and
4. Capability of degraded piping to withstand accident loads, NRR-79-24.

4.6.3 Status of Capabilities

The structural integrity of the reactor pressure vessel is very important. If the pressure vessel fails, the results could be catastrophic. This concern has long been recognized. Research started in the early 1960s under the AEC was directed to ensure that such an eventuality did not occur. Intensive research since then has resulted in the validation of linear-elastic fracture mechanics as a fracture-analysis methodology for pressure vessels having sufficient constraint, under elastic stresses, and at or below the transition temperature. This work also led to the establishment of fracture toughness requirements for reactor vessel steels and weldments which, if adhered to, would ensure the fracture-safe operation of nuclear pressure vessels. These requirements were incorporated in 10 CFR Part 50 and Regulatory Guide 1.99 and the ASME Codes. Because of the limitations of the linear-elastic fracture mechanics methodology, there is a large, but as yet undefined, element of conservatism in the requirements for fracture toughness when the reactor vessel steel is operating at or near the upper-shelf-toughness-energy zone. In this area, the vessel material is highly ductile, so that the linear elastic fracture-mechanics methodology cannot be adequately used to predict the fracture performance of the vessel. This inadequacy has taken on greater importance in recent years with the discovery that some of the early reactor vessels were fabricated with weld material whose irradiated toughness will eventually--during the operating life of the reactor vessel--fall below the upper-shelf-toughness limits set by 10 CFR Part 50 and Regulatory Guide 1.99. Thus, in the late 1970s, it became the aim of this research effort to develop a new fracture methodology to establish criteria for fracture-safe operation under elastic-plastic and fully plastic conditions.

At present, the fracture toughness criterion is governed by the reference fracture toughness (K_{IR}) curve of the ASME Boiler and Pressure Vessel Code. This curve defines the most conservative limits (or the so-called lower bound) of all available valid data on the slow-load, rapid-load fracture toughness and crack-arrest toughness for specimens up to 12 inches thick. Because only a few heats of steel were used to obtain the data for this original K_{IR} curve, additional data from more heats and more product forms of material are required, and they are being obtained. Because of the difficulty and expense of testing very large specimens, a minimum number of 4-inch-thick specimens are being tested in the irradiated condition, with a larger number of smaller specimens being used to establish the specimen size correlation.

The two major elements of research on response to accident transients deal with the response of the reactor pressure vessel to (1) large LOCAs and (2) small-pipe LOCAs, including main steam line breaks and other overcooling, pressurized transients. The first is characterized by an abrupt loss in pressure in the reactor pressure vessel, with the subsequent injection of large amounts of cold water by the ECC system. The second is characterized by a more gradual loss of pressure in the reactor pressure vessel, injection of water by the high pressure injection system (HPIS), possible injection by the ECC system, and a gradual repressurization of the reactor pressure vessel. Much work has been done since 1974 regarding the first condition, i.e., a large LOCA. A fracture-analysis methodology will have been developed and validated in FY 1981 using six thick-walled (6 inches thick) cylinders to permit a rational integration of all pertinent factors. It has been shown that linear-elastic fracture mechanics does characterize crack initiation, propagation, and arrest for the nonpressurized, severe thermal-shock conditions for large LOCAs. The most recent test demonstrated the beneficial effects of warm prestressing, a phenomenon that will, under certain conditions, severely limit crack initiation and thus further add to the maintenance of the structural integrity of the vessel. The last test of the series, planned for FY 1981-1982, is to demonstrate the applicability of all prior work to a vessel that more closely simulates in its geometry a real reactor vessel.

Piping systems are complex, beginning at the reactor vessel as nozzles and including tees, reducers, elbows, junctions, and straight runs of various thicknesses, diameters, and materials. Because the potential accident conditions are so varied, many supports, hangers, snubbers, and restraints are used. In recent years, particularly as a result of increasing attention given to protection against seismic events, more restraints of this nature have been added to the piping system. This has led to a significant lessening of piping system flexibility, with a potential reduction in its ability to safely respond to accident-type loadings.

4.6.4 Research Program Objectives

The objectives of the fracture mechanics program are:

1. To develop and validate experimentally fracture-analysis procedures and design criteria for predicting the stress levels and flaw sizes required for crack initiation and subsequent propagation and arrest in LWR pressure vessels and primary piping under elastic, elastic-plastic, and fully plastic conditions;

2. To show that slow-load fracture toughness, rapid-load fracture toughness, and crack-arrest-toughness results obtained from small laboratory specimens are truly representative of the toughness characteristics of the material behavior in pressure vessels and piping in both unirradiated and irradiated material conditions;

3. To provide definitive experimental validation for the analytical methods used in the prediction of crack initiation, propagation, and arrest that

could occur in a hot reactor vessel subjected to the injection of cold ECC water following a large or small LOCA;

4. To provide experimental and analytical procedures for determining the structural adequacy of the LWR reactor pressure vessels (RPVs) subject to postulated overcooling transients such as a main steam line break; to critically examine the entire generic area of the RPVs subjected to a pressurized thermal shock;

5. To develop analytical procedures and experimentally verify procedures for the analysis of degraded (cracked) pipe subjected to normal and upset loading conditions; to critically examine present criteria for postulated pipe rupture and to evaluate the leak-before-break concept for nuclear piping systems; and

6. To develop and experimentally validate analytical procedures for the evaluation of postulated post-pipe-rupture consequences; to critically examine present criteria for piping system restraint locations and configurations.

4.6.5 Research Program Plan

4.6.5.1 Vessel Performance

Recent achievements--particularly those dealing with the concepts of the "J" integral and "Tearing Instability"--offer significant promise for developing a new fracture methodology to establish criteria for fracture-safe operation under elastic-plastic and fully plastic conditions. Using these achievements (as well as new advances in the prediction of elastic-plastic stress states at the tip of a crack by means of three-dimensional finite element analysis), procedures for analysis and prediction of ductile fracture are to be completed by FY 1981. A multifaceted experimental program that will validate this methodology as well as develop the material property data necessary for its implementation is planned for completion by FY 1985. This work, being carried out at several laboratories and universities, involves the development of suitable small specimen tests, by FY 1982, for the determination of upper-shelf (ductile) tearing-material properties representative of thick-section pressure vessel behavior, for both unirradiated and irradiated materials. During the period FY 1983-1984, standardization of small-specimen tests and the development of a sufficient data base for all materials of interest will be achieved. Particular emphasis will be placed on the completion of a data base for ductile fracture toughness of irradiated low-energy upper-shelf plate and weld material by the end of FY 1983.

Validation of the elastic-plastic methodology will be carried out in the period FY 1983-1985 by three intermediate test vessel (ITV) tests conducted to determine initiation, propagation, and arrest of ductile tearing as could be experienced under both normal and postulated accident conditions. By the end of FY 1985, the correlation between toughness parameters development from small-scale (Charpy) specimens and from ductile-fracture specimens will be completed. This accomplishment will permit the development of recommendations for vessel analysis based on surveillance specimen Charpy values and elastic-plastic fracture mechanics methodology. These results will also allow, in FY 1985, recommended revisions to Regulatory Guide 1.99 dealing with the effects of radiation on pressure vessel steel. Continuing work in FY 1985-1987 will be focused on building a data base of fracture toughness for higher-strength pressure vessel steel, both in the unirradiated and irradiated conditions.

4.6.5.2 Toughness Characterization

Testing is now possible for development of data in the upper-transition zone and the upper-shelf-energy zone (of particular emphasis in Section 4.6.4.1) of the material toughness curves, using newly developed J-R curve test procedures. This test development led to a new critical examination of the mechanisms of the transition of metal fracture from a purely brittle (cleavage) fracture to a purely ductile (fibrous) fracture; it resulted in the initiation, during FY 1981, of a combined analytical and experimental program at several laboratories and universities that is aimed at improving appropriate small-specimen-developed J-R curve test procedures by the end of FY 1983.

A significant part of the FY 1983 work will be experimental verification of the model for fibrous-cleavage fracture as well as of the statistical model for small-specimen testing in the transition toughness zone. The cleavage-fibrous transition work will also permit finalization of a crack-arrest specimen for use in vessel surveillance capsules. The data base for the fracture toughness of unirradiated, present-practice reactor vessel steel in the transition zone will be completed in FY 1984. This will be followed in FY 1985 by the completion of the data base for irradiated fracture toughness of the same materials in the transition zone. Continuing work in FY 1986-1987 will concentrate on development of transition zone toughness data for new high-strength steels proposed for reactor pressure vessel construction, as well as development of revisions to the ASME K_{IR} curve for current and proposed higher-strength materials.

4.6.5.3 Response to Accident Transients

With the completion of the first series of tests on the response of the reactor pressure vessel to large LOCAs and to small-pipe LOCAs, the program has been redirected to address the generic problem of pressurized thermal shock. In this series of postulated accident transients, the structural adequacy of the RPV may be compromised by its repressurization, even though the thermal shock to the reactor pressure vessel is less severe than that for the large LOCA. In analyzing this type of accident transient, a complicating element in many cases appears to be that the stress conditions induced in the vessel wall are large, while the temperatures are such that most of the wall material remains in the upper-transition and upper-energy-shelf fracture toughness areas. In these areas, linear-elastic fracture mechanics methodology is inadequate to describe either the initiation or the arrest of a crack. The required analysis methodology will therefore have to rely on the currently developing elastic-plastic and fully plastic fracture mechanics methodology.

Early in 1982 an analytic model in the form of a computer code will be completed and delivered to the licensing staff for use in evaluating safety reports concerned with overcooling incidents. The model will be validated in a series of pressurized thermal shock tests starting in FY 1982 and extending through

FY 1984. These tests will be conducted on reactor pressure vessels that incorporate various levels of material toughness and appropriately sized flaws. These vessels will be subjected to pressure and temperature gradients that could be expected from postulated overcooling transients, small-pipe breaks, and main steam line breaks. In FY 1985, recommendations will be made to the licensing staff on a validated methodology for analyzing postulated pressurized thermal-shock accident scenarios to reactor pressure vessels, especially under higher-temperature conditions where linear-elastic fracture mechanics is not applicable.

4.6.5.4 Piping

A three-part research program has been developed to deal with piping problems: (1) using both deterministic and probabilistic methodology to develop the most probable location for a pipe break (perhaps permitting overall reduction in the number of restraints while at the same time optimizing their location); (2) determining critical crack size in a pipe component of the piping system and whether or not a pipe will leak detectably before a break could occur under normal operating or accident or upset conditions; and (3) determining the consequences of a postulated pipe rupture, particularly with regard to the surrounding piping and safety systems (helping to establish a more rational basis for number and location of piping restraints).

The work being done and planned for these three parts comprises both development of analytical methodologies and experimental validation of these methodologies, as well as large-scale proof tests of inservice-degraded piping. Both linear-elastic fracture mechanics and the newly developed elastic-plastic fracture mechanics methodologies will be employed where applicable. Small- and large-scale pipe tests will be conducted through FY 1984 to validate the analytical procedures. To be completed in FY 1981-1982 is a new finite element code capable of analyzing piping systems following a postulated pipe break. By the end of FY 1983, sufficient information on effects of jet force impingement, pipe whip, and requirements for piping restraint will be generated to allow recommendations for an initial revision of Regulatory Guide 1.46 and 10 CFR Part 50.

During the period FY 1978-1982, experimental efforts carried out in Europe and Japan have been and will be relied upon for data to validate our analytic efforts in the area of jet impingement and pipe whip. After extensive analysis, it has been determined that these foreign programs have not, and will not, provide sufficient data for an in-depth evaluation of our newly developed pipe whip analytic tool. To meet this need, it is planned to initiate late in FY 1983 a small-scale (3-in-diameter piping) test facility program to give the information required for this validation process. It is planned that this work will be completed and that a final revision of Regulatory Guide 1.46 will be prepared in FY 1985.

At the end of FY 1982, large-scale testing of inservice degraded piping will begin. Piping containing cracks taken from actual plants will be tested under normal and postulated accident conditions to determine their margins of safety. Newly developed analytical methods that can be used to predict these margins of safety will be extended and validated. In FY 1986, this program will result in the submission to NRR of recommended analytical procedures for use in the licensing process and in evaluating the performance of degraded piping under normal operating and postulated accident conditions. This work should form the basis for the possible establishment of a "Leak Before Break" criterion for certain elements of a nuclear power plant's piping. Necessary for this work will be the completion in FY 1984 of a large data base for piping material toughness in the elastic-plastic toughness zones.

The final planned element of the piping program is the development, by FY 1987, of a comprehensive interactive computer code and analysis procedure for evaluating piping systems by the licensing staff. This code will give the staff the capability to rapidly call up any part of a piping system on a CRT terminal and to analytically subject the piping system to any postulated loads for evaluating design compliance with construction codes and regulatory guides.

4.6.5.5 Results

The research program plan is presented below in terms of milestones by fiscal years.

1. Vessel Performance

FY 1982 Data base on ductile fracture to ıness of irradiated low-upper-shelf-energy welds will be completed; advanced analytic methodology for determining stress intensities for nozzle-corner cracks under pressure and thermal loads will be completed; nozzle-corner crack weld repair methods will be tested and evaluated; first intermediate test vessel (ITV) test for validation of elastic-plastic-fracture methodology in reactor pressure vessels will be completed.

FY 1983 Data base on ductile fracture toughness of irradiated low-upper-shelf-energy plate material will be completed; second ITV test for validation of elastic-plastic fracture methodology will be completed.

FY 1984 Correlation between Charpy energy and upper-shelf fracture toughness will be completed.

FY 1985 Third and last ITV test for validation of elastic-plastic fracture methodology will be compl^+ed; recommendations con.erning Charpy energy levels, upper-shelf fracture mechanics methodology and surveillance specimens for inclusion in Regulatory Gu'de 1.99 will be made; initial data on fracture toughness of higher-strength steel (unirradiated) will be compiled.

FY 1987 Initial data on fracture toughness of higher-strength steel (irradiated) will be compiled.

2. Toughness Characterization

FY 1982 A model to explain cleavage-fibrous fracture performance in thick-walled pressure vessels operating in the upper-transition range will be developed; a statistical model for small-specimen testing in upper transition will be develo.ed; ASTM crack-arrest-standardization round robin will be completed.

FY 1983 Cleavage-fibrous fracture model will be experimentally confirmed; a statistical model for small-specimen testing in upper transition will be experimentally confirmed; development of a surveillance capsule crack-arrest specimen will be completed.

FY 1984 Initial data base for present reactor materials (unirradiated) in upper transition will completed; recommendations will be made to ASTM for a standardization test for small specimens in the upper-transition zone.

FY 1985 Fracture toughness data base for present reactor materials (unirradiated and irradiated) in the upper-transition zone will be completed.

FY 1986 Initial fracture toughness data base for new materials (unirradiated) in the upper-transition zone will be completed.

FY 1987 Recommendations for revision of K_{IR} curve for current and higher-strength materials, plus consideration of effects of irradiation, will be completed.

3. Response to Accident Transients

FY 1982 Evaluation of series of thermal-shock experiments will be completed; an analytical model for use in evaluation of safety reports dealing with unpressurized thermal shock will be recommended to NRR; first pressurized thermal-shock experiment will be completed.

FY 1983 Second pressurized thermal-shock experiment will be completed.

FY 1984 Third pressurized thermal-shock experiment will be completed.

FY 1985 Evaluation of series of the pressurized thermal-shock experiments will be completed. Recommendations will be submitted to NRR as to a methodology for analyzing pressurized thermal shock for pressure vessels whose material remains in the upper-transition or upper-shelf fracture toughness zones.

4. Piping

FY 1982 Pipe whip studies will be completed; NRR will be supplied with a working methodology for assessing the consequences of a pipe rupture; development of a two-phase jet model, to replace the "Moody Model" specifically cited in 10 CFR Part 50, will be completed; probabilistic study of piping reliability will be completed; first set of experiments on intermediate-size (8-inch-diameter) piping for the validation of "tearing instability" concept will be completed; testing of large-diameter field-degraded piping will begin.

FY 1983 A pipe-to-pipe impact study will be completed; recommendations for revision of Regulatory Guide 1.46 will be submitted.

FY 1984 Extended fracture toughness data base for piping material will be completed.

FY 1985 Testing of large-diameter inservice-degraded piping will be completed; margins of safety for various flaw geometries will be established; final recommendations for revision of Regulatory Guide 1.46 will be submitted.

FY 1986 Evaluation of all piping test programs will be completed; recommended analytical methodology will be submitted to NRR for evaluation of degraded piping under normal operating, upset, and accident conditions.

FY 1987 Interaction piping code will be developed for staff review of pipe design compliance.

4.7 Operating Effects on Materials

This section addresses problems that exist in reactors in which the original component properties have become degraded or otherwise are less capable of assuring integrity under normal operational or accident conditions. A primary thrust of this research is to establish clearly the reasons for the degradation, flaw growth, and so forth; then to establish the ranges of values of these parameters; and finally to provide the data necessary to set limits on the degree of degradation that is acceptable or to determine ways that such degradation can be reversed or eliminated. Several specific areas are addressed, including the effects of neutron irradiation (wherein the reactor pressure vessel steel becomes embrittled during service, thus rendering it far less capable of resisting a thermal-shock accident). The corrosion-degradation and cracking of steam generator tubing and the effects of water chemistry and operating environment on the stress corrosion cracking of carbon and stainless steel piping are also addressed. These kinds of topics are the ones that deal directly with cracks, corrosion, embrittled steel and other areas that have a direct impact on the safety of the component in a nuclear system. The monitoring, control or limitation, or elimination of such known degradation is the aim of this program.

4.7.1 Regulatory Objective

The overall objective of this program is to determine the ways and the extent to which the LWR environment (including temperature, stresses, coolant, and radiation) changes and degrades the materials and components of the primary pressure boundary and safety-related systems, thereby reducing their capability to maintain integrity under normal operating, upset, and accident conditions. The research programs develop information on specific materials and systems for direct application and licensing decisions. The programs also develop data banks and trends, as well as the ranges of identified critical parameters leading to materials degradation; this information is used by the staff as background for specific and generic licensing decisions.

4.7.2 Technical Capabilities Required

The required technical capabilities in the area of operating effects is well spelled out by the listing of user requests below. The needs include an understanding of the effect of neutron irradiation on the fracture toughness of pressure vessel steel, which is known to become significantly embrittled in service, as well as a method of accurately calculating and predicting the neutron fluence and embrittlement in reactors, based on surveillance program results. Three separate user requests are concerned with the integrity or degradation of steam generator tubing and with the processes within steam generators that cause the cracks and degradation. More recent emphasis is being placed on the causes of cracking in BWR and PWR piping, especially the environmental parameters. Finally, because the coolant chemistry environment is so critical, studies of both the PWR secondary water chemistry and alternative BWR water chemistry schemes are planned. Specific user requests are listed below:

1. Toughness data on irradiated low-shelf weld metal, NRR-1975;
2. Steam generator tube burst strength and integrity, NRR-76-3;
3. Determination of susceptibility of SCC of Inconel-600 tubing, NRR-77-17;
4. Study of a replaced steam generator for various causes and forms of degradation, NRR-78-15.
5. Environmentally assisted cracking in LWR piping systems, NRR-1979.
6. LWR pressure vessel surveillance dosimetry improvement program, SD-1980.

4.7.3 Status of Capabilities

The important effects of neutron bombardment on reactor vessel beltline structural materials include an upward shift in the reference nil-ductility transition temperature by several hundred degrees Fahrenheit, a reduction in ductile shelf-level energy absorption strength, and a reduction in tensile ductility. Initial correlations have been made between results from irradiated small specimens and those from 2- and 4-inch-thick compact specimens tested in 1975.

It has been shown that neutron-induced embrittlement in ferritic pressure vessel steels can be significantly reduced simply by complying with the draft ASTM recommendation for upper limits of 0.10 wt percent copper and 0.012 wt percent phosphorus in the chemical composition of the steel. Furthermore, a mechanism by which copper affects neutron embrittlement in steels has been proposed. Systematic studies of postirradiation heat treatment at temperatures above the operating temperature of irradiated steel have shown that much of the preirradiation toughness can be recovered in this way to extend the useful life of reactor vessels with renewed fracture toughness capability.

Procedures currently exist for determining the neutron flux and fluence from surveillance irradiations in both experimental test reactors and power reactors. However, the variance associated with the calculations and application/correlation procedures is greater than uncertainties in the associated mechanical property measurements. Techniques and procedures developed for fast-reactor neutron dosimetry analysis are now being applied to LWR neutron dosimetry.

A preliminary test program was completed in FY 1979 to establish the most severe, but still realistic, loading and test parameters for the subsequent main test program. Testing is continuing under the selected conditions of cyclic wave form loading ramp time, loading hold time, and minimum-to-maximum loading ratio. To be completed in FY 1983 is an evaluation of long hold times on crack propagation in pressure vessel steels and heat-affected zones. Following on from this work will emerge, in FY 1984, information on the existence and magnitude of threshold levels for initiation of crack growth in these materials. Finally, all the results are to be incorporated into code recommendations for revision of the S-N curves for ferritic steels in the ASME Code Section III to reflect the environmental effects of factors (water chemistry, loading, hold time, etc.) studied in this program. Because of the large number of possible material and environmental conditions, an international cooperative program has been established. Under this program, researchers from all over the world meet periodically to compare and contribute data in a common effort to maximize results and efforts in this area.

Stress-assisted intergranular corrosion cracking in the BWR coolant environment continues to occur in seamless small- and intermediate-diameter austenitic steel piping. The primary factors causing this phenomenon are known. They include oxygen in the coolant, high stresses, and sensitization of the stainless steel. The exact combination of factors that actually produces cracking has not yet been conclusively established.

Cracking has also been observed in PWRs, especially in steam generator feedwater nozzles. These problems have been studied, with regulatory positions set forth in Pipe Crack Study Group Reports, NUREGs-75/067, -0531 and -0691.

Environmental effects in steam generators have caused wastage, cracking, and denting of tubing. Denting is particularly insidious because the large eddy-current signal from the dent itself precludes detection of any other degradation in the dented region. The large strains in the dented regions have caused stress-corrosion cracking from the primary side, and the cracking has been detected only after tube failure. The present concepts of the influence of crack size, wastage, and denting on tube integrity during both primary and secondary system overpressures are being validated.

Failures in nuclear steam turbines have recently begun to occur, apparently as a result of stress-corrosion cracking and corrosion fatigue in rotors and blade keyways. The interplay of metallurgical, fabrication, and water chemistry factors causing these failures is not understood at this time.

Continuing research by industry and EPRI over the past 5 to 6 years has focused on measurements of the concentration of water chemistry species and on pH and conductivity variations in LWRs as a function of reactor operating history; several BWR and PWR systems have been studied and results reported. In the course of this work, some new or improved monitoring instrumentation has been developed. In addition, PWR steam generator operating experience has indicated the need for improved control of water chemistry to prevent or mitigate corrosion degradation in steam generators.

When pipe fabrication and welding are in accordance with code and regulatory guide procedures, integrity problems still arise. For example, when regulargrade stainless steel piping is not solution-annealed in the mill or when the piping is repeatedly rewelded (either during fabrication or during subsequent repairs), significant metallurgical changes can be brought about in the pipe. These can result in initiation of intergranular stress-corrosion cracking.

The interplay of different welding parameters, metallurgical, and fabrication factors on continuing pipe integrity are not well established at this time.

4.7.4 Research Program Objectives

More specific objectives include:

1. Development of changes in the fracture toughness of steels, welds, and components that result from radiation, thermal aging, and environmental embrittlement and development of methods for mitigating this embrittlement, including annealing;

2. Development of methods for calculating, measuring, and predicting neutron flux and fluence in vessel surveillance capsules and the vessel wall itself;

3. Establishment of crack-growth rate of reactor pressure vessel steel, welds, and piping;

4. Identification and studies of the environmental parameters that cause cracking in PWR and BWR piping systems;

5. Studies of the environmental factors that cause degradation of steam generator tubing, support plates, and tube sheet;

6. Studies of the factors causing failures in turbines;

7. Evaluation of BWR and PWR water chemistry parameters and of the effects of changes caused by operating and offnormal conditions; and

8. Evaluation of repair welding in piping.

4.7.5 Research Program Plan

4.7.5.1 Irradiation Effects and Annealing

Irradiated specimens as large as 4 inches thick are to be tested; these results are to be correlated to the results of smaller specimens--such as Charpy-V specimens--that are contained in vessel surveillance capsules; this is to be done by the end of FY 1981. The correlation step is important; the only information available by which the progressive decrease in toughness throughout vessel life can be judged is from the small surveillance specimens. The materials of interest are the weld metals fabricated several years ago in which the weld-wire and flux combination resulted in relatively low upper-shelf toughness (about 75 ft-lb) for the as-fabricated condition. Through irradiation, the upper shelf is reduced to levels at or below 50 ft-lb. A drop below this level is significant because (1) it violates the Federal regulation in 10 CFR Part 50 and (2) it may signal that the toughness needed for vessel integrity may have dropped to an unacceptably low level. The NRC program aims to establish the real toughness associated with the 50 ft-lb criterion by the end of FY 1981 and the level of toughness needed for vessel integrity by FY 1982.

The program on annealing is concerned with recovery of the preirradiation toughness (and recovery of the preirradiation transition temperature) through postirradiation heat treatment. It has been well established that heating of the irradiated steel at 100° to 200°F above the irradiation temperature will effect a significant recovery of the preirradiation properties. Factors influencing this recovery are the time and temperature of annealing; however, considerations affecting these factors pertain to the overall systems, i.e., will the vessel and piping become warped because of the high heat, and must the core and internals be removed for such an operation?

All these factors are being investigated in the program. It has also been determined that when the steel is again exposed to irradiation, reembrittlement will occur. The rate and degree of reembrittlement as a function of total fluence is being investigated. Initial systems considerations should be available by FY 1982; the evaluation of the embrittlement-annealing situation for specific plants will be established in FY 1984 and onward as required; data on irradiation effects for newer higher-strength steels will be underway in the FY 1985-1987 time period.

4.7.5.2 Surveillance Dosimetry

The objective of this program, which started in 1977, is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effects of neutron exposure to LWR pressure vessels. The focus of the efforts underway is the power reactor pressure vessel surveillance program in which metallurgical test specimens of the reactor pressure vessel are placed in capsules at or near the reactor pressure vessel inner wall. They are then irradiated in the temperature and neutron flux/spectrum environment that is as similar as possible to the pressure vessel itself. The research underway has an end product of a series of 15 standards, procedures, and so forth, for calculating the fluence and spectrum, for measuring the results of neutron detectors, and for evaluating and correlating the radiation damage sustained in the surveillance capsules. The project is to be completed by FY 1985. Major test facilities used or established for this purpose include:

1. Pool critical assembly (PCA) at ORNL where transport theory calculations of a simulated pressure vessel wall/thermal shield assembly were made in FY 1980 followed by extensive measurements to validate the calculations, project to be completed in FY 1981;

2. Pool side facility (PSF) at ORNL, which contains irradiation capsules filled with metallurgical test specimens of vessel steels. In FY 1982, these will provide postirradiation results for embrittlement and flux fluence spectra for surveillance, vessel surface, 1/4t, 1/2t, and 3/4t of the vessel wall, so that an appropriate benchmark can be established and proposed for licensing use in FY 1983;

3. Surveillance dosimetry measurement facility (SDMF) at ORNL, which should be available in the FY 1982-1983 time period, where surveillance capsule detectors for a given reactor will be irradiated for counting and evaluating by the testing laboratory to certify that the detectors can make accurate dosimetry measurements; and

4. Power reactor dosimetry measurement facility (PRDMF), which will be available in about FY 1983. An operating power reactor surveillance facility will be used as a reference benchmark to obtain dosimetry and embrittlement data over one or two cycles of operating time so that other laboratories can certify that they can properly account for the geometry, core power, buckling, perturbation, and distance extrapolation, and so forth, actually encountered in a power reactor.

The corresponding standards arising from this research will be written by FY 1983 so that ASTM acceptance and NRC implementation can be accomplished by FY 1985.

4.7.5.3 Crack Growth Rate

A crack or flaw in a reactor component will grow longer and deeper with time. This growth will be influenced by the cyclic stresses imposed during normal operation (or accidents), by the temperature and water chemistry, and by the materials and material condition. The rate of crack growth must be known accurately for the environmental parameters specific to nuclear plants. A flaw, discovered through inservice inspection, may grow to a critical size during a given time into the future. If growth to a critical size is likely before the next scheduled shutdown (based on the data now being developed in research programs), the flaw must be removed. The research program is aimed at developing data on crack-growth rate--in fracture-mechanics terms--for pressure vessel steels and welds and for piping steels and welds. Both carbon and stainless steels are included in the test matrix. Testing is typically performed in autoclaves capable of loading a series of metallurgical test specimens at a temperature of 550°F and up to 2200-psi pressure in water of carefully controlled chemistry to simulate that of either PWRs or BWRs under normal operating conditions.

4.7.5.4 Environmental Pipe Cracking

Studies on the environmental effects that cause pipe cracking in LWRs are to be started in FY 1981 (following a comprehensive review of this subject in FY 1980). The objective is to develop an independent capability for prediction, detection, and control of pipe cracking in LWR systems. The program is directed both at pipe cracking in existing plants and the development of recommendations for plants under construction and for future plants. The impetus for this program comes from the continuing occurrences of cracks in both PWR and BWR piping systems. The program includes:

1. Development of the means to objectively and quantitatively evaluate crack and leak detection systems;

2. Definition of the role of stress, metallurgical variables, and environment on pipe-cracking susceptibility, including the influence of plant operations on these variables; and

3. Examination of practical limits for these variables to effectively control pipe cracking in LWR systems.

An important task in this effort will be to determine the role of BWR and PWR water chemistry on the susceptibility for pipe cracking in these systems.* This will be accomplished by determination of the probable local chemical environment in particularly susceptible locations, as a function of the bulk water chemistry, the transport and deposition of corrosion products, the local geometry, flow and heat transport conditions, and the local metallurgy.

*This work will be translated, in FY 1984, into initial validation on recommended changes in proposed fixes for BWR and PWR pipe-cracking incidents in existing plants. Recommendations will be developed in FY 1985 for avoidance of pipe-crack incidence in newer plants either under construction or in design stages.

Based on tests to evaluate the range of constituents in the bulk coolant chemistry that can be tolerated without initiating intergranular stress-corrosion cracking (IGSCC), operating limits on the bulk coolant chemistry will be proposed in FY 1986 for BWRs and PWRs.

4.7.5.5 Steam Generator Tube Degradation

Research to determine the burst and collapse strength of steam generator tubing began in FY 1976, and studies to develop a method for predicting stress-corrosion cracking in steam generator tubing began in FY 1977. A retired PWR steam generator acquired in FY 1980 was shipped to a research laboratory for more extensive studies of the causative factors of steam generator tube degradation. The original work in this area had goals of developing (1) validated models for the prediction of margins to failure under burst and collapse pressures and (2) leak rates for steam generator tubing found in service to be degraded. The first stage of the test plan called for tubing to be tested with machined flaws in the form of slots and areas simulating wastage. The next stage was validation of these tests with chemically induced corrosion cracks. Final validation with tubes that have been degraded in service in an operating steam generator is to follow. A parallel study is developing quantitative data and models for predicting the service life of Inconel 600 steam generator tubing under stress-corrosion-cracking conditions for operating and abnormal service conditions. Areas to be studied include factors affecting crack initiation and propagation such as temperature, stress, strain and strain rate, environments, metallurgical structure, and processing. This research is to be completed in FY 1982.

A test program on a retired steam generator will begin in FY 1982. The program will include:

1. Validation of tube-integrity predictive models with tubing removed from the generator (FY 1982);

2. Validation of stress-corrosion-cracking predictive models (FY 1982);

3. Development and optimization of tube-plugging criteria (FY 1983);

4. Determination of causative factors in steam-generator-corrosion degradation and optimization of water chemistry (FY 1984);

5. Evaluation of proposed chemical cleaning processes and procedures with respect to near-term integrity and long-term effects on corrosion, degradation, and safety (FY 1985); and

6. Nondestructive examination (NDE) development and validation, including optimization of inservice inspection procedures sampling plan and inspection period (FY 1983).

Final results in the form of licensing recommendations for control of certain generator operating and inspection conditions are expected in FY 1986.

4.7.5.6 Turbine Failures

The objectives of this program to be initiated in 1983 are, first, to characterize the stress-corrosion-cracking and corrosion-fatigue problems encountered in nuclear plant turbine rotors and blade keyways and, subsequently, to evaluate proposed corrective actions. Cracking in turbines is different from pipe cracking because of the different materials and environments encountered. Cracking usually occurs in turbine materials where caustic nitrates and sulfates are likely to concentrate (from the steam environment). The research program will include:

1. Conducting a thorough search of the literature on turbine failures, particularly in nuclear systems, and compiling and correlating the failure modes with respect to plant operating conditions and steam chemistry (FY 1983-1984);

2. Testing steam turbine steels in concentrated caustic environments to establish the cracking morphology and failure mode (through FY 1984);

3. Testing steam turbine steels under chemistry and conditions that are representative of both normal and faulted chemical environments realistically based on the operating history of nuclear steam turbines (through FY 1985); and

4. Evaluating proposed fixes for control of turbine cracking (FY 1985).

The data will be be accumulated through FY 1987; it will form an experimental basis for licensing evaluations of proposed corrective actions for turbines.

4.7.5.7 LWR Water Chemistry

Criteria for optimization and control of secondary side water chemistry cannot be easily derived from available operating reactor data; appropriate research is therefore being planned. It is scheduled to start in FY 1983 and will include the necessary parametric studies using test sections under realistic, controlled conditions. The test sections will incorporate both sound and degraded tubes with actual support plates, tube sheets, crevices, geometrical discontinuities, cracks, and sludge piles. The mockup will be operated under simulated thermal/hydraulic and chemical service conditions. Bulk water chemistries and chemistry changes, pH, and conductivity will be studied as a function of time for controlled startups, shutdowns, chemical intrusions, and corrective actions (additions) to the intrusions. These studies will be accompanied by pH and conductivity measurements and by corrosion-potential measurements of various components in the generator. Because some of the most important parameters in the corrosion degradation and cracking of LWR materials are the chemistry and electropotential present in crevices, crack tips, and sludge piles, measurements in addition to the bulk measurements will be made in these areas. Such measurements will include chemistry, corrosion potential, pH, and conductivity. The research information developed will be used for corrosion testing of materials under the appropriate conditions revealed from the experiments and for optimization of water chemistry control criteria for operating reactor steam generators to minimize corrosion degradation. The results will thus provide an experimental data basis for evaluating recommended PWR secondary water chemistry limits and control criteria by FY 1986.

4.7.5.8 Pipe Repair Welding

A clear need exists for research on the effects of welding on piping, especially stainless steel piping for both fabrication welds and repair welds. This need arises because the continuing integrity of piping is strongly affected by the fabrication and welding history of the pipe. Thus, beginning in FY 1982, research is planned to establish the parameters and the acceptable ranges of those parameters that affect the integrity of piping. These include the time and temperature of welding, travel speed, the number of cycles of applied heat, the weld interpass temperature, the metallurgical condition of the pipe as received from the mill, and the overall material chemistry and processing history. Other factors that will significantly affect the welding and repair of carbon steel pipe as well as of stainless steel pipe include residual stresses, postweld stress relief, and weld-joint geometry. Recommendations for developing a regulatory guide on repair welding of stainless steel will be prepared in FY 1983, while similar recommendations will be made for carbon steels by FY 1985. Validating the procedures for stainless steel repair welding is planned for FY 1985.

4.7.5.9 Physical-Metallurgical Degradation

A number of physical-metallurgical processes continue to exist in reactor materials; these processes can contribute to the material degradation. Because such processes usually take place over a long period of time (or are otherwise nearly invisible), they are often easy to neglect. These processes include the long-term aging and toughness degradation of cast austenitic stainless steel and hydrogen embrittlement. The toughness degradation of cast stainless steel is symptomatic of the aging embrittlement that can occur in materials operating in a metastable state.

A nuclear component must be capable of operating successfully and without degradation for some 40 years, and there is little or no information available on the stability of current structural materials over that long a time period.

Hence, starting in FY 1982, the toughness of stainless steel removed from long-term service will be evaluated; information from such studies should be available by about FY 1984. Long-term aging studies of appropriate materials will also begin, but data from these studies cannot be expected until about FY 1987 simply because appropriately long heat treatment periods are required to permit accurate assessment of long-term changes. While the effects of hydrogen embrittlement can be detrimental to a structure, typically they are not a problem for nuclear systems because of the construction materials and the operating temperatures. In some cases, however, problems can arise that must be considered. In the accident at TMI-2, a large hydrogen overpressure had built up in the primary system and remained for a significant time period at a variety of temperatures. As a result, the time-temperature-material circumstance of TMI-2 will be studied in FY 1983 to determine if a potential hydrogen embrittlement situation exists now, may exist for future operations, or existed at the time of the accident.

4.7.5.10 Results

1. Irradiation Effects and Annealing

FY 1982	Establish upper-shelf toughness and transition shift characteristics for current low-shelf steels and welds; establish a valid basis for toughness limits in Appendices G and H to 10 CFR Part 50; recommend time-temperature cyclic and systems parameters for inplace annealing of vessels.
FY 1984	Establish toughness data and annealing response information on specific plants, especially the older plants being reviewed under the Systematic Evaluation Program, for licensing decisions.
FY 1987	Establish and validate a fracture-toughness data base for newer pressure vessel steels coming into service.

2. Surveillance Dosimetry

FY 1982 Establish a PSF benchmark for predicting metallurgical changes and dosimetry in vessels based on surveillance data.

FY 1983 Complete writing and essential validation for 15 standards, guides, and practices in surveillance dosimetry; adapt standards, guides, and practices for regulatory use and implementation.

FY 1985 Gain acceptance as ASTM standards, guides, and practices.

3. Crack Growth Rate

FY 1982 Develop crack growth rates for irradiated steels and welds at high R ratios.

FY 1983 Complete evaluation of long hold times on crack propagation in pressure vessels and heat affected zones.

FY 1984 Establish the existence and magnitude of threshold levels for crack growth.

FY 1986 Complete revision of S-N curves for ferritic steels in ASME Section III to reflect environmental effects.

FY 1987 Incorporate data on austenitic stainless steels.

4. Environmental Pipe Cracking

FY 1982 Complete facilities for larger-scale pipe crack validation tests.

FY 1984 Provide initial validation or recommended changes in proposed fixes for BWR and PWR pipe-cracking incidents in existing plants.

FY 1985 Develop recommendations for avoidance of pipe crack incidence in newer plants under construction or in design.

FY 1986 Propose practical limits for environmental variables to control pipe cracking in LWR systems.

5. Steam Generator Tube Degradation

FY 1982 Validate quantitative models for predicting service life in Inconel 600 steam generator tubing; validate tube integrity prediction against burst and collapse from tubing removed from service on the basis of nondestructive examination data.

FY 1983 Propose and validate tube plugging criteria.

FY 1984 Initially validate causes in steam generator corrosion degradation and optimization of water chemistry.

FY 1985 Experimentally evaluate proposed chemical cleaning processes.

FY 1986 Develop recommendations for licensing control of steam generator operation to control corrosion and degradation.

6. Turbine Failures

FY 1984 Initially test turbine steels in normal and faulted chemistry environments of nuclear steam systems to establish crack formation and crack propagation rates.

FY 1985 Evaluate proposed fixes for control of turbine cracking.

FY 1987 Establish experimental basis for licensing evaluation of causes and proposed corrective actions to eliminate turbine failures.

7. LWR Water Chemistry

FY 1984 Assemble test sections and start water chemistry variables study.

FY 1986 Provide experimental basis for evaluating PWR secondary water chemistry limits and criteria.

8. Pipe Repair Welding

FY 1982 Initiate a parametric study of time, temperature, cycles, heat input, material condition, and joint geometry for weld-repair procedures in austenitic stainless steel in LWR piping.

FY 1983 Prepare recommendations for regulatory guides on critical parameters in stainless steel repair welding.

FY 1985 Prepare recommendations for updating regulatory guides on repair welding in carbon steels, and validate procedures for repairs in stainless steels.

9. Physical-Metallurgical Degradation

FY 1982 Initiate study of toughness loss in cast austenitic stainless steels removed from long-term service, and begin long-term aging control tests of cast stainless steel specimens.

FY 1983 Complete evaluation of potential embrittlement from hydrogen overpressures generated in primary systems through various accident and faulted conditions.

FY 1984 Develop initial findings of toughness loss in cast stainless steels from components removed from service.

FY 1987 Provide initial validation results from long-term thermally aged control test specimens; evaluate and validate, as possible, recommendations from vendors on prevention or control of such toughness degradation.

4.8 Nondestructive Examination

Nondestructive examination of nuclear reactor components is required during fabrication, before service, and at regularly scheduled shutdowns for periodic inservice inspection so that any flaws that are initially present or that develop in service can be detected, their significance is evaluated and unacceptable (potentially unsafe) flaws are removed to ensure or maintain safety. Possible generic-type defects that may be present in the remainder of the system must also be identified so that timely corrective action may be taken. Included among the data needed for safety evaluations using fracture mechanics and code procedures is accurate information on flaw size, shape, orientation, and location within the component. To accommodate this need, information is needed on details of tests and examination procedures that, if implemented, can ensure the accuracy of the flaw data recorded for evaluation. The research is thus aimed at developing and improving nondestructive examination procedures for fast and accurate detection, characterization, and evaluation of flaws in nuclear plant components so that licensing decisions can be made on the safety implications of those flaws with respect to future service of the plant.

4.8.1 Regulatory Objective

The regulatory objective of the nondestructive examination program is to provide the NRC staff a catalog of nondestructive examination techniques with which NRC has assurance that a given flaw can be detected and its severity can be accurately characterized.

4.8.2 Technical Capabilities Required

Nondestructive examination for inservice inspection of nuclear power reactors is both a requirement and a key factor in establishing the safety and integrity of operating reactors. Inservice inspection is depended on for identifying inservice-induced degradation and possible generic degradation problems. Only when one knows of the absence of flaws,--or the exact size, shape, and location of detected flaws--can one make intelligent and safe decisions about operating conditions that will influence the continuing integrity of primary system components.

Experience has shown that the current nondestructive examination techniques do not always give the results required with respect to reliability and accuracy of flaw detection and evaluation. Several NRC memorandum discuss and describe the needs for improvements in nondestructive examination. A March 10, 1977 NRC memorandum regarding the ultrasonic inspection of cast austenitic stainless steels, K. V. Seyfrit (IE) stated: "There has always been some concern about the capability for ultrasonic volumetric examination of the welds associated with such components, due to the high attenuation characteristics of the base metal....We are concerned, however, that the preservice and inservice examinations currently being performed may be generating a sense of false confidence, since we believe that, if any defects were to develop in the future, presently available volumetric examination techniques would not disclose them on a timely basis." The program on improved ultrasonic inspection for flaw evaluation discussed below addresses these needs.

An NRC memorandum dated August 20, 1977 from V. Potapovs (IE), on a "Branch Program Plan for Metallurgy and Materials Research Branch" states that there is an urgent need to define the flaw sizes that can be reliably detected by using current techniques and to identify improvements that can be made. This subject has evolved into Generic Issue A-14, "Flaw Detection and Characterization." The research program described below on reliability of inservice ultrasonic inspection addresses these needs. The same memorandum expresses the need for the development of acoustic emission techniques for hydrotest monitoring and continuous online monitoring. These needs also are addressed below.

Other specific research requests and endorsements for the research program on flaw detection and evaluation are:

1. Memorandum from S. Levine (RES) to V. Stello (IE), subject: Confirmatory Research on Continuous Internal Friction Flaw Monitoring of LWR Components in Service, November 6, 1979; and

2. Memorandum from H. D. Thornburg (IE) to L. S. Tong (RES), subject: Request for Study on Optimization of Eddy Current Systems for Inspection of Steam Generator Tubing in PWR Plants, November 18, 1977.

4.8.3 Status of Capabilities

Experience from inservice inspection of reactors and from round-robin tests has shown that the ultrasonic test techniques being used often have poor reliability for flaw detection and are inadequate for flaw sizing. Therefore, research began in late FY 1978 to quantify the reliability of current ultrasonic inservice inspection techniques, to identify the inspection parameters that contribute to unreliability, and to establish the changes required in these parameters to obtain, in the short term, considerable improvement in the reliability of current inservice inspection. Further, the reliability of new, improved techniques must be quantified to ensure that new systems give inspection reliabilities consistent with a high assurance of safety in inspected components.

A research project was started in FY 1975 aimed at dramatically improving the resolution capabilities of ultrasonic testing to obtain the accurate flaw characterization needed for safety evaluations. An entirely new technology has been developed for ultrasonic testing that uses the phase and amplitude information of signals and search unit position to form a highly accurate image of flaws. This signal-processing technique has been named the synthetic aperture focusing technique for ultrasonic testing (SAFT-UT). The SAFT-UT technique has been demonstrated in the laboratory to be effective in accurately characterizing flaws. The sensitivity of results to a specific operator, a specific calibration test, or a specific transducer and instrumentation--as is the case with current ultrasonic testing techniques--has been reduced or eliminated in the SAFT-UT and the other associated processing techniques. A field-implementable system based on SAFT-UT has been constructed and needs to be evaluated in the laboratory for its capability to accurately characterize flaws.

As an integral part of the primary system pressure boundary, inservice inspection of steam generator tubing is required; while most other primary system components are inspected by ultrasonic testing, eddy-current testing is normally used for steam generator tube inspection. The ASME-code eddy current inspection techniques that are presently used are fast, but they can produce unreliable inspection results because of the many independent variables that affect the signals. For example, it is extremely difficult to detect flaws in a dented tube region surrounded by corrosion products and the steam generator support plate. A research project was started in FY 1978 to develop improved eddy-current inspection techniques for steam generator tube in service inspection.

In this project, a three-frequency instrument was constructed and laboratory-evaluated with the capability for either separating and measuring or discriminating against variations in each of the following parameters: (1) tube diameter, including denting at the supports; (2) probe wobble; (3) the presence of supports around the tube; (4) tube wall thickness; (5) location (radial and axial) of defects in the tube wall; and (6) the size of wall defects. Preliminary field tests of the equipment will be conducted in FY 1981 to inspect for flaws in the tubesheet region of several operating steam generators, and the instrument will be upgraded for additional field tests.

Techniques are being developed for the continuous online monitoring of flaws and defects. Detection of flaws--and the monitoring of flaw growth during service--is one of the most powerful means of preventing unexpected failure of primary system components during service. Ongoing and planned research projects will develop acoustic emission technology for inservice leak surveillance and online detection and evaluation of crack growth, as well as internal-friction monitoring technology for predicting and detecting crack initiation.

A research project was started in FY 1977 to determine the feasibility of and develop the technology for (1) the online detection of flaws in nuclear reactors and (2) the evaluation of their significance on a continuous basis by measuring and analyzing the acoustic emission information. A laboratory baseline of data has been developed using a variety of specimens and testing conditions to characterize the acoustic emission signals and crack growth. In FY 1981, work will be completed on m[illegible]acks growing in pressure vessels tested in fatigue and burst under s[illegible]ated reactor operating conditions of varying stresses, temperatures, noise sources, thermal shocks, and so forth. These tests use a monitoring system that incorporates the flaw-severity and crack-growth models and the noise-discrimination models developed from laboratory investigations.

Using acoustic emission methods to detect coolant leaks during nuclear reactor operation offers potential improvement in sensitivity, response time, location accuracy, leak-source characterization, and leak quantification over present methods. Leak surveillance by acoustic emission methods detect the noise generated by leak flow to the atmosphere from the high-pressure, high-temperature coolant system. This noise is propagated through the material; it is detectable remotely by piezoelectric sensors mounted on the system.

Recently a technique was developed that has the potential for identifying the initiation of cracking during online monitoring, thus allowing an early warning system that precedes microscopic cracking. The technique uses the phenomenon of internal friction in materials and identifies changes in the specific damping capacity as a function of the integrity of the material being tested. A research program was started in FY 1979 to develop the technique, evaluation criteria, and instrumentation for the online detection of intergranular stress-corrosion crack (IGSCC) initiation. Four laboratory tests of IGSCC in 4-inch, Schedule 80, Type 304 stainless steel pipe were conducted in FY 1980. High-temperature pressurized water at 550°F and 1150 psig were used while the pipe was stressed and cycled. Internal friction monitoring of these pipe tests showed that the technique can detect the IGSCC phenomenon at an early stage. The technique identified a major change in material integrity at 15 to 25 percent of the total life of the pipe test carried to leakage. Instrumentation has also been installed in operating reactor pipe systems to determine the effect and applicability of the technique under reactor background noise and environmental conditions. Research in FY 1980-1981 will show when the technique actually senses changes in material integrity (believed to be at the crack-initiation stages). It will also quantify the effects of varying reactor operating conditions on internal friction-damping changes.

4.8.4 Research Program Objectives

The objectives of this program are:

1. To quantify the reliability of current inservice inspection techniques for primary system components and to establish the required parameter modification in these techniques to obtain improvements in the reliability of inservice inspections;

2. To develop, evaluate, and validate advanced techniques for flaw detection and evaluation during inservice inspection of primary system components and steam generator tubes; and

3. To develop and validate new techniques for the continuous online monitoring of crack initiation and growth in reactors during operation and for leak detection.

4.8.5 Reactor Program Plan

4.8.5.1 Inservice Inspection Techniques

1. Reliability of Inservice Ultrasonic Inspection: Specific objectives of this project are:

    a. Determine the reliability of ultrasonic inservice inspection performed on commercial LWR primary piping systems;

    b. Using fracture mecnanics analysis, determine the impact of nondestructive examination unreliability on system safety, and determine the level of inspection reliability required to ensure a suitably low-failure probability;

c. Evaluate the degree of reliability improvement that could be achieved with improved and advanced nondestructive examination techniques; and

d. Based on material, service, and nondestructive examination uncertainties, formulate recommended revisions to ASME Code Section XI and to NRC requirements needed to ensure suitably low-failure probabilities.

The project consists of four phases. Phase I was completed in FY 1980. The accomplishments of this phase include evaluation of the impact of inspection variables on inservice inspection reliability, estimates of current inservice inspection reliability, and fracture-mechanics analysis to determine the range of required nondestructive examination sensitivities. A RIL was prepared summarizing these results and making recommendations for changes in code requirements and inspection parameters that can be immediately implemented to improve markedly inservice ispector sensitivity and reliability. Phase II will include (a) round-robin inspections to determine inservice inspection reliability of primary systems and (b) fracture-mechanics calculations to determine the impact of inspection reliability and the required levels of inservice inspection reliability. Major results from Phase II will be available by the end of FY 1981, but round-robin testing will continue through FY 1983. Phase III, to be completed in FY 1983, will concentrate on the evaluation of improved conventional and advanced techniques for the purpose of establishing the level of reliability improvement that can be achieved. Phase IV will establish a unified set of inspection requirements based on service requirements. Recommendations for code modifications will be made throughout the program as data become available. Although this project concentrates at its beginning on piping inspections, the results will also be useful for pressure vessel inspection. Appropriate tests will be conducted so that in the final recommendations (to be available in FY 1985-1986) code improvements and the unified inspection requirements will relate to ultrasonic-based inservice inspection of the entire primary system.

2. Improved Ultrasonic Inspection for Flaw Evaluation and Detection: The SAFT-UT system described in Section 4.8.3 will be validated in FY 1981-1982

through performing flaw characterizations during actual inservice inspections of reactors and comparing these results to conventional inspection results. Code-acceptable SAFT-UT for flaw characterization should be available by FY 1982. Further developments in SAFT-UT will concentrate on technique optimization, flaw-display improvements, increasing processing speed to obtain near-real-time images, and automation of flaw characterization. These improvements will allow the application of SAFT-UT to reliable flaw detection and accurate characterization. An improved SAFT-UT system will be field validated for both detection and evaluation of flaws for inservice inspection by FY 1984 and should be accepted by the Code in FY 1985 or 1986.

3. Eddy-Current Inspection for Steam Generator Tubes: The three-frequency eddy-current instrument previously developed will be retested in the modified version in the laboratory on a range of samples, and the equipment will be field tested on full-length steam generator tubing. Based on these results, necessary modifications will be made for improved automatic data processing and storage before a final field validation of the improved multifrequency eddy-current inspection techniques. This validation will be done during an actual steam generator inservice inspection to be conducted in FY 1982. Final efforts in this development will be the presentation of data and of a code case aimed at gaining Code acceptance of the new techniques in FY 1983.

4.8.5.2 Continuous Monitoring Techniques

1. Online Acoustic Emission Surveillance of Operating Reactors: In FY 1982, based on the results of the pressure vessel tests described in Section 4.8.3, the acoustic emission monitoring system and evaluation models will be upgraded and an engineering-prototype monitoring system will be built. Meanwhile, in FY 1981, an operating reactor will have been instrumented with sensing arrays ready for monitoring. Monitoring of the reactor using the engineering prototype monitor will start in FY 1982 and continue for 2 years. This will permit data compilation under actual conditions so that valuable experience is gained and the system and technology can

be validated. In FY 1984, a final upgraded system will be validated online and a code case will be presented.

Technology for the acoustic emission monitoring of hydrotests required for inservice inspection must be developed because access conditions or other considerations make other inspection methods unfeasible. Based on the vessel testing conducted through FY 1981 and additional required data, acoustic emission testing and analysis procedures will be formulated in FY 1982. Evaluation and acceptance criteria for the hydrotest monitoring of reactor components to determine the presence of cracks and their possible growth in operating reactors will also be formulated at that time. These procedures and criteria will be validated using large-scale tests and actual component hydrotest monitoring during FY 1983-1984.

2. Leak Surveillance by Acoustic Emission Monitoring: A project to develop and improve the technology for application of acoustic emission to leak surveillance will begin in FY 1982. The research will (a) define and improve the sensitivity of the method for detecting leaks from actual cracks under reactor operating conditions, (b) evaluate and develop sensing systems for effective leak detection compatible with the reactor environment, (c) develop improved methods for accurate leak location, (d) develop methods for discriminating the acoustic emission from different leak types (crack versus pump and seal and so forth), and (e) develop methods and correlations for relating the acoustic emission to leak quantity and to the through-wall crack size. This technology will be developed and validated and actions taken for Code acceptance by the end of FY 1985.

3. Online Evaluation/Prediction of Crack Initiation: Based on previous results, a stand-alone quantified monitor will be built in FY 1982 and validation will be started by actual reactor testing for the automated early warning identification and location of IGSCC in stainless steel piping systems. By FY 1984, the development and validation for Code acceptance of the technique will be completed. Research through FY 1986 will extend and validate techniques, instrumentation, and methodology

based on internal-friction damping for the early warning and identification of fatigue damage, fatigue crack initiation, and radiation damage.

4.8.5.3 Results

The research program plan is presented below in terms of milestones by fiscal years.

1. Inservice Inspection Techniques

FY 1982 Establish the reliability of current inservice inspection for piping systems and recommend the Code changes to bring about the required improvements; provide experimental evidence and impact to licensing for development of technical positions for a guide on inservice inspection of stainless steel piping; validate flaw characterization by SAFT-UT by using a field-implementable system in actual inservice inspection of reactors; validate the improved multifrequency eddy-current inspection and evaluation system and methods over the entire range of flaws and conditions present in operating steam generators through actual inservice inspection of steam generators; formulate acoustic emission testing procedures and flaw evaluation procedure for hydrotest monitoring.

FY 1983 Validate reliability improvement of recommended changes to conventional Code techniques, and evaluate the level of improvement possible using advanced techniques; gain Code acceptance of SAFT-UT for flaw characterization; develop methodology for an engineering prototype of an improved SAFT-UT system for concurrent near-real-time flaw detection, characterization, and display; gain code acceptance of the improved multifrequency eddy-current flaw inspection system and evaluation criteria for steam generator tube inservice inspection as developed by ORNL.

FY 1984 Flaw-detection probability and fracture mechanics models for reactor component integrity assessment developed and evaluated;

field-validate improved SAFT-UT near-real-time flaw detection, characterization, and display using a field-implementable system on actual reactor inservice inspection; validate and gain Code/regulatory acceptance of acoustic emission testing and evaluation procedures and acceptance criteria for inservice hydrotests.

FY 1985 Recommend a unified set of inspection requirements using nondestructive examination flaw-detection reliability and sensitivity, component material properties, and service conditions to ensure a suitably low-failure probability; gain Code acceptance of the automated near-real-time SAFT-UT flaw detection, characterization, and display techniques for inservice inspection; evaluate nondestructive examination techniques for preservice and fabrication inspection.

FY 1986 Gain Code acceptance of the unified set of inspection requirements for inservice inspection of primary system components.

FY 1987 Improve and/or adapt new techniques for preservice and fabrication inspection; evaluate new techniques needed for inservice inspection.

2. Continuous Monitoring Techniques

FY 1982 Optimize an engineering-prototype acoustic emission monitoring system based on previous laboratory validation, and start reactor online monitoring; evaluate acoustic emission signal information for leak surveillance; validate the internal-friction-damping nondestructive examination technique for detection of intergranular stress corrosion crack initiation by online monitoring of reactor piping systems.

FY 1983 Optimize acoustic emission monitoring and flaw-evaluation methodology and equipment based on reactor monitoring experience; develop criteria and methods for acoustic emission leak monitoring for detection, location, source discrimination, and leak quantification; gain Code acceptance of internal friction damping-nondestructive examination technique for detection of intergranular stress corrosion crack initiation.

FY 1984 Recommend final criteria and methodolgy for online monitoring and flaw evaluation using acoustic emission; develop feasibility for internal friction damping-nondestructive examination application to fatigue damage and crack initiation.

FY 1985 Gain Code acceptance for online acoustic emission monitoring and flaw evaluation; validate acoustic emission leak-monitoring and evaluation methodology by testing on reactor; gain Code acceptance of online acoustic emission leak monitoring and evaluation; provide criteria and instrumentation for internal friction damping-nondestructive examination application to online detection and evaluation of fatigue damage.

FY 1986 Validate the internal friction damping-nondestructive examination technique by testing on reactor for detection and evaluation of fatigue damage.

FY 1987 Provide feasibility for detection and evaluation of radiation damage using internal friction damping-nondestructive examination techniques; establish feasibility of new techniques for online monitoring.

5. SEVERE ACCIDENT PHENOMENOLOGY AND MITIGATION RESEARCH

This chapter contains three sections related to LWRs--fuel melt behavior, fission product release and transport, and severe accident mitigation--and an additional section on fast reactors. Depending on the availability of funds, the program for fast reactors will be as responsive as possible to the needs of the Commission.

There is no data base for containment design under core-melt conditions. Current estimates use bounding arguments in such codes as MARCH and CORRAL. While such estimates may be useful for risk assessment, the Zion-Indian Point study (NUREG/CR-1409, -1410, and -1411) showed them to be inadequate for concept evaluation when applied to an existing plant. Moreover, they are certainly not sufficient for engineering detailed design and evaluation, whether the plant is old or new. Thus, the first of the three sections related to LWRs, on fuel melt behavior, is aimed at establishing a data base for regulatory use in evaluating detailed designs.

The fission-product release and transport section deals with the detailed processes that produce a radiological source term. As experience at TMI-2 showed, and as had been expected through the years, current regulatory guides provide a substantial margin of conservatism regarding source-term evaluation. In the context for which these guides were prepared, such conservatism is appropriate and reasonable. For safety assessment, especially when there are competing effects, a knowledge of the margins of conservatism and of the best estimate is needed.

The source term is central to the Siting Rule, establishes a key environmental load for engineered safety features, and sets the value of potential consequences from core-melt accidents, thereby establishing the measure of mitigation considered in the Degraded Core Cooling Rule. Finally, the source term determines which fission products have to be dealt with in emergency planning design of engineering safety features and auxiliary systems. This work clearly has high priority.

The goal of the severe-accident mitigation section is to develop the basis for assessing schemes to mitigate the effects of severe accidents on containment. While it is too early to know to what extent such schemes will be used in the licensing process, the possibilities range from doing nothing, to inerting containments to protect against hydrogen, to installing very large, strong containments capable of external cooling (as do some European plants).

5.1 Fuel Melt Behavior

The consideration of accidents involving significant amounts of molten fuel requires knowledge of how the fuel interacts: with water to predict if missiles or high steam pressures will be formed; with the pressure vessel to predict whether failure will be catastrophic or gradual; and, with concrete to determine how fission products, noncondensible gases, and aerosols are formed in the containment building. The Zion-Indian Point study revealed deficiencies in our ability to make these determinations and predictions. This program will remedy those deficiencies on a time scale consonant with rulemaking.

5.1.1 Regulatory Objective

The regulatory objective is to be able to predict with confidence the resulting effects of various degrees of accidents involving severe fuel damage or core melt and to have sufficient information for developing a degraded core cooling rule.

5.1.2 Technical Capabilities Required

The technical capabilities required are derived from the following questions about threats to containment:

1. Can fuel debris from a severe accident be cooled within the reactor vessel? What conditions must be satisfied to ensure coolability?

2. What conditions must be exceeded and what information does the operator need to conclude that a severe accident sequence is headed inexorably toward complete core meltdown and that attention should be focused on maintaining containment integrity?

3. Can a melted core breach the lower core-support structure and the reactor pressure vessel?

4. Can a steam explosion in the reactor vessel generate missiles to breach containment, or can rapid steam generation from early fuel-water interaction in the reactor cavity overload containment?

5. Can hot core debris be cooled in the reactor cavity?

6. Can a hot core melt through the basemat?

7. Can the containment be overloaded by slow pressurization?

A research program focused on these questions is responsive to the NRC Action Plan (Sections II.B.5 and II.B.8) and certain parts of the user request NRR-80-4.

5.1.3 Status of Capabilities

Analyses of TMI accident sequences performed for both the Kemeny and Rogovin Commissions showed that if restoration of cooling had been delayed a little longer than it was, substantial core melting would have occurred. The NRC Action Plan (developed since the TMI accident) calls for research on phenomena associated with fuel melting (II.B.5) and for rulemaking for degraded core cooling (II.B.8). Information from this research will provide key parts of the technical bases for regulatory decisions.

The state of technology for core melt accident assessment in WASH-1400 is embodied in the MARCH-CORRAL code. In WASH-1400 sequences, after core melt was reached, vessel melt-through and the subsequent dropping of melt into the reactor cavity was assumed. Evaluation of core-melt accidents for the Zion and Indian Point nuclear power stations revealed major phenomenological uncertainties in the MARCH-CORRAL simplified models. These uncertainties require improved data and analytical development for fuel-melt behavior for (1) accident management assessment, (2) consequence assessment for dominant accident sequences, and (3) design evaluations of possible mitigation features.

5.1.4 Research Program Objectives

The objective of the fuel-melt behavior research is to develop phenomenological data, models, and verified systems codes for analyzing severe-accident sequences that could proceed through one or more of the following phases of fuel melt: (1) fuel-melt relocation, debris formation, and debris cooling or remelt following severe core damage; (2) fuel-melt attack on lower core support, possible steam explosions, and vessel melt-through; (3) fuel-melt interaction in the reactor cavity and debris cooling or remelt; and (4) containment loading.

5.1.5 Research Program Plan

Fuel melt behavior research has been divided into five groups: (1) transition to fuel debris, (2) fuel debris behavior, (3) melt interactions, (4) steam explosions, and (5) severe accident analysis. Important information for this program is also available from close coordination with European programs at BMFT/FRG/KfK and EURATOM/ISPRA.

5.1.5.1 Transition to Fuel Debris

The program on transition to fuel debris includes the following objectives:

1. Single rod inpile experiments on fuel relocation phenomena for model development;

2. Multirod inpile experiments for fuel relocation model verification;

3. Experiments on molten fuel streaming, blockage formation, and core-support structure attack, both laboratory and inpile, for model development; and

4. Out-of-pile experiments on fuel debris formation and characterization.

This work has an important interface with Section 2.7, "Core Damage Beyond LOCA."

The objective of this program is to provide a data base and verified analytical models of the characteristics of fuel during its transition from a melting, severely damaged core to fuel debris in the lower vessel regions. The data base and the models are to be used in assessing the coolability of the debris under both invessel conditions. If the case is to be made that the fuel debris can be satisfactorily cooled and contained in the reactor vessel for a certain range of accident conditions, a much stronger research base will be needed than that required for simply assessing the exvessel threat of core debris to the containment.

1. Single-Rod Inpile Experiments on Fuel Relocation Phenomena: This work consists of a series of single rod, inpile separate effects experiments, supporting laboratory experiments, and supporting model development on key phenomena on the processes of fuel relocation following fuel failure in severe accidents. These are laboratory-type separate effects experiments on individual key phenomena to gain data for model development; they are quite different from the more common, semiprototypic inpile tests. They involve extensive diagnostics with emphasis on visualization techniques, fuel position diagnostics, ultrasonic thermometry, and the like; the test reactor simply provides a fission heating source. Data on fuel behavior and relocation are provided throughout the accident transient (rather than depending only on data from the single, end-of-transient time point available from postirradiation examination). (The experiments are an extension of similar highly successful liquid metal fast breeder reactor (LMFBR) safety experiments in the Annular Core Research Reactor (ACRR). The LMFBR experiments provided much information on key fuel disruption phenomena that could not be previously acquired.) Important phenomena to be investigated in these new experiments include streaming and freezing of the fuel, clad, and clad-oxide eutectic and the postfailure fragmentation

and relocation of the fuel. The experiments will start in FY 1982 and will be completed in FY 1984.

2. Multirod Inpile Verification Experiments on Fuel Relocation Phenomena: The primary objective of these multirod fuel relocation experiments in semiprototypic reactor environments is to verify the models developed in the single rod separate effects phenomenological experiments and associated laboratory experiments. These multirod tests will also provide a data base for model development and verification of streaming and blockage phenomena. PBF is a possible facility for such multirod tests.

3. Experiments on Molten Fuel Streaming, Blockage Formation, Lower Vessel Region Attack, Both Laboratory and Inpile: The purpose of this work is to acquire a data base for the development of analytical models of streaming and blockage formation by molten fuel and by the fuel, clad, and clad-oxide eutectic. The work includes (1) laboratory experiments with nonreactor materials on the basic processes involved and (2) fission heated experiments in the ACRR with reactor materials that can also provide continuous heating of the fuel in simple, well-characterized geometries. Similar LMFBR experiments, both laboratory and inpile, are currently underway; they will be modified and extended to cover LWR-specific conditions in FY 1982-FY 1985. Experiments on fuel-melt attack of the lower core support structure and vessel head will be performed in the ACRR and in the large melt facility (LMF) starting in FY 1982 and ending in FY 1986.

4. Out-of-Pile Experiments on Fuel Debris Formation: The purpose of this work is to develop a data base and, where needed, analytical models of fuel debris formation. The emphasis is on the completeness of the fragmentation of molten fuel debris upon freezing and on the characterization of the debris, in particular in regard to particle-size distribution. Invessel debris characterization is needed not only for beds of particulate fuel debris on the lower head or core-support structure, but also for

debris suspended in or below the original core region. Debris characterization under exvessel conditions is also needed. Data are needed for the full range of molten materials that occur in severe LWR accidents, including the various oxidation states and eutectics. Field-scale fragmentation tests in water--similar to sodium tests--will be performed, with thermite melts to cover as wide a range of the relevant melt-oxidation states and eutectic conditions as possible; however, the tests will still be quite limited.

Starting in FY 1983 and ending FY 1985, fragmentation experiments will be performed with the large melt facility (LMF) and fully instrumented test series (FITS) facility (see Section 5.1.5.4). Larger-scale experiments can be performed over a much larger range of fuel, clad, and clad-oxide states and eutectics than is possible with the self-heated thermite melts.

5.1.5.2 Fuel-Debris Behavior

The objectives of this task are (1) to determine the dryout coolability limits and postdryout characteristics of LWR fuel debris for use in assessing the threat to the reactor vessel and basemat and (2) to support the work on melt interactions with core debris retention materials.

The program objectives include:

1. Invessel debris coolability experiments in the ACRR and model development for fuel-debris dryout and extended dryout. Experiments using sodium (to simulate low pressure water) are underway now and will continue through 1984; experiments using high pressure water will start in FY 1982 and end in FY 1986.

2. Exvessel debris coolability experiments in the ACRR and model development for fuel-debris dryout and extended dryout. Experiments using atmospheric pressure water will be conducted in FY 1982-1984, with studies of gas effects on coolability to be conducted in FY 1983-1984.

3. Inpile, dry capsule, molten fuel interaction experiments with refractory core retention materials (e.g., MgO, graphite) are underway now and will end in FY 1983.

5.1.5.3 Melt Interactions

The objectives of this task are to (1) provide a technological base for the thermal, mechanical, and chemical interactions of fuel melt with steel, concrete, refractory, or sacrificial retention materials, including the presence of water, introduced before, during, or after the initiation of the interaction and (2) develop a verified computer model for predicting the interaction behavior of molten core materials with structural material for use in assessing basemat melt-through or retention device effectiveness and containment load sources through incorporation in systems codes.

The scope of this task includes:

1. Small-scale, scoping, and phenomenological experiments of thermal, mechanical, and chemical interactions of fuel melt with steel, concrete, refractory, and sacrificial retention materials;

2. Large-scale scoping or model-verification tests with core materials and simulants in the LMF;

3. Development of computer models to predict the interaction of molten core materials with concrete, refractory materials, sacrificial materials, and coolant;

4. Quantification of gaseous or aerosol source terms generated during the interaction;

5. Modeling decay heat redistribution with gaseous or aerosol sweepout of fission products;

6. Evaluation of the presence of water or other coolant introduced before or after the start of melt interaction; and

7. Maintaining a resident scientist at the KfK Laboratory to facilitate the exchange of core-melt research information.

Most of these activities are underway; all are expected to be completed by FY 1985.

5.1.5.4 Steam Explosions

The objective of this program is to improve understanding of the physical phenomena associated with the rapid, and sometimes violent, interactions that occur when the molten core materials contact water. The findings will provide a technical basis for predicting (1) under what conditions an explosive interaction will occur and (2) the probability that such an interaction will damage either the primary reactor vessel or the containment. The research will investigate the modes of contact between the molten core materials and water, details of triggering and propagating the explosion, and structural response. The steam explosion research is carried out at Sandia in small-scale single droplet experiments of melt into water to measure the triggering energy required as a function of system pressure, degree of subcooling of the water and melt composition. Larger scale fully instrumented test series (FITS) experiments in a 2 meter diameter by 3 meter high vessel evaluate the thermal-to-mechanical conversion ratio, propagation velocity and structural response. Structural response of the reactor vessel and the containment response to potential missiles is then calculated by structural response codes. Steam explosion research is expected to be completed in FY 1983.

5.1.5.5 Severe Accident Analysis

The objective of this program is to develop an integrated set of analytical methods to predict reactor behavior under a broad spectrum of severe accident conditions. This analysis will depend on computer codes and data from

Chapter 2, "LOCA and Transient Research," for the analysis of the early phases of the accident. In particular, the primary system thermal/hydraulics analysis will depend on systems codes; fuel damage analysis will rely on codes and data developed in Section 2.7, "Core Damage Beyond LOCA." This material will provide input to the subsequent analysis of the progression of the accident from severe clad damage to loss of core geometry and eventually to the evaluation of the threat to containment from all the modes of failure (see Section 5.1.2).

When the source and characteristics of the threat to containment have been defined, this project will be coordinated with the structural response analysis methods developed under programs described in Chapter 4, "Plant Operational Safety."

Over the last few years, various codes and models have been developed that can be used in the analysis of severe accidents. In every case, these codes have limitations precluding a rigorous analysis of the entire course of the accident. In particular, these limitations make it difficult to assess the impact of intervention in the course of the accident and the effectiveness of systems designed to mitigate consequences.

Recent analyses of core-melt accidents have depended heavily on the MARCH and CORRAL codes. They evolved from Battelle Columbus Laboratory (BCL) support for WASH-1400 and are systems codes that treat the beginning of the primary system transient through the development of the radiological source term. MARCH was used in the recent Zion/Indian Point Study; its use is described in Chapter 6 of the Report of the Zion/Indian Point Study, Volume I (NUREG/CR-1410), prepared by Sandia. MARCH uses simple models to derive an approximate sequence of events during meltdown. Given the uncertainties in the front-end probabilities, it may provide an adequate measure of the consequences for risk assessment, but the modeling needs for mitigation studies are more stringent. Proper evaluation of mitigation systems will require better understanding and modeling of key phenomena, as well as better integrated codes, to provide consistent treatment throughout the accident.

An improved severe accident analysis code will be developed in close cooperation with the experimental program because many of the defects of existing methods can be corrected only with improved models based on expanded data.

The SIMMER code will be considerably modified to analyze extended core motion and relocation as well as the interactions of core materials with water (steam) explosions. Again, the validity of the results will require better models of the phenomena which, in turn, depend on an expanded data base. The code development using existing and near-term experiments is scheduled to be completed in 2 to 3 years. Further improvements reflecting the state of the art will be incorporated as better understanding is achieved. Another important aspect of understanding severe accidents deals with analyzing the interactions of the core material with structural components outside the primary vessel. This analysis will depend on the CONTAIN code now under development, as well as on detailed special effects codes such as CORCON, USINT and the various aerosol-release and transport codes. A trial version of CONTAIN will be available in FY 1981. Improvements reflecting the progress of the experimental program will be incorporated as appropriate.

The development of these various methods requires careful integration in order to optimize calculational efficiency and to ensure that the model development and the experimental programs are closely coordinated.

5.1.5.6 Results

FY 1983 Begin multirod verification experiments in PBF on fuel relocation phenomena; begin fuel fragmentation experiments in the large melt facility (LMF) and the fully instrumented test series (FITS) facility; complete inpile dry capsule experiments in ACRR on the interactions of molten fuel with potential core retention materials; complete steam explosion experiments in the FITS test facility; document and release CONTAIN (LWR version) computer code.

FY 1984 Complete single rod inpile separate effects experiments on fuel relocation phenomena in ACRR; issue report on ACRR dry capsule experiments and models on the interactions of molten fuel with potential core retention materials; issue report on the FITS steam explosion experiments discussed in Section 5.1.5.4, models, and on the steam explosion threat to containment integrity; complete multidimensional core melt and relocation analysis package.

FY 1985 Issue report on single rod separate effects ACRR experiments and models on fuel relocation phenomena; complete inpile separate-effects experiments in ACRR on LWR molten fuel streaming, blockage formation, and lower vessel region attack; complete fuel fragmentation experiments in the LMF and FITS facilities; complete large scale melt interaction tests in the LMF; issue report on the gaseous or aerosol source term generated by core melt interactions; document and release integrated severe-accident analysis package.

FY 1986 Complete multirod verification experiments in PBF and fuel relocation phenomena; issue report on ACRR separate effects experiments and models and LWR molten fuel streaming, blockage formation, and lower vessel region attack; complete LMF experiments on fuel melt attack on the lower core support structure; issue report on fuel fragmentation experiments and models; complete inpile debris coolability experiments in ACRR for LWR accident conditions; issue report on the results of small-scale phenomenological and large-scale verification experiments and models on core melt interactions.

FY 1987 Issue report on PBF multirod verification experiments and models on fuel relocation phenomena; issue report on LMF experiments and models on fuel melt attack on the lower core support structure; issue report on ACRR debris coolability experiments and models for LWR accident conditions.

5.2 Fission Product Release and Transport

Fission product release and transport data and models predict the radiological source term for accident consequence assessment and site evaluation, and the radiological and thermal loads imposed by released fission products on engineered safety features and other safety related plant equipment and instrumentation. Such predictions are needed for rulemaking related to emergency preparedness, siting, engineered safety features, and degraded core cooling. While a significant amount is known about release and transport under LOCA conditions, and about the behavior of aerosols in containment, there are few data points for high temperature releases, and for transport in two phase flows. Moreover, recent studies emphasize the importance of addressing the chemical and thermal behavior of fission product compounds. This program addresses these issues. Since the emergency preparedness rule has been issued, and the siting rule will be published in the near term, this work is urgently needed. Results compatible with needs are expected in about 2 years.

5.2.1 Regulatory Objective

The regulatory objective is to develop a data base for fission product migration from the normal fuel configuration to the environment. This will include degraded cores, severely damaged fuel and fuel melt conditions as fission product sources. The data base will include information on plateout, chemistry, and other radioactivity-reduction mechanisms.

5.2.2 Technical Capabilities Required

Studies consistently indicate that the uncertainties associated with fission product release and transport-behavior assumptions (and models) are among the largest contributors to uncertainties in the risk to the public from severe accidents at nuclear power plants. This indication is not surprising for two reasons: (1) offsite consequences are directly affected by the magnitude, timing, and makeup of the source term released from containment and (2) there are large uncertainties regarding the actual potential source term. The ultimate objective of this program is to improve the quality of predictions of the potential time-dependent fission-product source term for species released from containment under accident conditions. Not all fission products behave the same; fallout, solution, and resuspension differ by element as well as under differing environmental conditions. Priorities for investigation are based on the radiological importance of a given species and the relative uncertainty associated with the current knowledge of the release and transport behavior of this species. To accomplish this objective, it will be necessary to understand and model:

1. Fission product vapor and aerosol release from the fuel during initial heatup, melting, and interaction with plant structures (including the reactor cavity concrete basemat);

2. Fission product and aerosol transport and deposition behavior within the reactor coolant system and containment;

3. Effectiveness of engineered safety and mitigation features in reducing the potential source term released from containment; and

4. Physiochemical form of radiologically significant fission products in the fuel, in the fuel/cladding gap region, and during transport within the primary system and containment in both gaseous and aqueous transport media.

Key data are needed in the near term (FY 1982-1983) to support rulemaking activities.

5.2.3 Status of Capabilities

NRC regulatory guides (i.e., 1.3, 1.4, 1.7, and 1.89) on siting and accident analysis prescribe source terms for evaluating the consequences of various design basis accidents. These source terms (which implement the approach defined by Technical Information Document 14844, "Calculation of Distance Factors for Power and Test Reactor Sites") represent accident conditions that far exceed the potential source term from a successfully controlled (Appendix K design basis) LOCA. These terms were later codified in Regulatory Guides 1.3 and 1.4. Thus, the source term assumptions in Regulatory Guides 1.3 and 1.4 specify that 100 percent and 25 percent of the equilibrium core inventories of the noble gases and iodines, respectively, are immediately released into the containment atmosphere following a LOCA. The offsite radiation doses for potential sites, evaluated with conservative meteorology, would often exceed the allowable dose limits specified in Part 100 when the regulatory guide assumptions are used without provision of compensating engineering safety features; therefore, appropriate features for mitigating the offsite dose were investigated as provided by Part 100. This led to the introduction of safety features such as containment sprays (with iodine-removal agents), containment air filters, secondary containment systems, negative-pressure containments, and so forth.

Because the source term released into containment was set by regulatory practice for siting evaluation (and was known to be conservative for design basis accidents), less subsequent research was done by either the industry or the government to investigate actual fission product release from the fuel or fission-product behavior within the primary system under accident conditions. Of the fission-product research accomplished, much was directed at assessing the effectiveness of the engineered safety features for the removal of iodine in various containment atmospheres.

Renewed interest developed and an intensive program to evaluate realistic source terms under LWR accident conditions was conducted during the reactor safety study (RSS). Because of the scarcity of applicable experimental data, large uncertainties were associated with the fission product release and transport assumptions included in the study. In fact, in certain areas so little information was available that only bounding assumptions could be made (for example, fission product attenuation within the primary coolant system).

Beginning about 1975, several studies were initiated by the NRC to investigate the release of fission products from irradiated LWR fuel rods under accident conditions and to develop models for fission-product transport behavior within the reactor coolant system. These programs have provided (1) realistic release models for fission product escape from fuel rods under LOCA conditions in the temperature range 500°C to 1200°C (tests have been conducted up to 1600°C) and (2) a mechanistic model (TRAP-MELT) for fission-product behavior within an LWR primary system under accident conditions up to and including fuel meltdown.

During the reactor safety study, a relatively simple computer code (CORRAL) was developed to model the behavior of fission products in the containment atmosphere. The original CORRAL code had relatively detailed models for spray washout of iodine-vapor species; however, the spray removal of particulate fission products and surface deposition of aerosols and vapor species were crudely modeled. Since the reactor safety study, work has been continued to improve these models but at a relatively low level of effort. As the experimental base is obtained, commensurate improvements in the models will be made.

In the area of aerosol behavior within containment structures, significant progress has been made under the fast reactor program which is broadly applicable to all aerosol studies. Experimental programs to characterize the generation, agglomeration, and surface deposition rates of Na, UO_2, and Na/UO_2 aerosols have been conducted. The results of these experimental programs have formed the basis for a number of mechanistic aerosol behavior codes, including HAARM, ZONE, QUICK, and MULTI-AEROS.

5.2.4 Research Program Objectives

The objective of this program is to develop models and correlations for (1) fission product release from LWR fuel over a wide range of accidents involving substantial damage, (2) fission product transport and behavior within the primary system and containment, and (3) fission product release to the environment under severe LWR accident conditions, including fuel melting. This completes the scope of a regulatory data base that spans the range from operational transients (see Section 2.6) to severe core damage (see Section 2.7) to severe accidents. The data and models developed in this program are used to assess the performance of accident mitigation features as well as to support the Siting and Emergency Response rules.

5.2.5 Research Program Plan

The following three sections describe the specific research projects for meeting the needs identified above.

5.2.5.1 Fission Product Release

Planned research programs to investigate and quantify the release of fission products and aerosols from the fuel include:

1. An experimental program to measure the release of fission products from commercially irradiated LWR fuel rod segments in a steam environment under elevated-temperature (1000°C - 2600°C) accident conditions (starting in FY 1981 and ending in FY 1987). The program uses fuel of varying ages including high burnup material. First results at high temperature (2000°C) are scheduled for the end of FY 1981 with higher-temperature tests to follow.

2. A program to investigate the release of aerosols from molten pools of core materials interacting with reactor cavity concrete and with core retention materials (ending in FY 1984);

3. Examination and analysis of samples of the TMI-2 core (FY 1982-1984); the schedule will depend on the TMI-2 cleanup schedule; and

4. Development and improvement of mechanistic models to predict the release of fission products from high-temperature fuel (GRASS-SST) and during interactions of the damaged and molten fuel with residual coolant and plant structures (now underway and ending in FY 1984).

5.2.5.2 Fission Product Transport

Research programs are planned in the areas of fission-product vapor and aerosol transport and deposition, including:

1. Continued improvement of the TRAP-MELT code to model fission product behavior within the primary coolant system under severe accident conditions, and the extension of the mechanistic, multicompartment TRAP-MELT code models to the prediction of containment fission product behavior (ongoing, to be completed in FY 1983). Results from this program will be factored into the CONTAIN code.

2. An experimental and analytical program (thermodynamic calculation) to provide model development data for the TRAP-MELT code in the areas of elevated temperature fission product vapor pressures, surface deposition rates and mechanisms, and fission product chemical reactions with steam, prototypical surface materials, and other fission products (ongoing, to be completed in FY 1982 but may be extended).

3. Continuation of experimental and analytical programs to develop models for containment aerosol fission product behavior applicable to both liquid metal and light water reactor accident conditions. The aerosol models will be incorporated into the TRAP-MELT, MARCH/CORRAL, and the CONTAIN code to predict overall fission product transport behavior. (To be completed in FY 1983.)

4. Modification and operation of a facility to test and verify the primary system and containment building fission product and aerosol transport codes (in FY 1982-1983).

5. Examination and description of potential leak paths for fission products via fluid systems that are connected to the primary system or containment sump bypassing the containment (FY 1982-1984).

5.2.5.3 Fission Product Control

Programs are planned to investigate and quantify the effectiveness of various engineered safety and mitigation features in reducing the potential fission product escape from containment. Included are programs to (1) investigate and quantify the radioiodine retention performance of impregnated activated charcoal absorbers under accident conditions, (2) evaluate alternative methods for filtration of radioiodines, (3) investigate techniques for the removal and processing of large volumes of radioactive noble gases, and (4) experimentally investigate and model the spray removal of aerosols under severe accident conditions. These programs are planned for completion by FY 1984.

5.2.5.4 Results

FY 1983 Complete experiments on the release of aerosols during the interaction of core melt with concrete and potential core retention materials; issue report on the improved, mechanistic, multicompartment TRAP-MELT code for the prediction of containment fission product behavior; complete experiments and analytical models for containment aerosols and fission product behavior under both LWR and LMFBR accident conditions for use in TRAP-MELT, MARCH/CORRAL, and the CONTAIN codes, and issue report; begin program of verification experiments for primary system and containment fission product and aerosol transport codes; issue report on experiments on the effectiveness of impregnated charcoal absorbers for radioiodine retention under severe accident conditions.

FY 1984 Issue report on the results of experiments and models on aerosol release during the interaction of core melt and potential core retention materials; issue report on the examination and analysis of TMI-2 core debris; issue report on model for the release of fission products from high-temperature fuel (GRASS-SST).

FY 1985 Issue report on the analysis of potential fission product leak paths via fluid systems that bypass the containment.

FY 1986 Complete experiments on the release of fission products from LWR fuel in a steam environment; issue report on experiments and analysis on the effectiveness of spray removal for reducing large airborne concentrations of aerosols.

FY 1987 Issue report on the results of experiments and models on fission product release in a steam environment; complete program of verification experiments for primary system and containment fission product and aerosol transport codes.

5.3 Severe Accident Mitigation

The results from the programs on core melt and fission product release and transport will be used in this program to investigate new features or strategies that might be developed to enhance the safety of the public. Each proposed new feature will be analyzed in detail to determine if it is beneficial and practical to require nuclear power plants to include it.

5.3.1 Regulatory Objective

The objective of this research is to enhance the public safety and reduce risk by analyzing, testing, and evaluating concepts for new operating strategies, devices, and systems designed to preserve the integrity of containment against failure. The use of such systems depends on a cost-benefit analysis within the context of safety goals.

5 3.2 Technical Capabilities Required

The analysis of design features to mitigate severe accidents consists of two parts: analysis of function, reliability, and system interaction to assess possible reduction in risk, and analysis of cost. Both analyses require significant engineering foundations for success, especially for backfitting where engineering constraints are tightest. As shown in the Zion-Indian Point Study, backfitting a conceptually simple device such as a filtered vent may increase risk if improperly done.

5.3.3 Status of Capabilities

The TMI accident demonstrated clearly that severe accidents beyond the design basis must be considered in the designing, siting, and licensing of nuclear reactors. The importance of maintaining containment integrity as a line of defense was forcefully made apparent. In WASH-1400, the following containment failure modes were identified: steam explosions, hydrogen combustion/detonation, overpressure from a loss-of-cooling capability, basemat melt-through, and isolation failure (i.e., penetration failure, valve left open, etc.) or interfacing-system LOCA. (The latter is a special case of isolation failure resulting from check valve failure between the primary system and the low pressure injection system outside of the containment.) The work in this program is intended to analyze, test, and evaluate methods to improve containment by improving cooling, by allowing for controlled filtering, and by venting to relieve overpressure before catastrophic failure or by increasing the pressure that can be tolerated without failure. Core retention devices will be investigated to prevent basemat penetration or to reduce the contribution of the core melt/concrete interaction to containment overpressurization. Finally, methods and devices to reduce the hydrogen threat to containment in a severe accident will be studied. Existing analytical models and experimental data base are probably adequate to estimate the risk reduction potential of various mitigation features, but they are not adequate to support design, particularly in backfitting existing plants. The analytical models and experimental data base are probably adequate to estimate the risk reduction potential of various mitigation features, but they are not adequate to support design particularly in backfitting existing plants.

5.3.4 Research Program Objectives

The objectives are to develop information through experiments, study, and analysis on improved containment systems, core retention systems, hydrogen control systems, and other mitigation methods to permit the making of knowledgeable regulatory decisions.

5.3.5 Research Program Plan

Some of the work in this program will begin in FY 1981; however, much of it will not begin until FY 1982 or later. Some of the effort will involve generalizing special studies started in FY 1980 to support the Zion/Indian Point study; other effort will apply research projects in the LMFBR safety research program to the study of LWR problems.

5.3.5.1 Improved Containment

This research includes evaluating the feasibility and utility of (1) filtered, vented containment designs, (2) improved systems for cooling containment, and (3) design improvements for building stronger containments. Design requirements and cost estimates will be prepared for each improved containment concept. Value/impact assessments will be completed, with special attention to the engineering problems associated with backfitting existing plants. Systems-interaction studies and separate effects tests will be included. Tests on vent filter system components to investigate the feasibility of design features will continue in an accident aerosols source term test facility at Oak Ridge through FY 1983. Work on improved containment cooling will concentrate on passive cooling concepts. Stronger containment designs will include new designs as well as improvements suitable for backfitting. Substantial results will be available in FY 1983, with the bulk of the program completed in the following year.

5.3.5.2 Core Retention Systems

This research includes evaluating the feasibility and utility of both sacrificial and refractory core retention devices designed with either active or passive cooling.

Melt interactions with representative retention materials are being studied, and refractory materials are receiving priority attention. With the availability of the large fuel-melt facility and the laboratory-scale material interactions test program now underway at Sandia (see Section 5.1.5.3), considerable technology will be available during FY 1981-1982. The objective of the research program on retention-materials interactions, to be completed in FY 1984, is to provide the data base necessary to allow extensive conceptual design and evaluation of possible core retention systems and to provide a sound basis for detailed engineering design in late FY 1983 or early FY 1984. The core retention systems research will include investigations of the ability of the system to interdict the liquid pathway for release of radioactive materials.

5.3.5.3 Hydrogen Control

This program includes evaluating the feasibility and utility of devices to control hydrogen in a severe accident either by preventing combustion or by controlling it. Controlling hydrogen by combustion using ignition sources has a high potential for effectiveness in many accident scenarios. It requires research to develop technically sound ignition means, and it places a high premium on effective, local, hydrogen monitoring. Research in this aspect of hydrogen control will concentrate on evaluating the effectiveness of distributed ignition sources, in developing and testing devices to provide a reliable and accurate assessment of hydrogen concentrations throughout the plant and evaluating the impact of controlled ignition on safety-related plant equipment.

Concepts that prevent combustion include inerting with nitrogen or using halon as a suppressant. Inerting with nitrogen is largely a passive system, but backfitting ice-condenser plants with Nc may be very expensive and difficult and could reduce overall safety by limiting access for plant maintenence and inspection. The principal task is to evaluate the total system impact from this proposed solution. The use of halon suppressant is potentially effective in preventing hydrogen combustion; however, its technical feasibility has yet to be demonstrated. Problems to be investigated include personnel hazards, consequences of decomposition products on the plant and equipment, and reliability. This effort will peak in FY 1983, with final documentation and program completion in the following 2 years.

The research program on controlling hydrogen is closely coupled to research on core melt and fuel damage so as to coordinate the development of the hydrogen threat to containment with its mitigation. Coordinated investigations will be carried out to establish the effects on equipment of controlled combustion, including radiant energy emission from localized hydrogen fireballs and deflagration pressure waves.

5.3.5.4 Results

FY 1983 Complete tests in NSPP on the effectiveness of vent filter system components; complete engineering design evaluation of the feasibility and cost effectiveness of backfitting core-melt mitigation systems for ice-condenser PWR and Mark III BWR.

FY 1984 Issue report on the NSPP tests on the effectiveness of vent filter system components; issue report on vent filter system conceptual design studies for both new plants and for backfitting old plants; issue report on study of improved containment cooling systems; issue report on conceptual design studies and evaluation of potential core debris retention systems.

FY 1985 Issue report on conceptual design studies and evaluation of stronger containment; issue report on the results of tests and analysis of the feasibility and effectiveness of various hydrogen combustion control devices.

FY 1986 Complete development of technical bases for core melt mitigation design criteria and standards.

FY 1987 Complete coordinated investigations into environmental signature for equipment qualification under severe-accident conditions.

5.4 Fast Reactor Safety

Research on this program would be intended to prepare the NRC to license and regulate breeder reactors. This type of reactor is sufficiently different from the light water reactors now in use that it will require some adaptation of regulations covering construction and operation. The responsible way to approach the problem is to gather data to understand safety issues and to develop methods of analysis in advance so they can be used when licensing and regulating decisions must be made.

licensing is renewed.

5.4.1 Regulatory Objective

The regulatory objective will be established soon after the Administration's program on fast breeder reactor development is announced. The current program considers only generic safety issues until a national program on fast reactors is resolved. The detailed choices should become fixed during the first year of the program, at which time technical objectives will be more sharply focused.

5.4.2 Technical Capabilities Required

The technical capabilities required for this program have been developed over the past years during safety reviews of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor (CRBR). Many of the safety issues to be resolved during the development of a commercial breeder industry have been identified during these reviews. It can be expected that the resumption of licensing review of the CRBR or another demonstration plant will sharpen the focus of this program by developing new research needs and improving the definition of those already identified. Correspondingly higher levels of research expenditure will be required under these circumstances as outlined below.

5.4.3 Discussion of Possible Program Options

A number of options are available for LMFBR safety research in support of licensing. They are listed as follows:

1. The current program could be continued. It is funded at about $15 million per year, and has remained about level for the last few years. Continuation of the current program would mean concentrating on the following general areas of research:

   a. Analysis - A complementary collection of accident delineation studies and computer codes to analyze the response of the reactor to transients under offnormal and accident conditions.

   b. Accident Energetics - An experimental program to furnish a data base and models for assessing the threat to the primary vessel from core disruptive accidents.

   c. System Integrity - An experimental program to furnish data and models to assess the response of the plant to thermal and mechanical loads during accidents.

   d. Aerosol Release and Transport - An experimental program to assess the radiological consequences of severe accidents and the usefulness of mitigation systems.

   A larger, more concentrated program would be developed when NRC is presented with a requirement to take action.

2. If the licensing of the CRBR should be resumed, research would be accelerated with funding at about $24 million in FY 1982. The program would depend on the recently completed LMFBR Accident Delineation Study (NUREG/CR-1507) to ensure structure and completeness. Some key areas from the study include:

   a. Alternative decay heat removal system reliability,
   b. Natural convective cooling effectiveness,
   c. The standing of reactivity insertion accidents in relative risk assessments,

d. The standing of local fault propagation accidents in relative risk assessments,

e. Containment integrity in core melt accidents with threat from basemat melt-through, overpressurization, missile penetration, and the like.

f. Energy associated with core-disruptive accidents (homogeneous vs heterogeneous cores).

3. DOE is now completing a Conceptual Design Study (CDS) of a 1000 MWe LMFBR. The study is in the final review and comment stage in the Department of Energy Headquarters and is due to the Congress by the end of March. If a decision should be made to proceed with the CDS reactor, a lower yearly expenditure ($15 - 18 million) for research will be needed because more time will be available to complete the necessary work. DOE would require several years to bring the CDS reactor design to the same level of detail now available for the CRBR. This time would be used by RES to develop an NRC program of audit and independent confirmation of the safety design basis of this new reactor.

4. In the event that no licensing action is anticipated in this decade, a minimum level of $8 million would be required to keep a cadre of scientists available to build upon for future need, at least as long as DOE continues to fund an LMFBR development program of several hundred million dollars per year. At the $8 million level, the research program would concentrate on analysis and small-scale experiments. Major experiments requiring the use of a test reactor or other expensive experimental facilities would be deferred.

5. A final option is the one currently budgeted; it calls for the FBR safety research program to end in FY 1981.

As soon as the Administration approach is defined, a detailed LMFBR research plan will be developed.

6. SITING AND ENVIRONMENTAL RESEARCH

Two parts of this program (Seismology-Geology and Meteorology-Hydrology) are described in this chapter. The remaining parts appear in Chapter 15 of this plan.

6.1 Seismology and Geology

Research on earthquakes and other geologic hazards is conducted to help ensure the safety of nuclear facility sites and to provide information to help resolve problems of interpretation and implementation of the Seismic and Geologic Siting Criteria for Nuclear Power Plant Sites. Knowledge of earthquake hazards and methods for their quantification have improved markedly in the past decade. Considerable judgment is still required, however, in assessing these hazards because of the limitations in data and state of the art of seismic and geologic analyses. The large part of the nation to be considered and the number of potential earthquake zones also contribute to the size of the problem. The kinds of data required include the distribution of present seismic energy release, the kinds and distribution of other deformational processes, physical criteria for discriminating between active and inactive faults, and the location and extent of subsurface as well as surface features of the earth's crust that influence the occurrence of earthquakes.

Explanatory models of earth crustal processes are being offered by the earth science community to explain earthquake occurrence in different regions of the eastern United States. While some or all of these models may be applicable in some areas, the impact of their direct application to seismic hazard analysis appears likely to continue the trend in upward revision of perceived danger from earthquakes in the relatively quiet eastern United States. Development of the explanatory and unifying hypotheses that are needed for realistic assessment of seismic hazards requires extensive surface and subsurface geological and geophysical information like that being gathered in these seismology and geology research programs.

6.1.1 Regulatory Objective

General Design Criterion 2 of Appendix A to 10 CFR Part 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes and tornadoes without loss of capability to perform their functions. Appendix A to 10 CFR Part 100 sets forth criteria pertaining to site investigations to assess the effects of earthquakes and other geologic phenomena to meet the requirements of General Design Criterion 2. Appendix A to 10 CFR Part 100 also sets forth considerations to guide the Commission in its evaluation of (1) the suitability of a proposed site, (2) the suitability of plant design bases established in consideration of site characteristics, and (3) reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public.

Appendix A (10 CFR Part 100) criteria, procedures, and methods are directed toward the following major objectives:

1. Estimating the severity of ground shaking at a site due to potential earthquakes for use in nuclear power plant design;

2. Assessing the potential for ground rupture that could affect plant structures as a result of fault movement;

3. Evaluating the effect on the site of phenomena associated with earthquakes such as seismically generated sea waves and ground failure; and

4. Assessing the potential for other geologic hazards at a site such as landslides and subsidence.

6.1.2 Technical Capabilities Required

The required capabilities for NRC site review that are addressed by this research are primarily those associated with the estimation of severity of ground shaking and the assessment of potential for earthquake ground rupture at sites. The ability to assess the potential for soil liquefaction resulting from ground shaking is also required.

Specific review capabilities include site and regional seismicity, relationship of earthquake occurrence to geologic and tectonic features of the region, and determination of earthquake-generation potential of geologic structures and tectonic provinces in the region. Also required are assessments of the characteristics of seismic wave transmission in the free field at the site, determination of the level and properties of the vibratory ground motion in the free field at the site in terms of time histories of ground acceleration, velocity and displacement, local site response characteristics, site response sensitivity studies, site soil investigations, site-matched records, and site seismic input criteria. Probably the most important capability required is that of assessing the ground motion that would result from the maximum potential earthquakes associated with each tectonic province or geologic structure within the region, considering site-dependent effects.

6.1.3 Status of Capabilities

A reevaluation of the Appendix A (10 CFR Part 100) criteria was begun in 1976 by SD and NRR as a result of difficulties encountered by the NRC licensing staff in applying the criteria. This reevaluation is continuing, and rulemaking to consider revising the regulation is planned. Difficulties encountered have included lack of clarity in the instructions for accomplishing criteria requirements, ambiguity, and restrictions on the ability to use advances in science and engineering pertinent to the assessment of site acceptability.

Many of the difficulties in implementing the seismic and geologic siting criteria are also the bases for research needs. Among the pertinent issues identified in the most recent assessment (SECY 79-300) of Appendix A shortcomings are:

1. Inadequate definition of tectonic provinces and lack of a tectonic province map acceptable for generic application;

2. Criteria for correlation of earthquakes with geologic structures;

3. Definition of capable faults and active but "noncapable" faults in areas where maximum seismicity is defined by association with a tectonic province, as in the eastern United States and parts of the West·

4. Specification of the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) in terms of maximum size, recurrence, free-field acceleration at site, etc;

5. Seismological aspects of specifying earthquake time histories for design;

6. Seismological aspects of specifying duration of strong earthquake ground motion for design; and

7. Use of probabilistic methodology in determining seismic input for both the OBE and SSE.

All projects ongoing at the beginning of FY 1981 had received endorsement from one or more of the Directors of the user offices. Many of the endorsements were requested for projects initiated under previous procedures for regulatory research or for work proposed in anticipation of regulatory needs. Current and recent research requests and endorsements are shown below.

Number or Identification:	Title
RR-NRR-76-8 (Same as SAFER-76-5) (June 7, 1976 and June 16, 1977: NRR endorsed SSMRP 2-28-79)	Quantification of safety margins inherent in seismic design
SD--Minogue Memo, 1-15-79	Request for seismic research
Request for Endorsement RES 1-25-79 (NRR endorsed 3-2-79) (SD endorsed 4-6-79)	Earthquake research programs (with attached project description 34 FINs)

Number or Identification:	Title
R-NRR-79-7 (3-23-79) (NRR endorsed 9-27-79)	Earthquake-induced foundation settlement
Request for Endorsement RES 5-18-79	Arizona geological survey proposal state catalog
RR-NRR-79-14 (completed, equipment on standby for future use)	Acquisition and deployment of a portable seismic array in New England (Maine earthquake)
Request for endorsement RES 10-4-79 (NRR endorsed 10-18-79)	Photo reproduction and cataloguing of historic seismograms
Request for endorsement RES 11-16-79 (NRR endorsed 1-11-80)	Proposal to study lateral variations in attenuation of ground motion in the eastern United States based on L_g-type surface wave
RR-NRR-79-29 (12-15-79) (program ongoing)	Increased seismic monitoring in the region around the Ramapo Fault
Request for endorsement RES 3-25-80 (NRR endorsed 4-17-80)	Recurrence interval for earthquakes at nuclear power plants
Request for endorsement RES 12-11-80 (NRR endorsed 1-27-81)	Investigation of epicentral region around the 7-27-80 Sharpsburg, Kentucky earthquake

6.1.4 Research Program Objectives

The overall objective of this research is to provide (1) improved bases for licensing decisions and development of standards applicable to the safety of nuclear facility sites and (2) authoritative information to assist in revising the siting criteria.

Integrated regional and topical programs in geology and seismology are conducted to provide the data and information for future synthesis into topical units that will resolve, it is hoped, the individual regulatory issues. An objective of this approach is to provide maximum real data for verification of underlying hypotheses, models, and observed empirical relationships used in resolving of the issues.

Because of the regional nature of the geologic characteristics that influence occurrence of earthquakes and because of the regional specialization of many individual geoscience investigators, programs of a similar nature are organized by region in different parts of the country. Because the majority of nuclear power plants are located east of the Rocky Mountains and because geologic and seismologic characteristics of this part the country require application of the tectonic-province approach to specifying maximum earthquake size, most of the research is conducted in the eastern United States.

In the western United States, significant problems exist with determining "capability" of faults and the maximum earthquake size and recurrence relationships associated with individual faults. However, determining earthquake hazard in the western United States is comparatively more straightforward than in the eastern United States. Also, extensive earthquake research and hazard investigations are conducted in the western United States by the U.S. Geological Survey and the National Science Foundation. For these reasons, only a small part of the NRC geoscience research effort is conducted in the west.

6.1.5 Research Program Plan

Delineation and characterization of tectonic provinces, as provided for in the Seismic and Geologic Siting Criteria (Appendix A to 10 CFR Part 100), have been the principal needs addressed by this research. Other regulatory applications of the research are to provide authoritative information to assist in revising the siting criteria, in evaluating the seismic hazard at particular sites during construction permit and operating license reviews, and in developing or documenting methodology for preparing regulatory guides and standard review plans. Siting concepts that have been proposed to improve safety or decrease risk are evaluated to further the NRC mission of protecting public health and safety; examples include underground siting, and site isolation to reduce amplitudes of seismic motions at building foundations.

6.1.5.1 Regional Programs

The programs consist of projects to monitor and interpret seismicity, to collect and compile surface and subsurface geological information, and to conduct tectonic analyses. Approximately 200 seismographs are deployed in six regional networks in the eastern and northwestern United States. High-resolution vertical seismic reflection profiling (Vibroseis) is being used in selected areas to obtain detailed information on structure and geometry of the earth's crust to depths at which most earthquakes occur (1 to > 25 km). Selected for Vibroseis profiling have been areas of occurrence of large earthquakes (Charleston and New Madrid), areas of recent occurrence of instrumentally well-located small earthquakes (central Virginia), and areas critical to tectonic interpretation (Wabash Valley-Cottage Grove fault). Results from the extensive Consortium for Continental Reflection Profiling (COCORP) Vibroseis profiles have demonstrated the many values of this technique, which was developed for petroleum exploration. This also may substantially impact tectonic theories and interpretations of the causes of intraplate earthquakes. Because of the very high costs, Vibroseis exploration will be limited, and profiles must be selected to provide the greatest amount of significant information relating earthquakes to geologic structures.

The six program areas that comprise the regional programs are:

1. Northeastern U.S. region,
2. Southeastern U.S. region,
3. New Madrid, Missouri region,
4. Anna, Ohio region,
5. Nemaha Uplift region, and
6. Northwestern U.S region.

Each of the eastern regions has been through a 5-year cycle of geologic studies and seismic monitoring, but the effective initial year of the northwestern U.S. regional program is FY 1981. Program redefinition in the eastern regions will begin in FY 1982 and will depend on the results beginning to emerge.

6.1.5.2 Topical programs

Topical programs to study specific technical issues will improve the ability of the NRC to assess seismic and geologic hazards. Measurements of earth-stress amplitude and orientation, geotechnical studies of methods for predicting soil failure, and studies of hazard evaluation methodology make up the three programs.

The earth-stress measurement program is a long-term effort but at a low level of support; it is intended to encourage needed efforts in this potentially large and important area of improving the understanding of basic causes of intraplate earthquakes. Projects in the geotechnical studies program are intermediate term, and present studies should be completed by FY 1983. The hazards evaluation program is made up of a number of short-term projects addressing specific topics or issues of interest ranging from data compilation and analysis for historic earthquakes to studies of seismic wave generation and propagation; nonseismic hazards such as volcanic effects are included.

6.1.5.3 Milestones

The schedule of major objectives and important milestones is shown below.

1. Seismotectonic-province map of eastern United States

FY 1983	Interim
FY 1986	Final

Necessary Map Components

FY 1983	a.	Seismological summary of controlling earthquakes used in design
FY 1983	b.	Methodology and justification for using recurrence intervals and maximum earthquakes in hazard assessment
FY 1982	c.	Statement of position of active and capable faults in the eastern United States
FY 1984	d.	Statement of position on areal extent of Charleston, S.C. seismic activity
FY 1985	e.	Characterization of earthquake hazards in southeastern areas other than Charleston
FY 1984	f.	Statement of position on northeastern extent of New Madrid fault zone
FY 1983	g.	Characterization of earthquake hazards in the Nemaha Uplift region

2. FY 1985 State-of-the-art summary of the role of vertical geodetic changes in tectonic evaluations

3. FY 1983 Recommendations for regulatory guide on evaluating transient soil deformation from earthquake loading

6.2 Meteorology and Hydrology

Two primary needs exist within the NRC that warrant research in meteorology and hydrology: (1) the need to locate and predict the behavior of radioactive plumes in both the air and water pathways, and (2) the characterization of natural phenomena (tornadoes, lightning, hurricanes) and their contribution to the safety (and risk to the public) of nuclear facility design. Research in meteorology and hydrology is directed toward obtaining high quality data and evaluating existing and proposed models used to describe the movement of radioactive particles and gases. The research program is comprised of three elements: (1) atmospheric dispersion: (2) severe storms characterization; and (3) ground water transport. The ability to address accurately and effectively questions related to the radioactive plume location and behavior and the contribution to the overall risk to the public from natural phenomena induced failures to nuclear facilities is urgently needed by the NRC. Federal regulations require that accurate assessments be made related to safety and environmental concerns for site characterization and emergency planning purposes.

6.2.1 Regulatory Objective

The capability to accurately predict the spread of accidentally released airborne radioisotopes is needed for two licensing and safety-related concerns: (1) emergency response planning, and (2) site characterization. A determination of the present and future location, transport, and diffusion of radioactive plumes is a requirement of Appendix E to 10 CFR Part 50 pertaining to an assessment of the adequacy of emergency plans. The complexities of dispersion processes are addressed in the site and safety evaluation phases leading to the granting of an operating license. Transport and diffusion of airborne radioactive effluents are integral components of assessing the overall risk to the public from nuclear facilities. Assessment of the consequences of accidental releases of radioactive material is determined, in part, by appropriate modeling of atmospheric dispersion processes.

6.2.2 Technical Capabilities Required

To achieve the regulatory objectives, capabilities are needed to identify, evaluate, and verify the current and proposed atmospheric transport and diffusion models used to predict the behavior of radioactive airborne effluents during accident conditions. Capabilities are also needed to characterize the severe storm hazards, tornadoes, and lightning, and their variation in space and time. Identifying and characterizing the behavior of radionuclides in groundwater flows is required to assess the potential consequences of loss of containment integrity.

6.2.3 Status of Capabilities

Research in atmospheric dispersion involves the development, execution, and analysis of empirical field and wind-tunnel tracer tests. Such tests are necessary to acquire high-quality tracer concentration and meteorological data needed to verify or evaluate atmospheric diffusion and transport codes used during conditions of accidental release.

Nuclear power plants are required by Appendix A to 10 CFR Part 50 to be designed to shut down safely when they are influenced by natural phenomena such as tornadoes. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants" was based on WASH-1300, "Technical Basis for Interim Regional Tornado Criteria." This study utilized only 2 years of coarse tornado data to formulate the current NRC position. Additional data and improvements in data quality must be acquired to evaluate the adequacy and accuracy of Regulatory Guide 1.76. Category I structures, systems, and components must be designed to withstand the effects of violent tornadoes and severe lightning strikes.

Following the TMI-2 accident, concern was voiced over the potential consequence of loss of containment integrity from a similar but more serious accident, by which the liquid contents might enter the short ground-water pathway segment between the plant and a nearby surface-water body. Generic criteria for site evaluation will be developed and investigations conducted to determine possible methods for interdicting contaminated ground water such as dewatering, formation of freeze walls, and construction of slurry trenches.

Previous atmospheric dispersion research results have been incorporated into Regulatory Guide 1.145. SD and NRR have endorsed the dispersion research program in its entirety. NRR-79-19 was separately endorsed by SD; NRR-79-5 has been received, and a reply is being formulated. ACRS endorsed the meteorological research program in its experimental, analytical, and evaluation phases.

SD endorsed the tornado research program in its entirety, particularly the contract with Texas Tech University. RES:RSR-135 recommends that work be performed at the University of Chicago. Research support has been requested by NMSS (RR-NMSS 79-5) for a site-specific tornado analysis, and requests continue to be received from NRR concerning the tornado characteristics for the SEP plants. The ground-water contamination study was recommended by the ACRS during the FY 1982 program review. User-office concurrence is in process.

6.2.4 Research Program Objectives

Atmospheric dispersion of radioactive effluents and severe storm characterization comprise the meteorological research program. The objective of the atmospheric transport and diffusion research program is to evaluate meteorological dispersion models used to describe the spread of radioactive plumes. The objective of the research on severe storms is to determine realistically the maximum tornado and lightning characteristics that can occur anywhere in the contiguous United States and to estimate the spatial and temporal variations of these characteristics. Hydrological research in this program pertains to the contamination, movement, and possible interdiction of groundwater that might follow from a core-melt accident; the objective is to develop basic information for establishing pertinent evaluation criteria for the hydrological characteristics of existing and future nuclear power plant sites. Probabilistic assessment of flood hazards and flooding effects are described in Section 4.5 of Chapter 4, Plant Operational Safety, and in Chapter 7, Light-Water Reactor Risk Assessment. Additional research on ground-water hydrology pertaining to waste repositories is described in Section 12.1 of Chapter 12, Waste Management.

6.2.5 Research Program Plan

The current atmospheric dispersion program was initiated in 1972 with the conduct of tracer experiments performed during low windspeed and inversion conditions over flat terrain. The structuring of the elements comprising the program made extensive use of previous work in this area which began in 1947.

Most of the previous experiments were conducted on an "as needed" basis; no formal long-range research program was developed. The effects on dispersion of topography, thermal stability, and turbulence are being examined empirically and analytically. Tracer experiments are being conducted over a range of topographical features with concentrations being measured out to 50 miles. Dispersion models are being evaluated using existing high-quality tracer concentration data sets. This evaluation will demonstrate the range of models applicable to different meteorological and topographical regimes. Such an assessment will provide a basis for selecting a given model for use in emergency response planning and environmental effects resulting from postulated accidental releases of radioactive effluents for site evaluation purposes. Field program plans have been initiated to obtain high-quality tracer concentrations and meteorological measurements under controlled conditions in flat, even terrain and in river-valley and shoreline environments. Each field program will take 4 to 5 years to complete because of the complexities inherent in conducting atmospheric tests to distances of 50 miles. For example, the shoreline experiments consist of field tests at the Gulf of Mexico, the Great Lakes, and the Atlantic Coast.

Research related to nuclear applications in tornado and lightning distributions and verification modeling of maximum severe storm characteristics were initiated in 1972; this work in tornadoes and lightning is essentially completed. Some tornado field measurements and damage assessments will continue in order to maintain expertise in this area and to add to the quantified observational data base that was almost totally lacking before the initiation of this program.

A series of field experiments will be conducted to obtain high-quality concentration and meteorological measurements over a range of terrains and to distances of 50 miles. Emphasis will be placed on river-valley and shoreline environments. Models will be evaluated for each test series in order to objectively assess the ability of each model to predict the maximum concentration and spread of the tracer. Such an assessment will provide a basis for selecting a given model for use in emergency planning and site evaluations. The schedule of major objectives and important milestones is shown below.

FY 1982	1.	Field experiments in flat, even terrain
FY 1983	2.	Field experiments in river-valley, rolling terrain
	3.	Field experiments in shoreline environments
FY 1983		a. Gulf Coast
FY 1985		b. Great Lakes
FY 1987		c. Florida Atlantic Coast
FY 1984	4.	Assessment of meteorological dispersion models for use in real-time response to emergency situations

PART B. SYSTEMS AND RELIABILITY RESEARCH

SYSTEMS AND RELIABILITY ANALYSIS

Systems and reliability analysis entails the application of reliability engineering techniques to nuclear safety issues throughout the nuclear fuel cycle. Before the TMI accident, systems and reliability research constituted the principal NRC research into accident sequences that can lead to severe core damage and fuel melting, and the advanced reactor safety work (See Section 5) was the principal NRC research into the phenomena of core damage and melting. The TMI-2 accident has highlighted the importance of both these areas of research and has led to substantial expansion of research in both areas. The technical research disciplines supported by systems and reliability analysis are (1) system reliability analysis, (2) accident sequence analysis, (3) risk assessment, (4) human reliability and human factors research applied to operations personnel at nuclear facilities, (5) operations research, and (6) severe accident consequence analysis.

Research in this area serves three basic functions: (1) the performance of risk assessments to gain perspective on regulatory safety issues and to support regulatory standards development, (2) the development of reliability evaluation techniques and criteria for direct application in the regulatory process, and (3) technical support for the Commission, the adjudicatory boards, and the regulatory offices in matters of reactor and fuel cycle risk assessment and fuel-melt accident analysis. Historically, the Office of Nuclear Regulatory Research (RES) has had the task of developing the methods for risk assessment and applying them as well. With the renewed emphasis on the development and use of risk assessment techniques, expertise is developing in this field in other parts of the NRC. The coming 5 years should see a substantial transition of risk assessment applications to other components of the NRC, and we will do all we can by development of methods and by training and assistance to facilitate that transition. The basic objective of systems and reliability research is to bring nuclear regulation into better congruence with the risks; that is, to identify and close gaps in regulatory requirements in risk-significant areas, to identify instances of off-target or unnecessary regulations, and, ultimately, to stabilize the regulatory process.

Systems and Reliability Analysis performs an integrating function for the phenomenological research performed under other RES programs. Risk assessment perspectives are being used increasingly to guide research priorities throughout the office. In addition, the research results developed elsewhere in RES provide some of the technical foundations for risk assessment and reliability analysis performed under this program, as well as provide direct input to regulatory decisionmaking.

The risk and reliability research supported by Systems and Reliability Analysis has the strong endorsement of the President's Commission on the Accident at TMI, the NRC Special Inquiry, the Risk Assessment Review Group, and the Advisory Committee on Reactor Safeguards (ACRS).

The budgetary unit "Systems and Reliability Analysis" is composed of (1) Methodology Development, (2) Human and Reliability Data Analysis, (3) Systems Analysis, and (4) Consequence Analysis. These constitute a disciplinary or functional decomposition of the research. However, many research projects cut across these categories, as does the regulatory context and justification; therefore, a different framework for the Division of Systems and Reliability Research has been developed for the purpose of this plan to reflect common regulatory objectives. These units are (1) Light-Water Reactor Risk Assessment, (2) Fuel Cycle Risk Assessment, (3) Human Reliability Research, (4) Reliability Engineering, Operations Research, and Decision Theory, and (5) Consequence Analysis.

7. LIGHT-WATER REACTOR RISK ASSESSMENT

7.1 Regulatory Objective

A principal objective of the NRC is to ensure that licensed nuclear power plants pose no undue risk to the public health and safety. Research in probabilistic risk assessment (PRA) applied to light-water reactors constitutes an independent validation of the effectiveness of the regulatory program with respect to public risk limitations. In addition, PRA provides a framework to gain perspective on many facets of LWR safety. For example, PRA is used to identify gaps in the reactor safety regulations, to give a measure of risk relevance to safety issues, and to measure the value of new regulatory requirements or standards to assure a more balanced and effective regulatory program which puts the emphasis on the real safety needs.

Since the accident at TMI, there has been a move to employ PRA techniques directly in reactor safety regulation as well as in the confirmatory role. In particular, it has been suggested that methods developed for PRA be employed in core-melt accident analysis and in studying the likelihood and consequences of a broad spectrum of accidents at reactor plants.

7.2 Technical Capabilities Required

Probabilistic risk assessment requires a cadre of practitioners, a set of methods or analytical tools, and a data base. The methods embrace the construction and evaluation of mathematical models. These models describe accident initiation, propagation, and consequences. The data base deals with the likelihood of initiating events and contributory failures, including human error.

When the principal result of PRA is to be an absolute prediction of risk, then the completeness of the methods and the scope and accuracy of the data base are of prime importance. In other words, in its confirmatory role, PRA is particularly demanding of the methods and data base.

When the principal objective is to shed light on particular safety issues, the accuracy with which PRA can predict other contributors to risk is less important. Thus, PRA is not so demanding of comprehensive methods and data in its role as a source of perspective on specific aspects of the regulatory program. However, there are so many contexts in which PRA perspectives might be useful that such applications can be quite demanding of the cadre of practitioners.

As PRA comes to be employed in casework safety analysis for licensees, the emergence of standardized, broadly understood methods and data becomes crucial. Large numbers of practitioners will be needed to evaluate licensee submittals.

7.3 Status of Capability

The cadre of experienced practitioners of PRA in the agency is sufficient only for either (1) an occasional confirmatory probabilistic risk assessment such as the Reactor Safety Study (WASH-1400) or (2) limited use of PRA to shed light on specific safety issues or system reliability problems.

The methodology in hand is sufficient for a wide array of issue-specific or comparative applications of PRA. However, the methods available today are not complete; a number of contributory causes of accidents or aspects of accident phenomenology cannot be accurately portrayed in quantitative risk-predictive models. Presently under development are methods or techniques to deal with fires and certain types of plant flooding. Also lacking are proven models of accident progression involving complex feedback effects, partial failures, human errors (particularly those of commission during accidents), some common-cause failures, design and construction errors, and some external events. Our understanding of core-melt and containment-challenge phenomenology and of the magnitude of radioisotope releases is incomplete. These imperfections in PRA methodology limit the accuracy with which absolute and comprehensive risk assessments can be made.

Much of the current PRA methodology has been given only arcane or incomplete documentation. It is not in the readily accessible, standardized, and widely acknowledged form that will be needed if PRA is to become a routine regulatory tool for use in plant-specific licensing evaluations.

The data base for probabilistic risk assessment is composed of estimates of the likelihood of initiating events, compound failures, and operator errors or operator repair actions. Human reliability research is treated in Chapter 9. The data base on the frequency of initiating events has not been updated by the NRC since the Reactor Safety Study. The Electric Power Research Institute (EPRI) has prepared a statistical analysis of transient initiators, EPRI NP-801, but this has not been validated by the NRC. Additional research is needed to pin down initiator frequency, particularly for rare but risk-significant failures in reactor auxiliaries such as AC and DC power systems.

Some component failure rates have been reevaluated recently to take advantage of reactor operating experience since the Reactor Safety Study. Generally, the industry-average generic failure rates are known within a factor of 10, though there are exceptions for highly reliable (low failure rate) components. Plant-specific failure rates and the variation in reliability among different makes and models of generically similar equipment is not well known, nor are the current industry reporting practices sufficient to support such data analysis. Typical outage times are less well known than failure rates for many safety-related components, and the likelihood that repair can be achieved inside accident conditions is still less known.

In general, component failure and unavailability data suffices for approximate probabilistic risk assessment, particularly using generic, industry-average failure rate data. The data is good enough to sort multiple random-failure scenarios into coarse likelihood classes, but not to make precise estimates of accident likelihood. The absence of a comprehensive data base on make-, model-, or plant-specific components, on repair rates, and on the effects of differing maintenance practices limits the precision with which both absolute and comparative probabilistic risk assessment can be performed.

Statistically correlated multiple failures are coupled by a causal link. They are of particular interest to risk assessment and nuclear safety because they can defeat redundancy or defense in depth. One can treat these common-cause failures as a problem in methodology (model the causal linkage) or as a problem in data (estimate the likelihood of coupled, multiple failures). Improvements have been and are being made in dealing with common-cause failures through both avenues of attack, but they remain among the factors limiting the completeness and accuracy of PRA.

7.4 Research Program Objectives

One of the principal constraints on the use of PRA in reactor safety regulation is the number of experienced practitioners in the NRC. Since the pool of practitioners outside the industry is not large and the prospects for substantial staff expansion is not promising, a principal objective of the research program in LWR risk assessment is the development of a training program and the publication of training materials that will enable the agency to fulfill the need for PRA practitioners by retraining currently employed engineers.

A second objective of the LWR-PRA research program is the development and maintenance of a current best estimate of the accident risks posed by commercial light-water reactors, together with estimates of the uncertainties in the risk assessment.

A third objective of the program is to develop standardized PRA techniques and data to enable PRA to become a routine licensing tool, particularly in comparative or issue-specific contexts in which its comprehensiveness is less crucial than in absolute, wholistic risk prediction.

A fourth objective of the program is to develop a number of applications of reactor PRA to shed light upon a variety of regulatory issues:

1. Research in support of reactor safety standards development and regulatory reform;

2. Research in support of improved techniques to evaluate the risk significance of reactor operational occurrences;

3. Research to expose safety-significant loopholes in regulatory equipment;

4. Research to establish a risk-relevance measure for use in determining priorities for regulatory issues and research and to improve the safety effectiveness of NRC staff utilization and value-impact assessment.

7.5 Research Program Plan

7.5.1 Staff Training

A series of training courses will be given to about 250 headquarters personnel each year in FY 1983-1985. These courses will give students an overview of the results of risk assessment and reliability engineering, as well as a passing familiarity with the strengths and weaknesses of the methods.

In addition, intermediate and advanced workshops are needed for those who might specialize in probabilistic safety analyses and risk assessments and for those who might review licensee submittals employing these techniques. Topics will include probability and statistics for reliability analysis; accident sequence analysis; system reliability analysis; core-melt and containment-challenge modeling; consequence analysis; reliability engineering and reliability assurance; human reliability analysis; and waste repository risk assessment.

Workshops should be conducted for about 30 agency professionals every 2 years in each of these areas.

I&E already maintains a highly professional training program for inspectors. RES will work with the I&E training staff to include in the I&E courses an overview of risk assessment results and highlights of those aspects of human and system reliability analysis that are of particular value in monitoring reactor operations.

7.5.2 Revised Reactor Safety Study

A revision of the Reactor Safety Study is scheduled for FY 1983-1987. Like the original, the revision is intended to develop the most accurate possible assessment of the risks posed by externally and internally caused accidents (excluding sabotage) a' commercial LWR plants. However, it will not be modeled on WASH-1400. There will be three principal tasks in the new Reactor Safety Study. One of these will be an intensive risk assessment of a single plant, which has been the subject of a previous Interim Reliability Evaluation Program (IREP) study. The objective of this intensive study will be to identify the strengths and weaknesses of less intensive risk assessments li' IREP. This task will use very thorough accident-sequence analysis, inclu .ng a full range of those sequences that stop short of core damage, those that end in core damage, and those that progress to full meltdown. Fires, floods, earthquakes, and other external events will be considered. The system reliability analyses will be more detailed than those used in WASH-1400 or IREP. A thorough analysis of operator intervention (in both errors and corrective actions) will be made. Diverse methods of system reliability analysis will be employed (e.g., sneak-circuit analysis). All recent improvements in failure rate data, methods for treating common-cause failures and human factors, core behavior phenomenology, containment-challenge phenomenology, and consequence analysis will be used. It will not be assumed that risks found in the plant studied apply to all reactors; however, the kinds of modifications to prior, less intensive risk assessments that are found to be necessary for the plant studied may be applicable to other reactors.

The second principal task in the Reactor Safety Study revision will be the evaluation of a family of risk assessments for all domestic power reactors. These assessments will range from those that are highly dependent on a model and incisive, to those that are assumption-free but merely bounding. In this way, risks to the public from nuclear plant operation can be compared with the level of dependence on modeling and on data assumptions that are not totally verified.

Each power plant can be characterized by a set of (unknown) occurrence rates for classes of accidents of varying severity (including precursor events). The consequences of each accident class can be analyzed. Several cases will be evaluated using no assumptions, weak assumptions, and strong assumptions on the comparative frequency of accidents of different severity at any one plant, on the frequency differences for similar accidents at different plants, and on the time dependence of the occurrence rates. These models will then be fit to the experience data on accidents and accident precursors to obtain risk assessments or risk-bounding assessments.

The third principal task of the Reactor Safety Study revision will be a synthetic industry risk assessment. It will be based on (1) the 20 or more published plant-specific risk assessments available at that time, (2) an uncertainty analysis based on the peer review of those studies, and (3) the insights from the intensive study of one plant on the biases or oversights in studies like IREP or WASH-1400. The synthetic analysis will be compared with the actuarial bounds developed in the second task. In addition, the synthetic analysis will be interpreted to identify the principal means of determining the accident risk at commercial nuclear plants. Risk-importance measures will be developed for classes of accidents and classes of component failure or for human-error contributors to these accidents. Guides for assigning priorities to regulatory initiatives, standards development, and research will be developed from the engineering insights.

7.5.3 Methods Development for Accident-Sequence Analysis

The development of methods for accident-sequence analysis for reactors is a new program. The first phase is the development of a hierarchical classification scheme for reactor accidents. This scheme is to provide a catalog of LWR accident scenarios of various kinds and degrees of severity that can be used in qualitative, deterministic, and probabilistic safety analysis; in operator training; in planning for emergencies; and in the analysis of operating experiences. One of the principal subtasks will be to continue the study of "precursor events" and their risk significance. This work will be performed in coordination with the Office of Analysis and Evaluation of Operational Data (AEOD). Methods will be developed to facilitate and codify the results of the analysis of events to determine:

1. The risk significance of the event;

2. Its potential risk significance in alternate sequences;

3. The adequacy of prior analysis of similar events; and

4. The adequacy of regulatory requirements in limiting the risk posed by similar occurrences.

This task will be aimed at meeting the needs of NRR, I&E, and AEOD in experience screening. It will include the processing of Licensee Event Reports (LERs).

A second principal phase of this program is improving event-tree analysis to overcome limitations relating to partial failure or core damage, the complex or varying chronology of the progression of some accidents, and the wide array of initial conditions from which accidents may evolve. Much of the initial work in these areas will be developed through "learn-by-doing" applied programs (for example, research for the degraded-core rulemaking and the revision of the Reactor Safety Study). However, in the late 1980s, the methodology development projects will be separated from applied risk assessment projects so that they can be digested and systematized.

7.5.4 Common-Cause Failure Analysis and Hazardous External Events

The goal of this research work is to develop models, data, and computer codes for the systematic analysis of common-cause failures and external events as they affect safety system reliability and accident probabilities and consequences.

Multiple concurrent component failures and external hazards are of particular importance to reactor safety; they can compromise the redundancy or defense in depth that is entended to ensure safety in the plants. There are several kinds of common-cause failures. Some multiple failures originate in the failure of a common support system; these are treated in network system reliability analysis techniques. Other multiple failures can be traced to common human intervention (operations or maintenance); these are treated in human reliability research. Two other varieties of common-cause failure are the subjects of this research program: (1) multiple faults originating in common design, fabrication, installation, startup, operating history, or service environment, and (2) external events such as earthquakes, site flooding from direct natural forces or the failure of flood prevention features, hurricanes, or tornadoes.

Much of the work in this area is paced by the data analysis program. The development of methods for common-cause failure analysis of pumps, valves, control rod drive mechanisms, and diesel generators is underway. As the data base development is extended to other classes of components, the development of methods of common-cause failure analysis also will be extended to those components. Some other approaches to common-cause failure analysis will originate from systems analysis research projects such as the project on design errors.

Current work relating to inplant flooding and river basin flooding will be completed by FY 1982. Work on seaside flooding, inplant fires, tornadoes, and hurricanes is planned to be done in the mid-1980s. This work will be coordinated with the Seismic Safety Margins Research Program described in Section 4.

7.5.5 Methods Development for System Interactions and Reliability Analysis

Research to improve upon current capabilities in system interaction analysis, system reliability analysis, and failure mode prediction are slated for FY 1982-1987. Improved mathematical models of system networks will be developed to deal with feedback effects, delayed and conditional fault propagation, and partial failures and to accommodate improved models for human reliability, both human error and human corrective action. Methods such as the "GO" codes, sneak-circuit analysis, logic-circuit simulation, diagraph methods, matrix methods, Markov methods, and dynamic simulation will be explored and adapted to the needs of nuclear safety system reliability analysis.

NRR needs a practical set of techniques to deal with systems interactions in licensing. A focus of the research into improved system reliability analysis techniques will be research to meet this need. A variety of reliability engineering techniques that show promise for systems interaction evaluation will be explored, e.g., fault trees, network models of fault propagation, failure mode effects analysis, and common-cause failure analysis software.

Improvements in probabilistic assessment are also planned to better resolve failure modes of equipment. Some types of failures lend themselves to preventive maintenance or to identification and repair before the fault has progressed from incipient failure to total failure. Methods to resolve such failure mode characteristics in data analysis and predictive system reliability analysis--coupled with improved assessments of the risk significance of hypothetical failures developed in applied systems analysis--will be of use to I&E and SD in focusing upon the more effective ways of reducing reactor accident risks, as well as improving safety analysis and risk assessment.

7.5.6 Development of a Reliability Data Bank

7.5.6.1. Assumptions and Background

By affirming SECY 80-507, Integrated Operational Experience Reporting System, the Commission has approved a staff proposal to assume responsibility for the management and technical direction of the revised Nuclear Plant Reliability Data System (NPRDS).

In recent years, RES has been updating some of the WASH-1400 component failure and human error probability or frequency estimates. Pumps, valves, control rod drives, diesel generators, and selected electrical equipment failure rates have been reassessed, based on LER data. Continuation of such studies is planned in this research plan. However, the limited scope and validation of these data analyses make them an insufficient basis for reliability or risk calculations should the regulatory process depend on the accuracy of these calculations. Therefore, the more comprehensive, better validated NPRDS is essential to the use of probabilistic risk or reliability assessments in licensing, in technical specifications, or in enforcement.

7.5.6.2. Data Bank Methodology

The objective of this research is to develop computer codes and procedures by which raw data on reactor operating experiences can be transformed into component reliability data.

Different measures of the reliability of components are revised as additional information about reactor operation is accumulated. These reliability parameters include failure probability per demand or per duty cycle, failure rates per hour of service, component unavailability because of testing or maintenance, and repair times. A number of safety-related components are covered. The data is assembled according to the type of component involved, and the average and range of reliability are obtained for generically similar equipment for all commercial LWRs, for all plants of the type, and for individual plants. Data sources include LERs, the NPRDS, and studies of selected plant operations and maintenance logbooks. The data analysis includes quality control of the data base through checks of the consistency of the three sources of data.

The resource requirements for the component-reliability data base-development program to develop a data base on component reliability are influenced by the following: (1) many types of components important to safety are not covered in the ongoing program, and (2) under the LER and NPRD systems, reporting procedures vary and reports are incomplete. As a result, at present it is not possible to implement automated data analysis procedures on a wide scale and, instead, more surveys of plant maintenance logbooks are required. Recommendations have been made to revise reporting requirements to ensure their adequacy and to standardize keywords. It should then be possible to automate and streamline a large part of the data analysis work done by the agency.

7.5.6.3 Development of Standards for Experience Reporting

AEOD has suggested that RES take over management of the reporting required under the NPRDS. SD, AEOD, and RES are collaborating on a possible rulemaking to change reporting requirements under the LER system and to require licensee participation in a modified NPRDS.

7.5.7 Analysis of Challenge Frequency and System Availability Data

A data base that treats challenges to safety functions in nuclear power plants is required. Data will be assembled and analyzed on the kind, frequency, duration, and severity of the challenges. In addition, an actuarial data base will be developed on the availability of safety functions (whether challenged or not) and on challenges in which one or more trains of a safety system fail to respond. This information is of direct application in safety reviews of operating plants and will permit the verification of calculated system reliability derived from component and human failure rates.

Safety system challenges covered in the data base include both inplant and external events such as inplant fires, floods, LOCAs, transients, and transient-induced LOCAs; site flooding; hurricanes; tornadoes; earthquakes; and transportation and industrial accidents.

Statistical analysis will include the intermediate and root causes of the safety challenges within the plant. This will facilitate the analysis of accident sequences in which the initiating event and the failure of the safety function share a common cause (e.g., a support system failure).

Only data on inplant and external floods are now being updated; assessments made for WASH-1400 are not yet being updated.

7.5.8 Risk and Reliability Analysis Methods for Use in Licensing

7.5.8.1 Methodology Development

In the FY 1980-1982 interval, RES will have turned over to NRR a variety of standardized procedures for system reliability analyses or risk assessments for direct use in licensing. These include (1) procedures for reliability studies of PWR auxiliary feedwater systems; (2) a checklist of dominant accident sequences found in risk assessments and of the design or procedural features responsible for the prominence of these sequences in the risk picture; and (3) standard procedures for the application of abbreviated probabilistic risk assessment to all licensed reactors under the Interim and National Reliability Evaluation Program.

Continuing methodology development is expected in FY 1983-1987 for risk and reliability analysis procedures to be used by licensees and submitted to NRR in licensing cases.

The elements of this program are:

1. Development of new methods to extend the scope and completeness of regulatory risk assessment;

2. Quality assurance criteria for licensee-performed probabilistic risk and reliability studies;

3. Development of a standard review plan for the evaluation by NRR of licensee submittals of risk or reliability studies;

4. Pilot-program applications of analytic procedures to verify their workability prior to turnover from RES to NRR (see Section 7.5.8.2); and

5. Validation of risk assessment and reliability prediction techniques.

7.5.8.2 Pilot Study Applications of Risk-Based Regulatory Methods

IREP Phase II is the prime example of pilot study applications of risk assessment and system reliability analysis techniques intended for use in the regulatory process. This program is designed to evaluate standardized techniques of probabilistic safety analysis that can subsequently be required of many or all licensees.

Similar pilot studies are planned in FY 1983-1987 for improved risk assessment techniques, including methods to deal with fires, floods, earthquakes, external hazards, improved models for human error and human corrective action during accidents, and improved techniques to distinguish core damage from meltdown outcomes.

7.5.9 Applications of Risk Assessment Results

7.5.9.1. Technical Implications of Risk Assessment Studies

By FY 1983, at least 11 reactor risk assessments will have been published by the NRC, and the industry is expected to have published several others. A group of research programs are planned to validate these studies and to develop regulatory applications of the results.

1. Handbook of Core-Melt Accident Sequences

A handbook will be assembled that describes the accident sequences that have been predicted to contribute significantly to accident risks. It will address the chronology, phenomenology, symptoms, consequences, and likelihood estimates for the sequences.

This program will draw upon improved methods for accident sequence analysis, upon improved understanding of severe accident phenomenology, and upon risk assessment results. It should be of use both inside and outside NRC in contexts that include safety design, safety analysis, standards development, operator training, emergency procedures, and emergency plans.

2. Risk-Importance Measures for Hardware and Procedures

A quantitative risk assessment of a nuclear power plant provides the basis with which quantitative measures of risk significance can be obtained for components and systems or for human interactions with the plant. One may calculate the change in overall accident risks resulting from a change in the likelihood of component failure, of human error, or of repair during an accident. Ratios of these changes are called importance measures.

Importance measures for component and system failures, human errors, and human repair actions will be calculated, together with their uncertainties, for some of the better reference reactor risk assessments available in the mid-1980s.

The importance measures will be checked against a Failure Mode Effects Analysis (FMEA) to verify whether the importance measures make sense. This is a valuable form of validation for risk assessment as it constitutes a disciplined approach to independently check the scope and limitations of the probabilistic study. Reports on the limitations of the particular study and on insights for the improvement of future risk assessments will be published.

Those importance measures for component failures and human inputs that do pass the test of comparison with FMEA results and engineering judgment will be published for use in safety analyses. They will also be used in a number of research programs, e.g., human reliability research, research in support of reactor safety standards development, improved methods for system reliability analysis, reliability engineering, methods development for ascertaining the risk significance of operating occurrences for AEOD.

3. Risk-Based Guidance for I&E Inspectors

This program, to be performed in close coordination with I&E, will provide guidance for I&E training and for I&E field operations to enhance the risk relevance and risk-reduction effectiveness of inspection and enforcement activities.

4. Occupational Exposure Versus Public Risk

Tradeoff studies will be made to compare the risks associated with occupational exposure in inservice inspection, test, and maintenance with the effect on societal risk of performing more or less such surveillance and maintenance. The objectives of this project are to identify an optimum frequency for high-dose surveillance and preventive maintenance activities and to identify those contexts in which additional automated or remote-surveillance equipment might be cost- or safety-effective.

5. Risk-Based Formulations for Technical Specifications

A program to employ probabilistic risk and reliability analysis results to optimize allowable outage times for safety components is continuing. The scope will be extended to include other limiting conditions of operation.

7.5.9.2 Applications for Reactor Safety Standards Development

Several specific research projects in support of reactor safety standards development will have been completed by FY 1983. These include (1) station blackout (TAP A-44), (2) DC power (TAP A-30), (3) risk surveys in support of siting, emergency planning, and degraded core rulemakings, and (4) evaluation of the risk-reduction potential and design criteria for vent-filter containment concepts and for add-on decay heat removal systems.

In FY 1983 and thereafter, we anticipate risk assessment studies in support of the standard engineered safety fracture rulemaking. (See also Chapter 10.)

7.5.10 Special Studies of Component Failure Rate Data

Through the component reliability data base development program described, many safety-significant areas that require special study have been identified that will require technique and software development. These include:

1. Abnormally high failure rates of certain components and, in particular, plants that may warrant flagging and followup by I&E, AEOD, or NRR;

2. Time trends in failure rates (e.g., high rates of failure in new plants and trends for equipment wearing out in older plants);

3. Greater-than-random occurrence rates for multiple concurrent failures or clusters of failures in components. (These suggest common-cause failures originating in component design, manufacture, installation, or service environment. Testing or maintenance occurrence rates for, and root causes of, these common-cause failures are of particular interest because they may compromise redundancy in safety systems.)

4. Wide variations in the reliability of components from plant to plant or among different applications or different makes and models of similar equipment. (These can be traced to root causes such as design, manufacture, installation, or testing or maintenance practices. For example, the plant owners' policy regarding preventive maintenance may be responsible for plant-to-plant differences in the rate for component failures).

8. FUEL CYCLE RISK ASSESSMENT

The scope of fuel cycle risk assessment embraces the development and research applications of probabilistic risk assessment to licensed nuclear facilities other than reactors.

8.1 Regulatory Objective

There are three objectives of fuel cycle risk assessment research in support of the broader agency mission to regulate nuclear activities to protect public health and safety. These are:

1. Provide risk assessment methods suitable for use as licensing safety analysis tools where appropriate;

2. Provide perspectives and techniques for use in regulatory standards development, including risk-based value-impact analysis, risk-based priority assignments, and risk-relevance or risk-importance assessments; and

3. Provide perspectives on the effectiveness of the regulatory program by estimating the residual nuclear risks for licensed activities.

8.2 Technical Capabilities Required

The three principal resources required for regulatory risk assessment applications are risk assessment analysis, the repertoire of analytic methods, and a data base. The methods entail mathematical modeling of the course and likelihood of scenarios in which radioactive materials are released in the biosphere or people are exposed to ionizing radiation. The analysis of the effects of uncontrolled radioisotope entry into the biosphere are treated under Consequence Analysis in Chapter 11. The data base addresses sources, environmental conditions affecting release and transport, and the likelihood of release scenarios.

The draft EPA standard for high-level waste (HLW) disposal is an example of a risk-based regulatory standard. The NRC will need the capability to determine if the EPA standard is being met for specific sites, to assess ways of implementing the EPA standard, and to assess standards for waste repositories with respect to risk limitation. The capability must extend to a variety of geologic media and waste forms.

The principal feature of probabilistic risk assessment that distinguishes it from conventional or deterministic quantitative nuclear hazard analysis is the incorporation of likelihood assessments. Thus, the technical capability that is the key to transforming dose commitment assessments for fuel cycle activities is the modeling and evaluation of the likelihood of the exposures.

8.3 Status of Capability

Risk assessment methods adapted for the analysis of geological repositories of HLW have been developed and are in use. A training program, described in Section 8.5, is in place to provide experienced practitioners within the NRC and a forum for regulatory applications of HLW repository risk assessment.

Current analyses have been confined to bedded salt repositories holding processed HLW. However, work is underway to generalize these studies to include spent fuel as the waste form and other geological media for the repository.

Adaptations of these methods are needed to deal with low-level waste repositories and uranium tailings disposal sites. The waste repositories would include sites of nuclear facility decommissioning. The agency also needs consistent methods and standards for risk assessment of the operation of nuclear facilities such as uranium mills, UF_6 plants, fuel fabrication and processing plants, and research and isotope production reactors. Scoping studies of the risks associated with such facilities are needed to provide a balanced perspective of their risks as compared to large LWRs and the waste disposal sites. Similar studies are needed to relate to previous analyses of nuclear material transportation risks and the risks associated with radioactive industrial, medical, and consumer products.

This research plan does not include risk assessment for breeder reactors or for reprocessing plants. In the event that the nation goes forward with LMFBRs, it will be essential to develop applicable probabilistic risk assessment tools for use in licensing breeders, particularly because DOE has chosen to use safety design techniques closely related to probabilistic risk assessment for LMFBR research and development. These techniques are probabilistic risk apportionment and a levels-of-assurance philosophy that entails probabilistic reliability criteria. These approaches to safety design will need validation in the licensing process through both PRA and conventional safety reviews. Thus, it is assumed that any decision to press forward with breeders and/or reprocessing would entail the addition of a corresponding program in risk assessment research in RES.

8.4 Research Program Objectives

The fuel cycle risk assessment research programs are of two kinds. The first are called risk estimation surveys or risk bounding studies. They are intended to be extensive rather than intensive; they entail relatively little methodology development or data analysis. Rather, they are based on existing methods and published data, however limiting these constraints may be. They are intended to give rough estimates--with upper and lower bounds--to the societal and individual risks posed by each NRC-regulated endeavor.

There are two objectives of these studies:

1. To provide laymen and nonspecialists in risk assessment with rough estimates and bounds upon the absolute risks posed by NRC-licensed enterprises for comparative purposes; and

2. To provide risk perspectives for policy decisions, standards development, and research planning.

The studies will document the factors responsible for the spread between upper and lower risk bounds. Thus, they serve as exploratory surveys of what research needs to be done in order to pin down the risk more precisely.

The second family of fuel cycle risk assessment research programs entails methodology development, data analysis, and research applications of more intensive and definitive risk assessments. The objectives of these programs are to improve the accuracy of risk assessment, develop research applications needed for standards development and the risk-based confirmation of the regulatory program, and to develop the cadre of fuel cycle risk assessment practitioners.

8.5 Research Program Plan

A variety of risk assessment research projects bearing upon the nuclear fuel cycle are scheduled for FY 1983-1987. These projects will be implemented in close coordination with NMSS and SD. Note that this research plan does not include a plan for risk assessment applied to reprocessing plants or breeder reactors. A program for risk assessment of HLW geologic repositories and spent fuel is treated in Section 12.

Fuel cycle risk assessment research projects slated for FY 1983-1987 are:

1. Risk estimation surveys for all NRC-licensed radiologic activities;

2. Development of improved methods to estimate the likelihood of release scenarios in waste repository risk assessments;

3. Adaptation of methods developed for HLW repository risk assessment to low-level waste repository risk assessment;

4. Risk assessment of uranium mining, milling, and fuel fabrication;

5. Decontamination alternatives for reactors subject to core damage or meltdown accidents;

6. Risk assessment of small research and isotope production reactors; and

7. Interoffice Waste Modeling Group (IWMG). The IWMG is composed of personnel from NMSS, SD, and RES who exercise computerized waste repository risk assessment models. This continuing program accomplishes:

   a. Technology transfer - personnel in user offices gain experience in the use of regulatory tools developed by RES;

   b. Risk assessment perspectives - sample problems tackled by the IWMG include realistic waste isolation scenarios. The results obtained are of direct use in the development of regulatory standards and perspectives; and

   c. Peer review - agency personnel with diverse skills and expertise critique the risk assessment models.

9. HUMAN RELIABILITY RESEARCH

9.1 Regulatory Objective

The safety of a nuclear power plant is strongly influenced by the reliability of its operations and maintenance personnel. The plants must be designed and procedures chosen to reflect the limitations of human response capabilities. The diagnostic and corrective actions of operations personnel during abnormal events are major contributors to safety. Conversely, human errors in test and maintenance and in response to emergencies are among the dominant causes of failure in safety systems. Therefore, the agency responsibility to regulate the safety of nuclear power plants directly translates into a responsibility to regulate human influences on reactor safety.

The objectives of the human reliability research program are to provide the technical foundations with which to regulate:

1. The selection, training, and qualification of reactor operators and maintenance personnel;

2. The design of the man-machine interface and the role of the operator; and

3. The content and review of procedures for plant operation, maintenance, and emergency response.

9.2 Technical Capabilities Required

Extensive technical resources are required for human reliability assessment, for operator training and qualification, for designing the man-machine interface, and for setting standards of procedures for normal operation, test and maintenance, and emergencies. These resources include the results of qualitative and quantitative studies of human capability and reliability in a variety of operational tasks, ranging from simple to complex, under high and low stress, and involving both single individuals and teams. An extensive human reliability data base is needed to identify--at least qualitatively--the principal performance-shaping factors that influence the speed and accuracy of human behavior.

This data base also can be used in attempting to relate the causes of exceptionally good or bad reliability to procedures, personnel selection and training, or to the design of the man-machine interface. Where possible, quantitative (probabilistic) data and models should be developed.

The most pressing need for human reliability assessment is for those who directly control and maintain the plants, i.e., the operators and those who direct the maintenance crew. There is also a need for human reliability assessment for all levels of nuclear power plant management.

9.3 Status of Capability

Before the accident at TMI, the NRC had made little use of the scientific disciplines of psychology or human factors engineering. Some work in human reliability prediction had been done to support probabilistic risk assessment, but the agency had neither sought nor established a scientific basis for operator selection and training, for the design of the man-machine interface, or for the style and content of procedures.

Educational, group theory, cognitive, behaviorist, and many other branches of psychology have resources of relevance to human aspects of reactor safety. In addition to academic science, a relevant engineering discipline--human factors engineering--has grown up in recent years, primarily in the aerospace and defense industries. The NRC is just beginning to survey these resources and plan the research programs to support the scientific regulation of human influences on reactor safety.

The behavioral sciences and human factors engineering are far less mature and coherent than the physical sciences and engineering disciplines. They lack the central framework of well-established theory of physics, for example. Nevertheless, scientific psychology and human factors engineering have assessed a great deal of knowledge relevant to the study and regulation of human influences on reactor safety. A great deal of progress toward a scientific basis for NRC regulations can be made, therefore, by adapting the relevant knowledge and practices of psychology and human factors engineering to the context of nuclear safety.

9.4 Research Program Objectives

Basic research concerning the reliability of reactor operators and maintenance personnel will support standards development, improved safety analysis, risk assessment, and system reliability analysis. The program will be closely coordinated with the work of the Human Factors Safety Division of NRR, with I&E, and with the man-machine interface research in RES. The long-range objectives of the program are to establish relationships between human reliability and regulated practices in:

1. Personnel selection and training,
2. The human role in plant operations and maintenance,
3. The design of the man-machine interfaces, and
4. The content and style of procedures governing operations, maintenance, and emergency response.

An additional objective is to develop the methods and data base with which to include qualitative and quantitative models of operator action in probabilistic risk assessments.

9.5 Research Program Plan

9.5.1 Survey of Academic Psychology and Human Factors Engineering

A continuing survey will be made of the diverse literature and active specialities with scientific psychology and human factors engineering for their relevance to or utility in nuclear safety analysis or safety assurance.

9.5.2 Analysis of Human Reliability Data

Data on the reliability of reactor operators and maintenance personnel are assembled from (1) reports of actual experiences in the nuclear industry (LERs and so forth), (2) expert opinion and human reliability data from outside the nuclear industry (3) experiences on control room simulators in operator training, qualification, and requalification, and (4) controlled experiments in operator or maintenance personnel reliability.

The immediate objectives of the human reliability data analysis program are:

1. To obtain gross-error ra

2. To correlate error rates with performance-shaping factors that originate in the environment, with the design of the man-machine interface, with stress levels, with personnel selection and training, with task characteristics, with written procedures, and with task manning practices;

3. To select among, or make parametric fits to, predictive models of human reliability, including those intended to model single and multiple human errors and errors made by individuals as well as by teams.

9.5.3 Maintenance-Related Human Error Model

A model will be developed to predict the likelihood and nature of human errors in reactor plant maintenance and surveillance tasks. It will be validated by comparison with inplant data and will be used within RES for risk assessment and in man-machine interface research. It will be made available to NRR for use in improving criteria for (1) maintenance procedures, (2) plant personnel training, and (3) design of the man-machine interface with respect to test and maintenance.

9.5.4 Risk Reduction Through Improved Controls and Displays

The effectiveness of improvements in controls and displays at reducing the risks of severe accidents at nuclear power plants will be assessed as follows:

1. State-of-the-art criteria for controls and displays that are optimized for human reliability in emergencies will be developed for reactor control rooms.

2. The effectiveness of the improvements in hardware in reducing human error rates, reducing response times, and enhancing operations will be measured where feasible and estimated where measurement is impractical.

3. Sensitivity studies will be performed on reactor accident risk assessment models to translate the effect of altered controls on operator reliability into a predicted effect on risk.

9.5.5 Operator Decisionmaking Strategies

The best decision rules for reactor operators to follow under a variety of emergency conditions will be obtained, based on reactor accident analysis, pattern recognition, and human decision-theoretic research. Behavior of groups of control room personnel and personnel mixes will be studied. Inferences will be drawn for the design of the man-machine interface, for staffing of control rooms, for operator training, and for the criteria for emergency procedures.

9.5.6 Personnel Selection and Training Factors

This program will attempt to identify those performance-shaping factors within personnel selection and training that have the most influence on reactor risk, to develop recommended criteria for selection and training, and to validate those criteria through data analyses and experiments.

9.5.7 Human Error Prediction Model

A mathematical model capable of predicting the likelihood and character of operator errors in a variety of routine and emergency tasks will be developed and verified against simulator and reactor experience data. The model is intended for use in risk assessment and other forms of safety analysis and as a tool in several of the above-described human reliability research programs.

10. RELIABILITY ENGINEERING, OPERATIONS RESEARCH, AND DECISION THEORY

10.1 Regulatory Objective

The NRC and its predecessor, the AEC, have regulated nuclear technology uses to generally qualitative standards that rely on phrases such as "no undue risk," and "adequate protection of the public health and safety."

Many commentators on the effectiveness of the NRC and all the formal inquiries into the accident at TMI have cited a need for more objective definitions of the agency's safety goals. A rulemaking initiative for a reactor safety goal is underway.

A closely related objective is the development of regulatory decisionmaking techniques that are more objective, more reproducible, better documented, and more clearly tied to a finding of necessity or sufficiency for safety.

The burden on the NRC staff to implement current regulatory practices is exceeding the capacity of the present staff. This suggests another dimension to the need to review and revise regulatory decisionmaking. There is a need to improve the effectiveness with which staff resources are used.

Many commenters have noted that the form of nuclear safety requirements together with the economic constraints acting upon the regulated industry have in many ways combined to create disincentives for improvements in safety. There are great costs and great delay potentially awaiting those who would attempt to improve within the framework of present regulatory practices. Ways must be sought to meet safety objectives without stifling improvements.

There is a need, therefore, to reexamine the form and style of nuclear safety requirements to develop more objective standards for safety and more efficient ways of review to ascertain that standards are met and to leave the way open for the development of safety improvements.

10.2 Technical Capabilities Requi ed

The studies of variants on, or alternatives to, the current regulatory decisionmaking practices and criteria require a variety of technical resou ces. These include risk assessment (treated in Chapter 7) combined with methods of management science. Operations research, systems analysis, and decision theory. These disciplines, adapted to the context of nuclear regulation, are practical tools with which to survey the field of alternative formulations of regulatory goals, decisionmaking practices, and requirements upon licensees.

10.3 Status of Capabilities

A research program is currently underway to develop the groundwork for, and several hypothetical formulations of, a criterion of acceptable risk. The Commission has decided to take up the issue by rulemaking. Whether or not a rule is adopted, it is expected that some followup research will be needed to develop techniques with which to measure compliance with acceptable risk guidelines or standards. This is closely related to the program to adapt reactor risk assessment techniques for use as licensing tools.

Reliability engineering as it is practiced in the aerospace, defense, and electronics industries is a model of a way to organize a reliability assurance or quality assurance program that might fulfill the NRC objectives for reactor safety while (1) avoiding disincentives fo safety within the industry, (2) diminishing the burden on the NRC staff, and (3) making regulatory decisionmaking practices obj tive.

Aerospace reliability engineering is a management system fo organizing the assurance of sufficient reliability in complex engineered projects like aircraft and spacecraft. It is an organized way of delegating responsibility for reliability while preserving auditability, the contro of interface problems, and the organized digestion of the lessons of experience.

Reliability engineering practices differ from conventional nuclear safety analysis by not focusing primarily on a safety (or reliability) review of a final design. Rather, it is a process embedded in the conduct of conceptual design, detailed design, procurement, construction, startup or prototype testing, operations, and maintenance. Were such a system to be employed in the design, construction, and operation of nuclear power plants, the NRC could be much less prescriptive about designs, design criteria, hardware qualification, and technical specifications. Instead, the NRC would be prescriptive about the management and conduct of the design, construction, and operations process; about the controls at the interfaces among designers, builders, and operators; and about the industry-implemented experience feedback systems to verify that the licensees were doing a responsible job of ensuring sufficient safety. The early development of the quality assurance requirements of Appendix B to 10 CFR Part 50 was an early attempt to transfer these management practices to the nuclear reactor industry. Nevertheless, much more could be done.

Aerospace, weapons, and electronics industry practices in reliability engineering offer a very promising model of how nuclear safety could be achieved and verified without many of the shortcomings of the current regulatory process. There is a limited example (the FAA) of the adaptation of these practices to the regulatory arena. However, these reliability management practices have not been effectively adapted to the unique requirements and constraints of commercial nuclear reactor safety. A substantial research effort is required to determine the feasibility of further adaptation and to delineate procedures for nuclear safety regulation along the lines of reliability management systems.

10.4 Research Program Objectives

The objective of the decision-theoretic research projects is to develop decisionmaking techniques for use by NRR and I&E or for delegated use by licensees. These projects employ decision theory techniques and criteria based on acceptable risk criteria or other deterministic or probabilistic decision guidelines. These techniques are intended to document, streamline, and make objective some of the regulatory decisionmaking practices.

The objective of the reliability engineering research projects is to assess the applicability to nuclear safety regulation of reliability assurance techniques pioneered in other industries for the management of complex engineered projects and to make the necessary adaptations in the procedures to nuclear regulation.

The objectives of the studies of alternative formulations of regulatory requirements and of requirement implementation are intended to lay the groundwork for regulatory reform that strengthens the assurance of nuclear safety while improving the effectiveness of resource utilization by the NRC and the industry.

10.5 Research Program Plan

10.5.1 Development of Regulatory Decision Aids

The principal current activities are (1) the development and preliminary evaluation of acceptable risk criteria and (2) construction of formal decision-making procedures. Areas to be pursued in FY 1983-1987 are:

1. Development of measures of compliance with the acceptable-risk guidelines;

2. Development of handbooks on the risk-based optimization of surveillance test, maintenance, and allowable outage times for safety systems; and

3. Development of decision aids for NRR and I&E with which to select shutdown or prompt or delayed retrofit orders when safety deficiencies are discovered.

10.5.2 Reliability Management, Assurance, and Reliability Engineering Research

1. Survey aerospace, defense, electronics, and FAA practices in reliability engineering for relevance to nuclear safety assurance;

2. Identify the required adaptations to make reliability management techniques appropriate to regulating commercial reactor safety; and

3. Examine the repertoire of reliability engineering techniques for methods to fill several potential weak spots in current reactor safety regulatory practices, particularly,

   a. Interface problems among supplier, the nuclear steam system, the architect-engineer, and the owner;

   b. Detection and evaluation of inadequacies in startup and surveillance testing through which errors in design, construction, or inservice failures might otherwise escape detection; and

   c. Methods to complement existing quality assurance to better ensure the functional reliability of active components.

10.5.3 Exploration of Alternative Formulations of Reactor Safety Requirements

An array of options for combining deterministic regulatory requirements with risk-based or reliability management-based requirements will be proposed. These will be passed through a three-phase evaluation process: (1) they will be screened by risk assessment for their effectiveness at risk limitation, (2) they will be assessed for burden of implementation, and (3) they will be assessed for the feasibility of transition from current to proposed requirements and safety evaluations.

Those features of the options that look promising will be combined and the less promising options weeded out. The most promising prospects for regulatory reform will then be further elaborated and tested for feasibility by trial application to specific reactors.

These research programs will be closely coordinated with NRR, I&E, and SD.

11. CONSEQUENCE ANALYSIS

11.1 Regulatory Objectives

Consequence analysis provides the regulatory process with the tool to place reactor safety into the context of public health and safety. A realistic capability to predict the impact of releases of radioactive materials into the environment is required to develop regulations that will be consistent across design, siting, and emergency response requirements. Consequences of accidents give quantitative measures such as illness, death, latent cancer and gen- effects, property damage, and economic impacts from which comparisons made of various strategies of reactor safety involving tradeoffs or value/impact relationships. Additionally, consequence analysis provides the detailed analysis of potential accidents required in the environmental impact statements. Finally, consequence analysis is needed to assess the risk from reactors so that comparisons can be made to other societal risks.

11.2 Technical Capabilities Required

Consequence analysis brings all the elements of risk assessment together into a total risk perspective. To predict the consequences of radioactive releases to the environment requires the consolidation of the following disciplines into a consistent unified model: accident probabilities; quantities and types of radioisotopes released and physical phenomenology (timing and energy) associated with the release; plume rise; meteorology; dispersion; deposition; dosimetry; health effects; demography; emergency response; economics; property damage; and statistical treatments. Technical expertise is required for all disciplines to develop a model that uses the most up-to-date information in as consistent and realistic a manner as possible. The relationships and sensitivities between the data and models is extremely important in the effort to realistically predict the distribution of consequences from a spectrum of potential accidents. Additionally, the results generated by such models must always be brought back into a proper perspective since the uncertainties associated with the results are very large, and the models must be tailored to specific applications. Training of qualified consequence analysts requires a knowledge of the full range of scientific disciplines from nuclear engineering to economics and demands an overview of the entire risk assessment methodology to adequately structure a comparison. The current emphasis in research is on consequence analysis associated with atmospheric releases. Further work is planned on liquid path releases as well.

11.3 Status of Capability

Consequence analysis research focuses on the Calculation of Reactor Accident Consequence, CRAC, computer code. CRAC was developed for the Reactor Safety Study (WASH-1400) and incorporates all the models and capabilities described in Appendix VI of this study. Since the completion of the study, a research program has been in place to investigate the sensitivities and relative importances of the models and data assumptions. CRAC has the basic ability to model a specific site and, with the proper accident scenario input, to generate risk frequency statistics. However, significant uncertainties exist in its ability to adequately perform this function.

Liquid pathway analysis has been done for postulated core melt accident releases from a whole range of reactor siting situations and reported in the draft Sandia Liquid Pathway Study. This work indicates that the atmospheric pathway clearly dominates the risk. Nevertheless, the risks of the liquid pathway are quite uncertain and are not well enough understood to define adequate interdiction measures for emergency situations.

11.4 Research Program Objectives

The principal objective of consequence analysis is to develop a realistic capability to predict the distribution of consequences from a release of radioactive materials. Research objectives are to improve the modeling capabilities by establishing priorities for related research areas such as meteorology and dosimetry and health effects. The following needs have been identified as requiring additional capabilities:

1. Site-specific capabilities;
2. Sensitivity and uncertainty analysis;

3. Liquid pathway assessments;
4. Emergency planning and protective measures;
5. Licensing support; and
6. Risk consistency.

Research into these areas will improve our understanding of the impact of radioactive releases.

11.5 Research Program Plan

11.5.1 Development of Site-Specific Consequence Model

The principal activity in this area is the development of improved methods to predict the offsite consequences of reactor accidents at specific locations. Subjects having the greatest uncertainties and, therefore, requiring the most attention include:

1. Trajectory and Improved Dispersion Modeling. Development and inclusion of improved meteorological models will primarily be based on continuing research in the field and concurrent sensitivity studies. Elements of this program to be considered [illegible] 1983 include long-range diffusion techniques, topographical and climatological considerations, and vertical diffusion sensitivities. Elements of research scheduled for FY 1984 include improved meteorological data acquisition, optimized weather sampling, and studies to evaluate modeling techniques that can make further use of new data. In FY 1985, studies will be undertaken to evaluate the influence of continuous versus discrete weather data on the diffusion distributions. For FY 1986-1987, a review of the diffusion modeling techniques and incorporation of new improved models will be performed. This entire effort will be closely coordinated with the meteorological efforts of the Site Safety Research Branch, DRSR, and will be advised by an expert meteorological panel on a continuous basis.

2. Deposition and depletion considerations will play a large part in choosing and developing an appropriate dispersion modeling approach. With the

acquisition of better meteorological data in FY 1985, additional modeling approaches could be developed.

3. Dosimetry and health effects. A continuous effort is underway to develop, and make publicly available, the most up-to-date dosimetric data. This effort will continue throughout FY 1983-1987. Evaluation of health effects research will also be an integral part of Consequences Analysis in order to revise the health effects models. This effort is closely coordinated with the agency's health physics staff. In addition, advice of the nation's leading health physics experts is periodically obtained to formulate appropriate models. New models will be developed in FY 1986.

4. Property damage and economic modeling. In the FY 1983-1985 period, research will be performed to develop modeling approaches and techniques for decontamination and interdictive measures. Studies to evaluate cost/benefit tradeoffs will be performed in FY 1986.

5. Data acquisition. In addition to the above tasks, new data will be acquired from a number of different sources including the Census Bureau for demographics, the interagency Digital Information Display System for land use, and Coast and Geodetic Survey for topographic data. Improvements in data will be scheduled for FY 1983.

11.5.2 Sensitivity and Uncertainty Analysis

Research is required to evaluate the sensitivity and uncertainty of the consequence model improvements. The following tasks focus on this very important part of the consequence analysis effort:

1. Benchmark studies and comparison with other modeling approaches (FY 1983-1987);
2. Parametric studies (FY 1983);
3. Response surface modeling and uncertainty analysis techniques (FY 1984-1985);
4. New siting impact studies (FY 1986-1987);
5. Emergency response impact studies (FY 1983-1984); and
6. Source term impact studies (FY 1983-1984).

11.5.3 Liquid Pathway Consequence Assessment

Modeling of exposure pathways from ground water and other liquid pathways is planned to develop criteria for interdictive measures as proposed in the Siting Policy Task Force Report, NUREG-0625. This program should be completed in FY 1983. This program should be completed in FY 1983.

11.5.4 Emergency Protective Measures Research

In conjunction with other Federal agencies (FEMA, EPA, and BRH), this program will review past experiences in emergency response and formulate new modeling approaches for evaluating the impact of various emergency strategies. Included in this effort is research to evaluate information on respiratory protective measures and protective strategies, devise modeling approaches for risk assessment purposes, and make applicable recommendations. This program will continue throughout the 5-year period and its results integrated directly into the regulatory process.

11.5.5 Plant-Specific Event Tree/Consequence Emergency Procedures Guides

The objective of this program is to prepare detailed guides for emergency response purposes describing the potential dominant accident sequences, event trees, and consequence estimation in a consistent and integrated manner. This effort has potential benefits for plant operators and offsite incidence response personnel. It is a natural follow-on to the IREP/NREP studies and the Severe Accident Sequence Analyses and will develop usable tools for accident response and diagnosis. The program will begin in FY 1983 and continue throughout the 5 years.

11.5.6 Risk Consistency Studies

In our efforts to understand the consequences associated with releases of radioactivity into the environment, it will be necessary to evaluate other techniques and approaches that estimate consequences of accidents from other hazards to ascertain if there are meaningful comparisons to be made. This program is structured to evaluate other risk studies and prepare assessments of the appropriateness of comparisons with other modeling techniques or data assumptions. The plan for this effort is to initiate the evaluation of a specific hazard evaluation (such as liquid natural gas, toxic chemical substances, or coal production) starting in FY 1984 and going through 1987.

PART C. SAFEGUARDS, FUEL CYCLE AND ENVIRONMENTAL RESEARCH

12. WASTE MANAGEMENT RESEARCH

The societal risks and environmental impacts associated with management of radioactive wastes are judged by the NRC staff to be greater for uranium recovery wastes, less for low-level waste, and least for high-level waste. On the other hand, the uncertainties in NRC assessments of these impacts and risks are greatest for high-level waste disposal and least for uranium recovery. This suggests that the emphasis given in public debate to high-level waste disposal is related more to the level of uncertainty than to the level of risk. This is partly because the large uncertainties make it difficult to demonstrate the validity of the expert judgments concerning relative risk.

The NRC waste management research program is aimed largely at reducing the uncertainties involved in assessing risk or the safety performance of waste management facilities. This requires improved information on the characteristics of the wastes and on the phenomena that influence the capability for isolation of the wastes. The specific research needs for high-level and low-level waste management and uranium recovery depend on the regulatory approach and the status of NRC technical capabilities in each of these areas. These factors are discussed below, with respect to each of the three research programs.

12.1 High-Level Waste Management

12.1.1 Regulatory Objective

The regulatory objectives of the high-level waste management program are:

1. To provide and implement the framework of regulations and criteria that will ensure that the disposal methods developed for all types of radioactive waste are consistent with the achievement of the goal of safe, long-term waste disposal;

2. To ensure that waste forms and containers used for high-level radioactive wastes permit safe handling and transportation of the wastes (according

to the requirements of 10 CFR Part 71 and 49 CFR Part 173), are compatible with requirements for safe and effective high-level waste disposal facility operation, and will contain the wastes long enough to ensure that potential offsite releases of radioactive materials are within EPA standards;

3. To ensure that the site and its environs have characteristics that permit compliance with health and safety regulations (according to 10 CFR Parts 20 and 60) and with environmental quality requirements (according to NRC, EPA, and State regulations) during construction and operation of a high-level waste repository, as well as after closure; and

4. To ensure that high-level waste repositories and ancillary facilities are designed, constructed, and operated to provide compliance with health and safety regulations (according to 10 CFR Parts 20 and 60) and environmental quality requirements (according to NRC, EPA, and State regulations).

12.1.2 Technical Capabilities Required

The NRC must have the capability to evaluate the performance of waste forms and containers, of engineered underground facilities, and of the surrounding geologic media relative to the long-term isolation of high-level radioactive wastes. This requires an understanding of the physical phenomena that control the rate of release and transport of radionuclides and the relationship between those phenomena and the design and site features of the waste repository. Included also is the ability to identify and assess the risks (probability and consequences) associated with disruptive natural phenomena or man-caused events. These capabilities are necessary for establishing NRC regulatory standards and guides and for evaluating generic and specific sites and engineering designs.

The NRC must also be able to assess safety performance during routine operation, waste retrieval, and following repository closure and to assess the consequences of potential accidents. Compliance with the NEPA requires a capability to assess the health and environmental impacts that can result from facility construction, operation, and closure. It also requires comparative assessments of alternative sites and alternative disposal methods.

In all these areas, a capability must exist to assess the uncertainties in the predictions in order that conservatisms to compensate for the uncertainties may be included in performance or design standards.

12.1.3 Status of Capabilities

The quantitative effects on the stability of waste forms resulting from variations in the composition and processing methods used for waste solidification are known for only a limited number of waste forms and for periods of only a few months or years. The understanding of material degradation processes and the mechanisms involved is not sufficient to predict the very long-term behavior of high-level wastes. Long-term predictions from existing models have large uncertainties that are at present difficult to define.

Although the current technology for handling spent-fuel assemblies and other radioactive materials and current safety-assessment methods (such as shielding and accident analysis) are generally applicable to high-level waste package handling, the effect on safety of operating deep under ground and of handling solidified wastes with higher specific fission product loading has not yet been evaluated.

There is a major deficiency in the capability to define the degree of uncertainty in very long-term predictions of site characteristics and repository integrity needed for high-level waste isolation. There are large uncertainties about the sufficiency or validity of:

1. Indirect methods for identifying and characterizing rock and geologic structures and their physical, chemical, and hydrological properties;

2. Methods for evaluating rock mass seals, including borehole plugging, and extrapolating assessments of their integrity over very long periods;

3. Effective methods for predicting ground-water flow and radionuclide migration during very long periods and for predicting potential effects caused by the construction or operation of a repository;

4. Methods for long-term predictions of geomorphic and climatic processes and other events and their potential impact on a repository;

5. Thermal effects on repository performance; and

6. Monitoring instruments and methods with predictable long-term reliability.

The applicability of present methods of forecasting the impact of repository facilities on the local economy, land-use patterns, transportation, and community structure and institutions has not been evaluated.

12.1.4 Research Program Objectives

The objective of this research program is to provide a better understanding of natural phenomena and processes and to evaluate state-of-the-art analytical and engineering methods that are used in site characterization and repository design, construction, operation, and closure.

Specific objectives are to:

1. Identify failure mechanisms that affect long-term waste isolation capability;

2. Identify technical requirements that may be needed to mitigate the consequences of accidental or unplanned movement of radionuclides; and

3. Define the uncertainties or confidence levels in the data, analytical methods, and predictions for each of the above areas of concern.

12.1.5 Research Program Plan

Ongoing research to understand the long-term radionuclide release rates from waste forms and containers for high-level wastes will be continued. The project will develop the understanding necessary to assess and predict the longevity of waste package materials under repository conditions and to quantify the separate effects of important factors (such as temperature and radiation) on the degradation of the waste form and canister materials. Research on waste forms such as glasses, supercalcines, and iron-alloy containers will be completed during FY 1984. Studies of the more advanced waste forms such as SYNROC and coated particles and container materials such as titanium alloy will extend through FY 1986.

Research that assesses the effectiveness of geophysical techniques for identifying geologic structures and hydrologic and stratigraphic boundaries will continue through FY 1983. Related research that will continue beyond FY 1986 will include assessment of the state of the art and limitations and uncertainties of methods to characterize faults and joints, including their genesis, surface texture, geometric distribution, nature of infilling, and interconnection of fractures. This research will also assess the importance of knowing the age of fractures and the capability of geophysical methods to characterize fracture systems. The capability for in situ measurements of large-scale physical and hydrologic properties of rocks will also be investigated.

Research will continue through FY 1986 on the measurement of ground-water flow relative to the potential migration of radionuclides. This research will include the determination and field testing of laws governing movement through fractured rocks. The effects of convection, diffusion, and dispersion on radionuclide transport in ground water, including changes induced by heat, will be evaluated. Nonradioactive tracers that can be used in hydrologic transport tests and for possible placement in a repository for monitoring purposes will be identified. The assessment of ground-water dating techniques, an important tool for establishing past rates and for predicting future rates of ground water movement, will be completed in FY 1982.

Research started in FY 1981 on the effects of heat on the chemical properties of rock and on geochemical processes will continue through FY 1984. Research planned to compare the importance of slow processes such as solid-state diffusion of radionuclides into minerals with the more rapid sorption processes involved in radionuclide migration will start in FY 1982 and continue through FY 1985. Current research is assessing the validity of existing distribution coefficients for use in geochemical models by comparing distribution coefficients determined from laboratory batch and column methods and from field measurements. This work, which includes assessment of predictive models, will continue through FY 1986.

Ongoing engineering geologic research evaluates the effectiveness of methods for assessing rock mechanics and the chemical properties of rock with a minimum disruption of the rock. Emphasis will be given to the limitations and uncertainties in extrapolating data to predict repository performance. Beginning in FY 1982, this research will also assess the effects of heat on rock properties such as strength, rigidity, integrity, chemical reactivity, and adherence to sealing materials to assess the stability of underground openings and to assess uncertainties introduced because of the heat.

The ongoing assessment of the migration of naturally occurring radionuclides from ore bodies over very long time periods (which is a natural analog to radionuclide migration from a high-level waste repository) will continue through FY 1987.

The capability for predicting long-term climatic, geologic, and hydrologic changes with a potential for causing significant impact on the integrity of a repository or its environs will be evaluated to identify the specific needs for additional research.

In FY 1983, research will also be initiated to assess the probable effects of a repository on the economic and social structure of the surrounding local area. This work will adapt methods of forecasting the impacts on employment, housing, land-use patterns, public revenues and expenditures, transportation, and community structure and institutions. The effort will be continued through FY 1986, with increasing emphasis on ways to mitigate adverse impacts.

Evaluations of methods that may be used to predict the thermomechanical response of the rock mass include field tests of their validity. These tests will provide the technical bases for thermal-loading criteria for high-level waste repository designs that will keep uncertainties within acceptable limits. This effort, continuing through FY 1987, will follow the research on the thermal conductivity of rock and will be directed toward evaluating specific proposed repository designs.

Ongoing research determines the solubility of different chemical species of radionuclides at temperatures and pressures expected in a repository and the potential for interaction of the waste package with water, repository rock, and the engineered structure. This will lead to the start of research, in FY 1983, to assess the effectiveness of engineering the geochemical systems to immobilize radionuclides or retard their migration.

Research to assess the effectiveness of materials and methods that may be used to backfill and seal shafts and tunnels for a high-level waste repository will begin in FY 1983 and will be coordinated with ongoing research on borehole sealing.

Ongoing research on engineered barriers to radionuclide migration within a high-level waste repository will continue beyond FY 1986; it will be coordinated with the hydrologic and geochemical studies for site characterization.

Ongoing research will continue through FY 1986 to identify critical parameters that need to be monitored and to evaluate the effectiveness of monitoring instruments and methods during construction and operation and following closure of a repository; it will include evaluations of the long-term reliability and accuracy of monitoring instruments, temporal and spatial distribution of sampling points, and methods for data reduction.

An ongoing assessment of the types and the expected effectiveness of ways to mitigate the consequences of accidental releases of radionuclides will continue through FY 1987.

A program of closely related projects to develop methods for evaluating the performance of repositories in different media and the risks associated with phenomena that could initiate or accelerate releases of radionuclides to the biosphere will continue through FY 1987. The results will be coordinated with other projects in the high-level waste program to reduce the uncertainties in performance prediction and risk assessment.

12.2 Low-Level Waste Management

12.2.1 Regulatory Objective

The regulatory objectives of the low-level waste management program are:

1. To ensure that low-level radioactive waste forms and their containers permit safe handling and transportation (in accordance with the requirements of 10 CFR Part 71 and 49 CFR Part 173), that they are compatible with requirements for effective disposal facility operation, and that they retain the wastes for sufficient time to ensure control of potential offsite releases of radioactive and other toxic materials;

2. To ensure that the disposal site and its environs provide a natural barrier to the migration of radioactive and toxic materials and have characteristics that permit compliance with health and safety requirements (10 CFR Part 61) and environmental quality requirements (NRC, EPA, and State regulations) during construction and operation of a facility and after it is closed; and

3. To ensure that disposal facilities are designed, constructed, and operated to provide compliance with health and safety requirements (10 CFR Part 61) and environmental quality requirements (NRC, EPA, and State).

12.2.2 Technical Capabilities Required

The NRC must have the capability to predict and assess the properties and performance of waste forms and containers before, during, and after they are placed in a waste disposal facility. There is a trend toward requiring longer-term integrity of low-level waste forms and containers than is currently practiced. The establishment of such requirements would need to be supported by a capability to predict long-term performance.

The NRC must also be able to assess the adequacy of the site characterization and of the information and methods used to predict site performance. This should include predicting the effects of facility construction, operation, and closure on the site's ability to function as a barrier to waste migration and assessing site performance during facility operation and after closure. Capabilities are needed to assess the potential for natural phenomena, man-caused events, and human intrusion, as well as their effects on the long-term performance of the site and facility.

The NRC must be able to ensure compliance with health, safety, and environmental requirements with regard to the design of the facility and plans for construction, operation, and closure. The NRC must also be able to predict facility performance during normal operations and under accident conditions (including conditions influenced by both natural phenomena and offsite, man-caused events such as explosions) that could damage the facility and cause releases of radioactive or toxic materials from the site.

The NRC must have the capabilities to evaluate the adequacy of proposed radiological and environmental monitoring programs and to assess their effectiveness after they are implemented.

Current interpretations of the NEPA require the capability to assess the health and environmental impacts that can result from facility construction, operation, and closure. In addition, the NEPA requires comparative assessments of alternative sites and alternative disposal methods.

Common to all these capabilities is the need to be able to assess the uncertainties associated with the required predictions and assessments and to introduce conservatisms into performance or design standards to compensate for the uncertainties.

12.2.3 Status of Capabilities

Based on existing data, empirical models can be reliably used to predict the behavior of most candidate materials for waste forms and containers for periods up to about 100 years. Extended predictions of long-term durability of waste forms and containers to periods greater than 100 years as stated in the proposed regulation (10 CFR Part 61), are based mainly on scientific and engineering judgment and the uncertainties associated with these longer-term predictions are undefined.

The ability to predict the environmental impacts of construction and operation of shallow-land burial sites appears to be adequate. However, the prediction of the long-term capability of the site to prevent offsite migration of radioactive and toxic materials is subject to increasingly large uncertainties as the time period to be predicted is extended. These uncertainties could be significantly reduced if more systematic and precise information is obtained for use with existing or improved models.

Methods used by the NRC for assessing potential occurrences and effects of natural phenomena and man-caused events on shallow-land burial sites are generally derived from existing guides for the evaluation of reactor sites. However, the adequacy of the existing methods and information for evaluating sites for low-level waste disposal facilities other than shallow-land burial has not been determined.

The NRC has defined an urgent need for improved design and operational procedures for shallow-land burial facilities at sites with medium to high precipitation. The need to establish standards is related primarily to methods for waste emplacement and compaction or to other methods to prevent the subsequent creation of voids that lead to trench-cap collapse and infiltration of water. Comparative assessments of trench-cap designs are needed to evaluate their long-term stability.

Improvements are needed in monitoring methods, including sampling and analysis methods, instrument calibration, use of tracers, and statistically valid monitoring networks to ensure early warning of significant variations from planned or predicted performance during construction and operation following site closure.

No detailed evaluation of alternatives to shallow-land burial is available. Nor has there been any determination of the applicability and adequacy of the available information and methods for assessing alternatives to shallow-land burial (such as mined cavities).

Information on the mechanisms of water movement and the distribution coefficients of different chemical species of radionuclides in the unsaturated zone (i.e., soils above the water table) and in a variety of soil types is not adequate to provide realistic predictions. Distribution coefficients have not been measured in the field at specific sites for comparison with data obtained from laboratory tests. It is becoming increasingly clear that the information on the sorption and geochemical properties of soil relative to radionuclide migration produced by conventional laboratory tests is not sufficiently realistic to be useful in models for predicting radionuclide migration at shallow-land burial sites.

Explicit risk assessment methods for low-level waste disposal facilities objectives are not available.

12.2.4 Research Program Objectives

The objectives of the research program are to provide improved understanding of phenomena, analytical methods, and data to improve the NRC capability to predict low-level waste isolation performance and to provide a better technical basis for regulatory standards and risk assessment.

12.2.5 Research Program Plan

Solidified wastes that result from alternative techniques for processing wastes from normal operations and accidents are being tested and evaluated. It is planned to continue this research through FY 1986. As the program progresses, evaluation of the stability of waste containers (e.g., by corrosion testing) will be given greater emphasis, and the research on the characteristics of waste forms will include tests of full-scale reactor waste forms and wastes from decommissioning, industrial, medical and other institutional sources. Waste forms resulting from volume reduction methods (such as incineration) will be tested and evaluated as materials become available from DOE and from commercial volume reduction development programs. An assessment of the feasibility and cost effectiveness of segregation of wastes will begin in FY 1982. Evaluation of the nonradioactive toxic or hazardous components of wastes and of methods for nondestructive testing of free-standing water in waste packages will continue through FY 1983.

Field tests at existing commercial shallow-land burial facilities will be conducted through FY 1984 in coordination with supporting laboratory tests and experiments. The research tests will include assessments of erosion rates, characterization of degraded wastes in trenches, hydrologic measurements in the unsaturated zone, comparison of distribution coefficients for different chemical species of radionuclides in different geologic media, nonradioactive components of wastes (such as chelating agents that may influence radionuclide migration), assessment of the effectiveness of monitoring methods, and the effects of vegetation on radionuclide containment or migration. The data obtained from the field work will be used to test existing and proposed predictive models and to aid in identifying important characteristics of sites and their environs as they relate to the control of radionuclide migration and the integrity of the facility.

Ongoing research to test the effectiveness of improved trench-cap designs by using nonradioactive, nontoxic tags to trace water movement will be essentially completed in FY 1983. However, the results of these early projects may identify a need for limited, longer time monitoring of experimental trenches. Tests of the effectiveness of proposed engineered barriers in preventing the migration of radionuclides and other toxic materials are planned to start in FY 1984.

By FY 1983, an assessment of geologic alternatives to shallow-land burial will identify those that are feasible and cost effective. In FY 1984, field tests will begin that will provide specific information for each disposal method and will provide a basis for a comparison of the alternatives. This research will consider site suitability, facility design and operation, monitoring, and facility closure for each of the alternative methods. During FY 1984-1985, the emphasis will change from research on shallow-land burial to research on burial in deeper geologic media. During the later part of the planning period, a feasibility and cost-effectiveness study for engineered storage will be initiated. The results of research on site characteristics for the high-level waste repository are expected to make a significant contribution to research on mined cavities for the disposal of low-level waste.

The research on shallow-land burial and geologic alternatives will include research to assess potential improvements in site and facility monitoring, to assess the effectiveness of designs and procedures for closing sites, and to identify and assess ways to mitigate consequences in the event of accidental releases of radionuclides.

Research on methods to assess risks to the environment and the public health and safety from low-level waste facilities will be completed in FY 1985.

12.3 Uranium Recovery

12.3.1 Regulatory Objective

The regulatory objective of the uranium recovery program is to ensure that uranium recovery operations and their associated waste management programs are conducted in full compliance with effective standards, guidelines, and regulations for the protection of health, safety, and the environment in accordance with the Atomic Energy Act of 1954, the Energy Reorganization Act of 1974, the National Environmental Policy Act (NEPA) of 1969, the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), and applicable State statutes and requirements.

12.3.2 Technical Capabilities Required

The evaluation of applicant proposals and the development of appropriate regulatory guidance for uranium recovery operations require complex, multidisciplinary analyses addressing a broad range of environmental and safety concerns. The NRC staff must be able to:

1. Assess the completeness and validity of site characterization information provided by an applicant (including evaluation of methods of obtaining, reducing, and analyzing raw data);

2. Assess the performance of waste disposal facilities, including operations and final reclamation plans, with respect to the degree, reliability, and permanancy of isolation provided, in terms of ground-water protection, long-term surface erosion control, and resistance to unusual disruptive forces, both natural and manmade;

3. Assess the nature, magnitude, and probability of radiological and nonradiological environmental and health and safety impacts of facility construction, operation, and decommissioning, as well as those impacts that result from land reclamation or waste stabilization of waste disposal operations; and

4. Assess the short- and long-term radiological and nonradiological impacts of alternatives to the proposed action, as well as their costs and benefits.

Common to all these required capabilities is a need to understand the level and importance of the inherent uncertainties, so that design and performance standards can be determined that provide reasonable degrees of confidence that adequate protection will be provided, particularly with regard to the long-term performance of the waste isolation programs.

12.3.3 Status of Capabilities

Through the preparation of site-specific Environmental Impact Statements (EISs) in support of numerous individual licensing actions, and through preparation of the Generic EIS (GEIS) on uranium milling, the regulatory staff has gained considerable knowledge and experience. The final GEIS, published in late FY 1980, is the primary basis for the design and performance objectives now being established as tailings-management regulations. Research efforts completed to date have focused primarily on quantifying and characterizing radioactive releases and their concentrations in the environment to support the GEIS and licensing tasks. The work was largely directed at identifying and quantifying the environmental impacts associated with existing and past uranium recovery operations and practices, and the results were used in the development of the present staff capability for radiological impact assessment.

Through this work and rulemaking efforts toward defining minimum criteria for final waste disposal and reclamation, many areas were identified in which additional information is necessary to provide a technical basis for making regulatory and licensing decisions. Uncertainties exist in several key areas. For example, no tailings disposal site has been reclaimed by application of the thick earthen covers specified in the NRC regulations. Instead, such issues as practicability, effectiveness, reliability, and vulnerability to natural or manmade disturbance are being addressed by staff judgments made on the basis of limited available information.

12.3.4 Research Program Objectives

The objective of the uranium recovery research program is to improve the NRC capability to evaluate applicant proposals and to develop regulatory guidance for controlling uranium recovery facilities so as to prevent or mitigate deleterious health and environmental impacts. Specific research objectives are to test and evaluate methods to:

1. Assess the effectiveness, benefits, and costs of various engineering and milling processes and disposal alternatives for reducing tailings impoundment seepage of toxic and radioactive materials and ground-water degradation resulting from deep mine or pit disposal;

2. Assess subsurface physical conditions with respect to the potential for short- and long-term migration of toxic and radioactive constituents of seepage and the associated environmental impacts, including their magnitude, distribution, and duration;

3. Assess the methods, practices, monitoring, and analytical models necessary to define and control in situ mining solution movement, and assess available restoration techniques and abilities;

4. Test and improve monitoring practices, techniques, strategies, and equipment for assessing release of radioactive and toxic materials from facility components, the subsequent environmental transport of the materials, and their impacts on the environment and on the health and safety of the public;

5. Assess the reliability, durability, and cost effectiveness of interim stabilization techniques for reducing particulate suspension and radon gas exhalation from exposed tailings surfaces;

6. Assess the effectiveness and long-term reliability of clay caps and thick earthen covers or other materials for reducing radon gas releases;

7. Assess the practicability, effectiveness, and reliability of self-sustaining vegetative covers and rock covers to control surface erosion of tailings;

8. Assess methods for predicting and evaluating the long-term integrity of reclaimed tailings disposal sites, as opposed to the stability of reclaimed tailings areas alone; and

9. Assess the instruments, techniques, and procedures for verifying acceptability of site decontamination, decommissioning, and reclamation before license termination.

12.3.5 Research Program Plan

Research on tailings neutralization and other alternatives for immobilizing toxic materials in tailings will continue through FY 1987; it will include an assessment of dewatering underdrains, nitric acid and other alternative leaching processes and segregation of sands and slimes from the tailings piles.

Research that examines subsurface disposal of uranium tailings and the potential for and the consequences of contaminating the ground water will continue through FY 1987. Results from an ongoing project that is assessing leachate movement from ponded uranium mill tailings are critical to this work. The project includes examination of adsorption capacities, natural dispersion and diffusion, and transport rates. It is directed toward developing an accurate and reliable predictive analytical model for waterflow and transport of radionuclides in the partially saturated zone.

Research on methods for predicting and verifying the subsurface distribution of solutions used for in situ mining will continue through FY 1986; associated work will assess methods for indirect measurement of subsurface conditions by use of the measurement of electrical resistance and by radar.

Assessment and improvement of methods, equipment, and instruments for measuring and assessing releases of radioactive and toxic materials from various facility components and for verifying the acceptability of site and structure decontamination will continue through FY 1986.

Research on interim stabilization of tailings will continue through FY 1987. This research includes assessment of water sprays and chemical stabilizers and application of oil, kerosene, and shallow earth covers. In addition to interim stabilization, ongoing research on the effectiveness and reliability of rock covers for controlling surface erosion by wind and water over long time periods will continue through FY 1987. In FY 1982 research will be initiated on self-sustaining vegetative covers for long-term surface erosion control and will continue at least through FY 1987.

Ongoing research on the effectiveness of earthen covers to reduce radon releases will continue through FY 1987 and will include assessment of the effects of thickness, order of placement of materials in the covers, their moisture retention capabilities, and defects in cover materials due to drying, subsidence, and penetration by plant roots and burrowing animals.

Research will continue on methods for predicting the long-term effects of geomorphic processes on disposal sites. In FY 1983 research will begin to assess methods for predicting the potential for, and effects of, extreme natural phenomena (including prolonged heavy rainfall, earthquakes, and intense storm activity) on reclaimed tailings areas and disposal sites.

13. SAFEGUARDS RESEARCH

A problematic issue in safeguards regulation is how to introduce a risk perspective when so many of the human factors involved are not susceptible to meaningful assessment. The NRC safeguards research program has been addressing this problem by developing a systematic framework for considering the significant risk factors and identifying those that might be quantitatively evaluated. Methods are being developed for conditional evaluation of safeguards effectiveness based on assumptions concerning adversary attributes.

A method for assessing safeguards adequacy should be capable of demonstrating whether a safeguards system satisfactorily meets the NRC safeguards objectives. In the case of a licensing review, this means assessing the level of protection afforded against a defined range of threats. This assessment requires a definition of the threats in terms that are specific enough to permit assessment, a relatively detailed knowledge of the safeguards system and how it operates, and some means of predicting the outcome when the safeguards system encounters the postulated threats.

These same steps are performed whether the entire process is carried out inside the head of an expert or is reduced to a code and carried out in a computer. These alternatives are, in fact, representative of two schools of thought on this matter. One school argues that the assessment of security is inherently highly intuitive and cannot be modeled or coded without losing the subtlety and flexibility of the intuitive approach. The other school argues that the process is largely systematic and that by modeling the systematic part, the documentation of the expert methods makes them more widely available and more subject to discipline and rigor, thereby contributing to the achievement of uniformity and excellence.

The value of intuition based on experience can hardly be overstated, especially in complex analyses. Nevertheless, the need for NRC to communicate the basis for its regulatory decisions to the licensees and the public requires a balance between a flexible, intuitive approach and systematic documentation of the information and reasoning used to make licensing decisions.

13.1 Regulatory Objective

The objective of the safeguards regulatory program is to ensure that NRC licensees protect public health and safety and national security against theft or sabotage through the development, implementation, and maintenance of effective safeguards programs at nuclear reactors and fuel cycle facilities and during the storage and shipment of special nuclear material (SNM) and spent fuel.

13.2 Technical Capabilities Required

To achieve this regulatory objective, NRC must be able to:

1. Establish a technical basis to aid policy decisions concerning the nature of safeguards requirements;

2. Define design basis threats in terms that enhance uniformity and effectiveness of safeguards design and assessment;

3. Evaluate the level of safeguards protection against any threat within the design basis definition; and

4. Establish acceptance criteria for safeguards system performance based on a determination of the level of protection deemed adequate.

These capabilities help provide answers to the questions:

1. What is the threat?
2. How good are existing safeguards?
3. How good do safeguards need to be?

13.3 Status of Capabilities

The current design basis threat is stated in broad terms that cover many possible variations. Studies to characterize nuclear threats have been based on analogies to criminal and terrorist attacks on nonnuclear targets. The results of these studies have provided insights into the general kinds of strategies and capabilities that might be represented in attempts at nuclear theft or sabotage. There is, at present, no guidance on how to use such information in the design or evaluation of nuclear safeguards systems to meet NRC performance standards. Such guidance would help to ensure uniform consideration of threats and systematic evaluation of the assumptions and considerations used in the design or evaluation.

Evaluation of the effectiveness of a licensee's physical protection system is based on documented procedures that have evolved from field experience and the Comprehensive Evaluation Program (CEP), which was a detailed, onsite review of safeguards performance and vulnerabilities at each major licensed fuel cycle facility. A similar evaluation program (vulnerability assessments) for reactors is being developed. The "Fixed Site Physical Protection Upgrade Rule Guidance Compendium" (NUREG-0669) will be used to provide guidance to licensees who, in turn, must provide certain specified data on their facilities that will allow assessment of licensee performance and compliance with the physical protection upgrade rule. Both the above-mentioned methods for the evaluation of the effectiveness of a licensee's safeguards system continue to rely to a considerable extent on implicit professional judgments. Guidance that would take into account the effect of possible adversary strategy on the performance of safeguards components and subsystems is lacking. In the future, these methods may be supplemented by some of the quantitative evaluation techniques being developed under the NRC research program.[1,2] The methods being developed provide a means for assessing safeguards system response as a function of adversary characteristics and strategies. In addition, they provide technical aids for evaluation that have the advantage of offering a systematic and reproducible

[1]"Safeguards Network Analysis Procedure (SNAP)," Sandia National Laboratories and Pritsker Associates, NUREG/CR-1245, draft, December 1980. "Safeguards Automated Facility Evaluation (SAFE)," Sandia National Laboratories, Vols. 1, 2, and 3, NUREG/CR1246, draft, November 1980.

[2]"Structured Assessment Approach (SAA)," Lawrence Livermore National Laboratory, NUREG/CR1233, draft, October 1979. "Safeguard Vulnerability Analysis Program (SVAP)," Lawrence Livermore National Laboratory, NUREG/CR1169, April 1980.

analysis of licensee safeguards. These evaluation methods have received considerable interest outside the NRC. However, these methods have not yet been refined through extended field application to licensed facilities nor have they been sufficiently validated. (This lack of validation is primarily due to the many--and uncertain--human factors involved in safeguards adversary interactions.)

The acceptability criteria in the guidance compendium for the physical protection upgrade rule were derived from broad, performance-oriented goals by establishing subjectively derived criteria for the characteristics of physical protection subsystems and components. Similar methods are being used in upgrading the material control and accountability (MC&A) rule, but to a lesser extent. The criteria are intended to be independent of threats; therefore, they do not reflect the way in which the effectiveness of safeguards will vary according to different threats

The current ability to evaluate societal risk arising from malicious acts has been very limited. It has included reviews of analogous terrorist acts and crimes as indicators of the likelihood that various types of adversaries would attempt similar acts at licensed nuclear facilities. The results of these reviews provide useful background information, but they cannot in themselves be codified to provide specific guidance for the design and evaluation of nuclear safeguards. The consequences of deliberately malevolent acts are also important in understanding potential societal risk. One study that considers the consequences of "successful" malicious acts has been conducted, and the NRC has reviewed the events that would follow a "successful" theft, as indicators of the probability of whether an adversary would be apprehended before his accomplished mission of theft could actually cause societal harm. However, to date there has been no attempt to systematically evaluate the relationships among these various effects.

13.4 Research Program Objectives

Recognizing that the nature of safeguards will always require that a large subjective component be involved in regulatory judgments, the safeguards research program is aimed at providing a more systematic, technical, and scientific framework so that regulatory judgments can be as uniform and comprehensive as possible. This general objective encompasses three specific research program objectives:

1. Analytical Methods - Develop, test, and refine analytical methods to aid in assessing or improving the effectiveness of licensees' safeguards components, subsystems, and systems in terms of the outcome of an adversary attempt at theft or sabotage.

2. Experimental Data - Conduct experiments to measure the performance against various adversary strategies of safeguards components, subsystems, and systems as a basis for validating or improving analytical methods, for acquiring necessary data for regulatory application of the analytical methods, and for gaining needed insights and information for improvements in safeguard regulation.

3. Special Studies - Perform special studies related to improvements in rulemaking, licensing, and inspection. The studies will include technical confirmatory work requested by NRC-user offices, as well as research to increase the utility of the risk perspective in the safeguards regulatory program.

13.5 Research Program Plan

During the preceding years, the safeguards research program has been successful in developing, testing, and applying analytical methods and computer programs to the evaluation of safeguards effectiveness at licensed nuclear facilities. The methods developed cover both physical protection and material control and accounting systems. These methods have been successful because they provide the means to determine more uniformly and objectively the effectiveness of safeguards measures that has been demonstrated by their utility in the rulemaking and licensing process. It is expected that some of these methods will also be useful for inspection. Development of a new method (Strategic Analysis) to aid in the establishment of acceptance criteria for safeguards system performance will also be initiated in late FY 1981.

In future years, it is planned to increase the application of previously developed methods to the solution of particular safeguards issues and problems of a research nature. In particular, efforts will be made to more closely couple evaluative methods with existing and future regulations. Thus, it is anticipated that greater emphasis will be placed on the special studies research objective with significantly less emphasis on the analytical methods objective. Safeguards issues requiring special study include reactor sabotage (especially by an insider), human factors, security force training and evaluation, interactions between reactor safety and safeguards, improved material management capabilities, risk and consequences assessments, and the requirements for short-term and long-term storage of waste nuclear materials. In addition, reprocessing and breeder safeguards requirements may need to be studied in keeping with possible shifts in national policy. Safeguards problems associated with decommissioning a facility, as well as the safeguards implications of energy parks will require research.

Research in these areas will be closely coordinated with efforts by other NRC offices, as well as other agencies such as the Department of Energy (DOE) and the Department of Defense (DOD) to prevent duplication of safeguards work.

13.5.1 Analytical Methods

Analytical methods to evaluate the effectiveness of physical protection and material control and accounting systems that have been developed during the past several years are currently being tested and/or employed by the Office of Nuclear Material Safety and Safeguards (NMSS) staff to assess their applicability to the safeguards regulatory process. One method has been directly applied in the licensing of nuclear reactors and others in rulemaking. Still others show great promise for licensing, and the Office of Inspection and Enforcement (IE) is assessing certain methods for use in regional inspections.

In the future, it will be necessary to make modifications, refinements, and improvements in existing analytical methods as the direct result of requirements to perform special studies. Inclusion of human factors aspects of the interactions between the adversary and the safeguards system as recommended by the Advisory Committee on Reactor Safeguards (ACRS) will be addressed. Also, new methods may be required to perform a systematic assessment and evaluation of the insider reactor sabotage problem. Methods and procedures will be developed for assessing safeguards impacts on the safety requirements (and vice versa) for reactor designs. Work will continue on the development of methods (Strategic Analysis) for systematically determining safeguards acceptability criteria from preassigned levels of risk and other general regulatory requirements. These will include formalized methods to aid in the development of regulations and criteria to safeguard reactors and other fuel facilities against nonviolent as well as violent adversaries.

13.5.2 Experimental Data

Experimental validation of models used in the evaluation process will stress the design and use of independent field tests of safeguards and the comparison of these results with predictions generated by the models. Thus, efforts in this area will tend to increase over the next 5 years. The tests will be subject to the availability of DOE field-test sites and other sites such as the Allied-General Nuclear Services (AGNS) facility at Barnwell, S.C. A major result of these tests--planned to start in FY 1982 and conclude in FY 1987--will be the expansion of the present limited understanding of the impact of human factors on the problem of evaluating the effectiveness of safeguards. The results of these tests will be compared with the results from other developed models and computer programs and with expert opinions. The research will include collecting data to support staff application of the analytical methods, to support the use of these techniques in parametric and sensitivity studies, and to support special studies.

Bounds will be established on the magnitude and form of potential radiological releases that might be caused by shipping cask closure failure or general structural failure as a result of explosive attacks (spent-fuel cask secondary violation). The effects of sabotage on spent fuel at reactor and storage sites will be determined.

13.5.3 Special Studies

In the coming years, greater emphasis will be placed on the performance of special studies. These studies, which are of a research nature, will help to provide answers to safeguards problems and to resolve safeguards issues. Some of these studies have not been performed previously because NRC did not possess the necessary tools, i.e., the analytical methods supported by experimental data. However, the maturing of the safeguards research program in the very recent past by the development of improved analytical methods allows the accomplishment of important special studies. Some refinement to the methods will be necessary as the studies proceed.

Typical of the special studies needed to aid in the establishment or improvement of a safeguards technical base are the following:

1. Safety/Safeguards Interface - The impact of safeguards requirements on plant safety and vice versa is an important concern in power reactors. Safeguards problems may arise because of increased numbers of workers on site implementing new safety requirements or because of emergency situations at a plant. Any new safeguards requirements must be seriously considered for their safety impact before implementing them.

2. Reactor Sabotage by Insider - Because of the great complexity of reactor plants and the number of vital areas in each, it is difficult to protect against an insider intent on causing radiological sabotage. New methods for addressing the insider reactor sabotage problem require development, and solutions will undoubtedly require close coordination of safety and safeguards.

3. Human Factors - As was clearly shown by the accident at TMI, the response of human beings in an emergency is of crucial importance.

The same is true of safeguards systems on which the security force will be called upon to perform all important functions in a safeguards incident. It is therefore important that security force tactics and training be evaluated and that recommendations be made for their improvement.

4. Reprocessing and Breeder Safeguards - The safeguards requirements of nuclear fuel reprocessing and breeder reactors will need to be examined if it becomes national policy to endorse their use. In particular, it will be necessary to study the safeguards implications of a greatly expanded transportation network for special nuclear material.

5. Short-Term and Long-Term Waste Storage - Determine if there are requirements for institutional protection during the storage of nuclear waste and nuclear fuel that require examination.

6. Energy Park Safeguards - As in the case of breeder reactor safeguards, it would be necessary to study the special requirements of safeguards for energy parks if their use should become part of national policy.

14. FUEL CYCLE AND MATERIAL SAFETY RESEARCH

This section addresses safety aspects of all NRC licensed activities except mining, milling of ore, reactor power operation, and waste management. It covers not only activities involved in processing reactor fuels, but also ancillary reactor systems and the transportation of all NRC-licensed materials as well as their use in medicine, research, industry, and consumer products.

The risks associated with the licensed activities in this section are generally not readily compared to other risks regulated by the NRC. However, the use of cost effectiveness and cost-benefit considerations in regulatory decisionmaking (including the allocation of resources) requires an improved ability to compare risks and risk-reduction proposals. Thus, RES is beginning a major effort to develop risk assessment methods that will permit comparison, across the entire fuel cycle, of risks, uncertainties in risk estimates, and cost-benefit ratios for risk-reduction proposals.

A major element in risk assessment is the prediction of safety system performance under accident conditions. The current fuel cycle and materials safety research program will improve the NRC ability to predict the performance of safety systems (including administrative procedures) and consumer products under conditions that are judged by the NRC regulatory staff to represent the principal sources of risk in each licensed activity.

14.1 Fuel Cycle Facility Safety

14.1.1 Regulatory Objective

The broad objective of the NRC's fuel cycle facility safety program is to ensure that operation of nuclear fuel cycle facilities does not present undue risks to the public health and safety from accidental releases of radioactive material.

14.1.2 Technical Capabilities Required

To assess and evaluate the capabilities of fuel cycle facility safety systems to prevent or mitigate the consequences of accidents, the NRC staff must be able to:

1. Define--in terms of scenarios and bounding initiating parameters--the nature of major credible accidents that could lead to the release of significant quantities of radioactive material to the environment;

2. Define and evaluate operating parameters and related physical conditions for accident sequences (e.g., flows, pressures, and aerosol generation rates) associated with credible accidents; and

3. Assess the response of fuel cycle facilities during accidents, including the performance of safety systems, and the resultant source term for the release of radioactivity into the environment.

14.1.3 Status of Capabilities

The current approach used in licensing is to evaluate the performance requirements on plant safety and confinement systems based on conservative assumptions regarding accident and accident effluent characteristics to determine if postulated accidents would result in a safe response. These assumptions are based primarily on an extrapolation of data derived from historical accidents that have taken place in DOE production facilities. Development of experimentally validated, realistic methods for analyzing facility responses and radionuclide source terms resulting from major accidents in fuel cycle facilities would improve the NRC ability to:

1. Estimate the overall conservatism in licensing assumptions regarding accident source terms and therefore to justify the imposed design and operating requirements in terms of necessity, adequacy, and relative cost; and

2. Standardize the nature and scope of facility safety reviews by the licensing staff by use of an experimental data base derived from actual equipment and systems and validated analytical methods.

14.1.4 Research Program Objectives

The specific objectives of this research program are to develop the necessary data on accident initiation and sequences and to provide methods for analyzing the facility response required to support assessment of specific fuel cycle facility accidents. This includes assessing the uncertainties in both the assessment methods and the physical data base.

Research in this program is also intended to support analysis of accidents in nuclear power plants that take place outside the containment structure and do not directly affect the safety of the reactor core, e.g., accidents in the spent-fuel storage pool.

14.1.5 Research Program Plan

Development of a data base and best-estimate analytical models for assessing radioactive releases from major accidents in LWR fuel cycle facilities began in FY 1979 and will continue through FY 1985. The facility types for which accident scenarios and parameters are being defined for this study are fuel fabrication, spent-fuel storage away from reactors, and waste solidification facilities, as well as the proposed Oak Ridge National Laboratory Hot Experimental Test Facility for the advanced fuel reprocessing experiments. Accidents for which experimental data and analytical methods are being developed include fires, explosions, spills, equipment failures, tornadoes, and criticality. Final documentation of study results will include the basis for selection of accident types and parameters; available experimental data for the range of variables for all accidents; and description of methods for predicting aerosol generation, transport, deposition and resuspension, gas dynamics, and component response.

Research is planned to define the accidents of major consequence that are possible in facilities of users and manufacturers of products containing radioactive materials and to determine parameters necessary for estimating the accident consequences. The information will be compared with the input, experimental data base, and analytical capabilities developed for accident analyses of LWR fuel cycle facilities to determine what, if any, additional experimental data or analysis method development are required to predict accident consequences in these facilities. This program has been developed on the assumption that limited additional development and testing of analytical methods and data will be needed to extend the capability to these facilities.

The need for a capability to evaluate accident consequences in alternative or advanced fuel cycle facilities depends on circumstances that are difficult to predict at this time. Initiation of such studies requires specific design information sufficient to establish accident scenarios and relevant accident parameters followed by an assessment of the applicability of the existing data base and analytical methods for assessing accident consequences. Nevertheless, for planning purposes, it has been assumed that a need will exist and that sufficient information will be available by FY 1984 to allow the start of a study similar to that described previously for facilities of users and manufacturers of products containing radioactive materials.

The analytical methods and supporting physical data developed for LWR fuel-cycle facilities will be extended during this period to enable realistic analysis of accident consequences in portions of nuclear power plants not directly affecting reactor safety (e.g., the radwaste system). The same basic procedure will be followed as described above, i.e., (1) define accident types and significant parameters that could produce major consequences; (2) determine the adequacy of available experimental data for realistic evaluation of the consequences of these accidents, and conduct additional experiments as needed; and (3) evaluate the adequacy of current analysis models for realistic evaluation, and modify models as needed.

The order in which the follow-on studies to the basic LWR fuel cycle study will be performed (i.e., facilities using radioactive materials, advanced fuel cycle facilities, and nonreactor accidents in nuclear power plants) is flexible; whenever possible, priorities will be established in the order of decreasing risk.

Criticality safety, as a special accident issue, has been the subject of continuing research by the NRC. The efficacy of the several techniques developed to analyze and protect against criticality are being demonstrated by comparison with a wide range of experimental data that will continue throughout the FY 1982-1987 period. In FY 1982-1983, criticality research will be directed toward spent-fuel storage, fuel transportation, and bulk fuel criticality for LWR fuel. In FY 1984-1987, criticality research will be extended to advanced fuel cycle materials; this will include analysis and validation of characteristics of the fuel type during bulk processing, element assembly, storage, and shipping.

14.2 Transportation Safety

14.2.1 Regulatory Objective

The objective of the transportation safety program is to ensure that radioactive material transportation activities are accomplished in a manner that properly protects the public health and safety under both normal and accident conditions.

14.2.2 Technical Capabilities Required

The establishment of regulatory standards for transportation safety and the evaluation of licensee submittals in this area require the following capabilities:

1. The ability to identify and characterize normal and accident transport conditions that are relevant to either potential releases of radioactive material or direct radiation exposures;

2. The ability to evaluate the effectiveness of alternative transport procedures and package designs in reducing the probability or magnitude of radioactive releases or direct radiation exposures; and

3. The ability to assess radiological consequences and risks associated with the release of radioactivity during the transport of radioactive materials.

14.2.3 Status of Capabilities

The current regulatory approach to ensuring transportation safety uses package performance test requirements as a surrogate for the actual conditions associated with normal and accident environments. Package acceptance criteria, based on these performance test requirements, are expressed in terms of limits on material release, shielding loss, and content subcriticality. Licensee compliance with these requirements can be demonstrated by actual package tests or through analytical assessments of package performance. In general, analytical methods have and are being used to evaluate package designs against the current performance test requirements; however, the sophisticated analytical capabilities necessary to evaluate the complex, high-payload package designs of the future have not been adequately validated.

The application of risk assessment methods to the transportation of radioactive materials has contributed significantly to an understanding of the level and sources of risk. However, the Commission's comments on SECY-79-593 note that, because of public concern, accidents that would result in large consequences must be considered in the regulations without regard to the overall level of risk that they would produce. To implement this policy, suitable analytical methods, transportation system information, package and content test data, and input for the required value/impact appraisals will be needed. Information is needed far beyond what is currently available.

14.2.4 Research Program Objectives

Research objectives in the area of transportation safety are to:

1. Provide information, based on historical experience and experiment, that will allow severe, modal-dependent accident environments to be characterized by a set of package performance tests;

2. Identify the value/impacts of regulatory controls on "nonpackage" aspects of the transportation system;

3. Develop data and analytical methods for (a) evaluating the durability of shipping packages in severe accidents, (b) characterizing quantities (chemical and physical form) of radioactive material releases, and (c) establishing reasonable and practical posttest package acceptance criteria;

4. Provide methods and supporting test data for assessing the structural, thermal, shielding, and criticality performance of advanced shipping-package designs; and

5. Identify and assess factors in normal shipping operations that have a high likelihood of adverse impact on regulatory compliance (e.g., human errors, maintenance schedules, and component degradation while in use).

14.2.5 Research Program Plan

Projects will be initiated throughout the FY 1982-1987 period to provide the methods and test data necessary for adequate evaluations of packages when they are judged against current transportation regulations. In FY 1982-1983, studies will be initiated to provide information on brittle fracture of containment vessel materials and to evaluate radiative heat transfer between fuel assemblies and packages under LOCA conditions.

Development of the SCALE code for analysis of thermal, shielding, and criticality behavior has been underway for several years and will be completed in FY 1981 for evaluations pertinent to LWR fuel shipments. The code will be considered for updating to make it applicable to advanced fuel cycle shipping and storage.

Special studies will be performed, starting in FY 1982, to establish experimentally the chemical and physical form of any radioactive material that would be released in severe transportation accidents involving irradiated fuel. For other radioactive materials and their packages, release fractions will be determined as a function of accident severity to improve the understanding of transportation accident risk.

It is expected that by FY 1985 shipping requirements for advanced fuel cycle materials will be sufficiently well defined so that RES can initiate confirmatory research studies. Unique requirements for containers associated with these fuel cycles will be determined, and studies similar to those performed for the LWR fuel cycle will be initiated to assist the licensing staff in evaluating the safety of these containers. Areas requiring study will include cooling requirements, nuclear criticality requirements, structural loads on components, material leakage characteristics, and the safety implications of the physical and chemical forms of the material shipped.

Two projects are planned to evaluate specific problems associated with normal shipping-container usage. The first project will evaluate both human engineering problems in the design and human factor aspects in the use of radioactive material packagings that affect shipping-container safety and compliance with shipping regulations; this project will develop the information necessary to support the development of guides and recommendations to correct problems encountered. The second will be a project to evaluate the significant potential safety effects that can result from normal use of shipping containers (such as the capabilities of various seals) and to establish procedures, criteria, and tests for an NRC or licensee package testing and retesting program. These projects will be initiated in FY 1984 and FY 1985, respectively.

Several projects, referred to collectively as the Transportation Modal Study, were initiated in FY 1980-1981 to evaluate the performance of shipping containers in extremely severe accidents in air, rail, ship, and truck transport. Studies are under way to (1) characterize severe accident environments as a function of transport mode, (2) determine effects of these environments on existing packages, (3) develop performance test requirements based on these characterizations, and (4) determine whether changes in the regulations are required. The program includes definition of the accident forces, evaluation of various acceptance standards through value/impact analyses, and studies of transport system logistics. The value/impact analyses will include assessment of operational and administrative controls, including the impact of emergency planning and response alternatives and chemical and physical form of materials shipped. Although the basic modal study will be completed by FY 1986, a maintenance budget has been proposed through FY 1987 to allow for recognition of current developments or novel transport systems, and to respond to any shortcomings identified during staff application of results to regulatory use.

14.3 Radioisotope Safety

14.3.1 Regulatory Objective

The objective of the NRC's radioisotope safety program is to ensure that uses of radioactive materials (source, special nuclear, and byproduct materials) for industrial, medical or research purposes, are conducted in a manner that meets EPA environmental standards and is consistent with the ALARA principle in 10 CFR Part 20.

14.3.2 Technical Capabilities Required

In order to perform safety assessments relating to the use of radioactive materials in industrial and medical applications and in consumer products, the NRC must be able to evaluate:

1. The uses of the material in relationship to alternatives for achieving a comparable social benefit;

2. The confinement capability of the facilities, devices, and safety systems proposed to protect the public and workers under conditions of normal use; and

3. Procedures and controls for avoiding inadvertent, incorrect use of radioisotopes, e.g., misadministration of medical or research applications.

4. The potential exposure of the general public to radiation as a result of material loss from releases resulting from product exposure to severe environmental conditions (including disposal) or from the failure of safety systems or protective devices.

5. Technical bases for establishing de minimus levels for regulation of radioisotopes.

In addition, in order to identify generic deficiencies in the regulatory standards or in equipment or facility designs, the NRC staff must also be able to study and analyze actual failures or malfunctions in the protective systems employed in the use of these radioactive materials.

14.3.3 Status of Capabilities

Although the NRC has imposed limited design, testing, and quality assurance requirements on the manufacture and use of a wide variety of specific products containing radioactive materials, a data base sufficient to permit the establishment of comprehensive, uniform product standards does not now exist. Moreover, the NRC has no existing, dedicated capability for independent validation of applicant-generated product test data, development of new or improved test methods, nor for conducting postaccident examination of systems or devices that have failed in use.

14.3.4 Research Program Objectives

Research objectives in the area of product safety are to:

1. Provide data needed to supplement the existing NRC studies of the adequacy of current regulations, standards, and guides applied to products containing radioactive material and, as necessary, provide data needed for their improvement; and

2. Establish, through independent testing of "off-the-shelf" products containing radioactive materials, the ability of these products to meet NRC-imposed design and testing standards and to determine the degree to which these products survive acceptably simulated test conditions representative of severe normal use, accident, and disposal environments.

14.3.5 Research Program Plan

In order to conduct an effective, independent testing program for consumer and industrial products that use radionuclides, NRC needs to obtain "off-the-shelf" products containing NRC-regulated radioactive materials and on a routine and continuing basis develop data on:

1. The ability of these products to satisfy NRC-imposed design and testing requirements; and

2. The potential hazards that could occur (e.g., leakage of radioactive materials and loss of shielding) as a result of imposing testing conditions representative of severe accident, disposal, and normal use environments.

In order to accomplish this plan, it is first necessary to review and compile the NRC-imposed design and testing requirements and to define applicable severe environmental conditions. For completeness, the study will also determine the design and testing requirements imposed on similar products by other countries.

Data from tasks 1 and 2 above will also be evaluated to establish the degree to which NRC-imposed design and testing requirements (task 1) result in products that acceptably survive the severe environmental conditions (task 2). If these comparisons are unfavorable, the program will provide data on modifications appropriate for NRC regulations and standards applicable to these products.

15. RADIOLOGICAL PROTECTION RESEARCH

The use of radioactive materials by man unavoidably results in some increase in human exposure to radiation. Safety regulations are aimed at limiting the unplanned release of radioactive material or radiation from licensed activities, while radiation protection regulations are aimed at reducing exposure to all regulated sources of radiation. Different limits are set for occupational and public exposures, but both are further guided by the principle that exposure should be as low as reasonably achievable (ALARA).

Regulatory decisions in this area are largely judgmental and often dominated by nontechnical considerations. The effects of exposure to low levels of radiation are small enough to be difficult to demonstrate and quantify with precision. The Committee on Biological Effects of Ionizing Radiation in their report, The Effects on opulations of Exposure to Low Levels of Ionizing Radiation, says, "Estimates of risk at low doses depend more on what is assumed about the mathematical form of the dose-response function than on the data themselves," and further stated, "The Committee does not know whether does rates of gamma or X rays of about 100 mrads/yr are detrimental to man." A second difficult problem arises in the attempt to achieve a balance of societal risks and benefits where the risks and benefits of a specific activity are not equally shared among the individuals in society.

The function of research in this domain is to develop information in those areas where its lack is impeding efficient regulatory action by NRC. Consequently, the program is conditioned not only upon the magnitude of risk or uncertainties in risk but also upon the institutional requirements of the regulatory process.

15.1 Siting Standards and Criteria

15.1.1 Regulatory Objective

The objective of siting standards and criteria is to ensure that both proposed and alternative sites are systematically analyzed and compared from the perspectives of public safety and protection of the environment.

15.1.2 Technical Capabilities Required

To meet this objective, the NRC must have the ability to:

1. Compare the environmental and physical characteristics of candidate sites, and assess the potential effects of nuclear facilities at the site on public health and safety and on the quality of the environment;

2. Identify and assess regional (as opposed to strictly local) impacts on the environment resulting from nuclear facility siting, and

3. Forecast changes in demographic and land-use patterns over the operating lifetime of a nuclear facility, and assess methods of avoiding potential conflicts with siting standards and criteria.

15.1.3 Status of Capabilities

Although substantial information already exists on many factors considered in siting decisions, some of the ways in which these factors are evaluated in considering alternative sites can be improved. Current siting policy and practice are described in the Report of the Siting Policy Task Force (NUREG 0625). A system enabling those responsible for regulatory decisions to make quantitative evaluations of different types of environmental impacts should be further developed. Additional use of cost/benefit analysis could considerably advance the technical basis for the comparison and evaluation of alternative sites.

More information is needed on how population densities and land-use patterns at potential sites influence the effectiveness of evacuation and other emergency response actions. The technical basis for methods of assessing population densities and land-use patterns should be improved, with an emphasis on predictive modeling of changes to be expected over the operating lifetime of a nuclear facility. Evidence from existing nuclear power plants indicates that the establishment of a nuclear station itself causes population density to increase and land-use patterns to change, especially as a result of the effects on local taxation. Improvements in models for forecasting population and land-use pattern changes would allow more detailed comparison of sites on the basis of averages over the operating lifetime of the plant. Because unforeseen events could result in unfavorable population densities and land-use patterns after a nuclear station has been licensed, methods of ensuring that these items remain within acceptable limits (such as zoning controls) should be evaluated.

15.1.4 Research Program Objectives

The objectives of site evaluation and selection research are to improve the methods for comparing alternative sites proposed by licensees on the bases of public health and safety and environmental risk; to obtain additional necessary information on the relationships of population density, land-use patterns, and siting alternatives for incorporation into the nuclear station site-selection process; and to provide validated technical bases for standards and siting criteria.

15.1.5 Research Program Plan

Work will continue on the development of a modeling system and an integrated data base for providing the information needed for comparing alternative siting choices on a regional basis. Models for forecasting regional demands for expansion of electricity-generating capacity and impacts on the environment will be linked to form a prototype modeling system. The system, which initially will be applicable to the New England region, will improve and facilitate the consideration and evaluation of tradeoffs among power plant siting alternatives. After the New England regional model is verified in FY 1982, it will be adapted to other regions beginning in FY 1983 and continuing through FY 1985.

Analytical methods will be developed for evaluating alternative sites and alternative plant systems. This will reduce the need for site-specific, time-dependent, detailed information applicable to future sites without impairing the overall quality and value of environmental impact assessments. Specific plant systems and alternatives will be examined to identify those that will minimize environmental impacts while taking into consideration engineering feasibility, safety assurance, and reasonable cost/benefit balancing. This work will include evaluating single vs. multi-unit sites and is planned to start in FY 1982 and extend through FY 1986. Additional research considering similar bases for siting standards and criteria applicable to other nuclear facilities will be pursued throughout the planning period.

For the next 10 years, changes in land value and use patterns and population density will be observed with respect to the proposed population density criteria. This information will be assessed as a basis for testing predictive models and for developing more realistic and comprehensive methods of forecasting land-use changes.

The feasibility of land-use controls or other means to prevent encroachment of undesirable land-use patterns into the immediate vicinity of nuclear power stations will be studied. During FY 1983, the most effective methods of maintaining acceptable land-use patterns on a cost-effective basis will be determined.

In FY 1982, studies will continue on how evacuation and other emergency response requirements at proposed sites for nuclear facilities are influenced by population density and land-use patterns. Research will determine what major obstacles to efficient emergency plans are posed by high-density residential areas, local transportation patterns, various types of commercial development, alternative land use in the vicinity of nuclear facilities, and other similar factors.

15.2 Emergency Preparedness

15.2.1 Regulatory Objective

The objective of the emergency preparedness program is to assess and ensure the suitability of licensees' emergency plans and to maintain NRC capability to protect the public health and safety in emergency situations that may be associated with licensed activities.

15.2.2 Technical Capabilities Required

To meet this objective, the NRC must have:

1. The ability to provide technical evaluation of primary offsite data and to establish the criteria and standards for postaccident data collection, processing, and recovery decisions;

2. Assurance that reliable system exists for communicating information to emergency operations centers in the emergency-preparedness network and to other Federal, State, and local authorities, i.e., land-line systems such as telephone, teletype, and radio linkage;

3. The ability to make independent assessments of licensees' emergency preparedness procedures regarding timely notification (e.g., warning systems to alert 100 percent of the people within 5 miles and 90 percent of the people within 10 miles within 15 minutes);

4. Reliable data acquisition systems, with collation and analysis capabilities maintained in readiness to support engineering judgments; and

5. A verified plan for implementing NRC emergency procedures, making certain that there is a clear delineation of NRC singular responsibilities and those responsibilities that interface with emergency actions of other authorities.

15.2.3 Status of Capabilities

The Commission's emergency response capability is being improved by training personnel and add[illegible]ipment to the NRC Operations Center. Instrument testing and evalu[illegible] is in progress, and systems are being studied that will provide remote reactor-status information links, environmental monitoring, data acquisition and processing, and communications.

The realignment of emergency preparedness roles among the several Federal agencies has resulted in changes in the detailed duties and responsibilities of the NRC; thus, additional knowledge and equipment will be required so that revised planning can be effectively executed. Overall planning is being coordinated with the Federal Emergency Management Agency (FEMA) to identify gaps in designated responsibility and interfacing problems and to identify potentially overlapping responsibilities and authorities under the new emergency preparedness plans. This assessment should ensure that all foreseeable needs can be met and should avoid wasteful duplication beyond the minimum redundancy required for confident emergency responses in keeping with the concept of defense in depth.

15.2.4 Research Program Objectives

The primary broad objective of this research program is to support the NRC effort in concert with FEMA and to be well prepared to respond to emergencies with the capability to meet the Commission's responsibilities to other Federal agencies and to State and local authorities. A second objective is to give the NRC the capability for acquiring the timely technical information it needs to be well prepared to respond to nuclear emergencies.

15.2.5 Research Program Plan

This program is still formative and will continue with the development and testing of emergency radiation detection and measurement instrumentation that began in FY 1977. The emergency instrument testing effort is scheduled to be completed in FY 1983. There is a continuing research effort to provide the bases for reliable emergency measurements by testing the performance of radiation detection and measurement instruments that are simple to operate and can withstand the adverse environments that might result from an accident. A portable air sampler and a counter for radioiodine collection and measurement will be evaluated in FY 1982. An ongoing program evaluating warning systems and testing the audibility of various devices used to alert the public to an emergency will produce some results before FY 1982 with further verification of performance to follow through FY 1983.

Human factors affecting the response of nuclear power plant staff and the general public will be studied beginning in FY 1982. Throughout the planning period, human response under stress, considering all classes of emergencies, will be assessed as a continuing program. The socioeconomic effects of accidents will be evaluated as a further consideration of the elements of emergency preparedness influencing site selection and as a measure of environmental impact potential. In the period FY 1983-1987, research considering the efficacy of countermeasures, recovery, and mitigative actions to be applied in the event of an emergency will be conducted.

15.3 Decommissioning

15.3.1 Regulatory Objective

The objective of NRC's decommissioning regulatory program is to ensure that the decommissioning of all types of licensed facilities is conducted so that the site contamination levels are reduced to acceptable levels, public and occupational exposure from routine decommissioning operations is as low as reasonably achievable (ALARA), and generated wastes are placed in a safe and storable form.

15.3.2 Technical Capabilities Required

To establish decommissioning standards and review licensee plans and applications, the NRC must have the ability to:

1. Evaluate the nature and distribution of radioactive contaminants within the facility; evaluate methods and techniques for decommissioning applicable to facility types for effectiveness, safety, and costs; and estimate the nature and volume of the wastes that will be generated;

2. Evaluate the degree to which siting, construction, design, and operating procedures described on initial applications will facilitate eventual decommissioning;

3. Estimate the reliability of licensee cost estimates for decommissioning so that financial responsibility can be established and ensured; and

4. Evaluate the residual contamination following decommissioning to ensure that it meets existing standards for radiological safety.

15.3.3 Status of Capabilities

The NRC is currently developing detailed regulations and guides for decommissioning nuclear facilities that will establish both the acceptable levels of occupational exposure, and the acceptable levels of residual contamination for unrestricted site use based on EPA standards. These regulations are based in large measure on studies performed for the NRC Office of Standards Development by Battelle Pacific Northwest Laboratories. These studies evaluate cost, safety, and effectiveness of techniques for decommissioning based on existing data. They require assumptions regarding residual plant contamination levels, the effectiveness of specific decommissioning techniques, and acceptable levels of residual contamination levels for unrestricted use.

Actual nuclear plant decommissioning operations have provided relatively little data. Data from decommissioning operations and an understanding of alternative decommissioning methods are important for supporting NRC actions on licensees' proposals for decommissioning their plants and for generic actions on decommissioning regulations, policies, standards, and guides.

15.3.4 Research Program Objective

The objectives of the decommissioning research program are to:

1. Collect and verify data from nuclear facilities undergoing decommissioning (LWRs, fuel cycle facilities, and facilities of users or producers of radioactive systems that are representative of licensed plants currently in operation);

2. Use these data to assess critical assumptions made in previous (SD/PNL) studies and to improve the accuracy of the cost estimate and exposure models;

3. Develop and verify analytical models to assess the costs, safety, and waste characterization set forth in future license applications;

4. Assess new approaches for improving the decommissioning process with respect to cost, safety, waste generation, and residual radioactivity levels; and

5. Assess state of the art in establishing residual radioactivity levels at decommissioned sites and the impact of these levels on the cost and waste disposal needs of decommissioning.

15.3.5 Research Program Plan

The research program has been formulated to develop an experimental data base to support evaluation of decommissioning alternatives, particularly where the earlier SD/PNL studies required assumptions with major potential significance. The program is also formulated to provide insight regarding the potential for significant, unanticipated aspects of decommissioning.

Beginning in FY 1981, several planned decommissionings of nuclear power plants and fuel cycle facilities will provide opportunities for obtaining actual field data. Data will be gathered on costs, contamination and radiation levels, effectiveness of decommissioning techniques and methods, waste volumes, and characteristics and assessment of postdecommissioning residual radioactivity.

Three programs were initiated in FY 1979 to provide an extensive experimental data base on LWR decommissioning. These programs, which will continue thorugh FY 1984, will analyze long-lived activation products in LWR reactor vessels and biological shields, contamination levels and characterization of radioactivity in other areas of the plant, and the value of decontamination as a precursor to decommissioning.

The data acquired from these programs and from other sources will be used to test the generic analyses developed for SD and to develop and verify more precise predictive models. This work on improvement of predictive models will be initiated in FY 1983.

In FY 1985, an examination of the special problems associated with decommissioning the facilities of users or manufacturers of products containing radioactive materials will be initiated. This research will include generic studies of decommissioning such facilities, supported by data gathered at operating facilities or those undergoing decommissioning.

Although the program described above is primarily designed to support development of regulations and guides to implement the ALARA concept, it is also expected to identify and document potential technical improvements in decommissioning methods. Starting in FY 1984, a small continuing effort of laboratory studies will assess the basic feasibility of potential improvements and their contribution to risk reduction.

15.4 Effluent Control

15.4.1 Regulatory Objective

The objective of NRC's effluent control regulations is to ensure that licensees keep routine releases of radionuclide effluents to unrestricted areas ALARA taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations.

15.4.2 Technical Capabilities Required

To meet this objective, the NRC must have the ability to identify the sources and levels of routine releases of radionuclides and to determine the technological feasibility and cost of alternative means for reducing their level. This is required for all regulated facilities, including reactors, fuel cycle facilities, and facilities of users and manufacturers of products containing radioactive materials.

15.4.3 Status of Capabilities

The ALARA principle is implemented in current LWR license reviews by determining that the effluent control system will, for assumed conditions representative of the plant's operating life, limit the release of radioactive material to values that meet the quantitative dose standards established in the NRC and EPA regulations. These dose standards limit individual exposures to values that are a small fraction of those resulting from normal background radiation.

As an example, these limits for LWRs specified in Appendix I of 10 CFR Part 50 are: (1) annual exposure from liquid effluents will not exceed 3 millirems total body or 10 millirems to any organ, and (2) annual exposure from gaseous effluents will not exceed 10 millirads for gamma radiation and 20 millirads for beta radiation.

The ALARA principle is also applied to individual fuel cycle facilities and other facilities licensed to handle radioactive materials where differences in process, design, and operation require individual cost-effectiveness evaluations.

Current methods for predicting the response of effluent control systems in LWRs are based on an evaluation of the general performance of individual system components using the Gaseous and Liquid Effluent (GALE) evolution codes. These methods are based on a wide range of operational data, and, in general, they provide realistic estimates considered to be adequate.

15.4.4 Research Program Objectives

Specific objectives of the research program relative to LWR effluents are to:

1. Provide parametric data relative to liquid gaseous, and solid effluents as a function of plant design and operation during all operating phases and throughout the plant lifetime;

2. Provide data on the actual performance of effluent treatment systems in operating LWRs as a function of effluent characteristics and system operation; and

3. Provide a basis for evaluation of the licensee techniques and capabilities for measuring and monitoring effluents.

Specific objectives of the research program relative to fuel cycle facility effluents are to:

1. Provide data on the actual performance of effluent treatment systems and components in LWR fuel cycle facilities as a function of effluent characteristics and system operations;

2. Provide data needed to supplement that obtained by NMSS on the actual releases from a wide variety of facility types associated with the use and manufacture of products containing radioactive material and on the performance of effluent treatment systems and components as a function of effluent characteristics, system operation, and system design; and

3. Identify novel effluent characteristics associated with alternative or advanced fuel cycles, and develop analytical methods and performance data on effluent treatment systems in sufficient time to make licensing decisions on proposed effluent treatment system performance.

15.4.5 Research Program Plan

Research initiated in prior years to evaluate effluent treatment system performance in operating LWRs as defined in Section 15.4.3 will continue through FY 1983, at which time measurements at six LWRs will be completed. Additional data will be collected on liquid, gaseous, and solid effluents and on the performance of the radwaste system. In these plants emphasis will be placed on (1) the relationship of the actual ventilation system performance to the as-designed performance, (2) the changes in crud and radionuclide transport in the primary coolant as a function of changes in power, coolant flow, or coolant chemistry as needed to supplement existing data, (3) the performance of waste solidification systems, and (4) the behavior of iodine in the containment. When the project is completed, a final report will be prepared discussing the data and conclusions from the entire program, including measurement problems and provision of preferred measurement techniques to ensure valid assessment of performance.

Past studies of advanced effluent treatment systems have included noble gas stripping by countercurrent flowing freon and cleanup of liquid wastes by reverse osmosis. Additional advanced effluent treatment concepts, particularly those applicable to holdup of radioactive gases in fuel cycle facilities and waste solidification, will be identified in FY 1982. Candidate processes will undergo laboratory feasibility testing under conditions representative of appropriate LWR and fuel cycle facility applications starting in FY 1983.

Inplant measurements similar to those performed in LWRs will be initiated in FY 1983 in some non-reactor facilities to determine effluent characteristics and effluent treatment system performance under operating conditions.

15.5 Exposure and Health Effects

15.5.1 Regulatory Objective

The objective of the radiation exposure and health effects regulatory program is to ensure that licensees maintain public and occupational radiation exposures ALARA, taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations.

15.5.2 Technical Capabilities Required

To meet this objective, the NRC must have the ability to:

1. Confirm that EPA standards for permissible exposure to radiation have been properly implemented through NRC regulations;

2. Assess the effects of measured, absorbed neutron dose in terms of gamma or x-ray dose equivalent;

3. Calculate radiation doses received by individual organs following inhalation or ingestion of radioactive material of the types encountered in NRC regulated activities, based on how the material is metabolized;

4. Predict mortality and morbidity resulting from radiation exposure of populations, based on demographic projections;

5. Measure and evaluate the performance of protective equipment such as respirators;

6. Test and evaluate the effectiveness of dosimeters and instruments used to detect and control occupational radiation exposure; and

7. Evaluate the effectiveness of quality assurance programs for radiation monitoring methods and instruments.

The technical capabilities required to maintain worker radiation exposures ALARA through design and operational specifications are addressed in the occupational protection section (Section 15.6).

15.5.3 Status of Capabilities

The NRC currently adheres to the linear dose-effect theory which tends to be conservative for low doses of low linear energy transfer (LET) radiation, such as gamma rays and x-rays, but might not be conservative for high LET radiation.

A proposal has been made that the value used for the relative biological effectiveness (RBE) of neutrons should be increased. It is not clear whether the effect of neutrons has been underestimated, or if the effect of X and gamma rays has been overestimated.

The NRC is similarly considering the effects of alpha emitters such as uranium. Retention, translocation, and rate of excretion by occupationally exposed human beings of a number of compounds occurring in the nuclear fuel cycle are relatively unknown. For example, uranium hexafluoride, ammonium diuranate, and uranium oxides are all compounds present in the manufacture of fuel for light water nuclear power plants. Mixed oxides of uranium and plutonium occur in the fast breeder reactor fuel fabrication systems. Internal dosimetry, bioassay, and exposure standards each require verified data concerning these compounds to reduce uncertainties.

Currently available health effects estimates do not consider age, sex, or race or changes in the population at risk. These omissions increase the uncertainties in the estimations of deaths and incidences of ill health resulting from exposure to radioactive and nonradioactive pollutants.

Research findings indicate that commercially available personnel dosimeters do not properly measure the neutron dose received by workers in nuclear power stations. Dosimeters and other instruments that measure neutron doses and dose rates must be tested to provide a basis for guidance on improving licensee performance.

Radiation monitoring and occupational radiation exposure control are accomplished by the licensee using integrated systems and health physics procedures and practices that conform with NRC regulatory guides. Instruments and methods should be improved to ensure that measured and calculated doses reported by different licensees are comparable and based on acceptable standards.

15.5.4 Research Program Objectives

The broad objectives of exposure and health effects research are to:

1. Improve understanding of the relationship between exposure to radiation and the magnitude of the biological effects produced;

2. Provide information on the metabolism of inhaled and ingested compounds containing radionuclides not previously investigated;

3. Improve the methodology for predicting deaths and illness as a result of radiation exposure in specified populations; and

4. Evaluate the effectiveness of protective equipment and of instruments used to detect radiation.

15.5.5 Research Program Plan

Followup studies of human populations exposed to low levels of radiation are in progress. One such study will determine if exposure to diagnostic levels of iodine-131 in childhood produces an increased risk of developing benign or malignant thyroid neoplasms. Another study will determine whether exposure to thorium in a thorium-processing facility produces an increase in cancer deaths or exposure-related illness in the workers.

A study was initiated in FY 1981 to compare the effects of low doses of neutrons and gamma rays on mice; it will continue through FY 1987. Both life-shortening and genetic effects will be studied to assess the relative biological effect of neutrons.

Research is in progress to improve understanding of the metabolic behavior of inhaled or ingested material in a number of animal species. Studies of inhalation of mixed oxides of uranium and plutonium and of uranium hexafluoride such as might occur in industrial accidents will be completed in FY 1983. Studies are under way to investigate the effects of inhalation exposure to several chemical forms of yellowcake. The gastro-intestinal absorption of plutonium and other actinide elements of varying concentrations is being investigated to supplement the earlier work done on these elements. Data on distribution and retention of plutonium and other actinides injected into monkeys is being analyzed to improve metabolic models. Exposure records of internal depositions of uranium in human beings are being compiled by means of bioassay measurements of exposed uranium mill workers. These data are being correlated with whole-body counting and with the analysis of tissues from deceased mill workers to improve the bases for regulatory guides and standards applicable to occupational exposure.

A computer code is being developed that will reduce the uncertainties in predicting the number of deaths and years of ill health experienced in a specified population following exposure to radioactive and nonradioactive pollutants. The code will primarily provide an improved method for predicting population dynamics in a specified area and for estimating ensuing health effects.

Dosimeters and instruments used to detect and control occupational radiation exposure to neutrons will be tested through FY 1985. A primary consideration in regard to the performance of radiation detection and measurement instruments is their traceability to the National Bureau of Standards (NBS) system of calibration standards. Beginning in FY 1981, a program on the quality assurance of radiation measurements was begun with the NBS to (1) calibrate and characterize the thermoluminescence dosimeters (TLDs) and Readers used in the NRC TLD direct-radiation monitoring network, (2) develop radiation fields, exposure chambers, and other equipment for calibrating radiation-detection instruments, and (3) develop a quality assurance program for laboratories that calibrate radiation survey instruments used by NRC inspectors.

15.6 Occupational Protection

15.6.1 Regulatory Objective

The objective of the NRC occupational protection program is to ensure that licensees maintain occupational radiation exposures ALARA, taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations (in accordance with 10 CFR Part 20.1).

15.6.2 Technical Capabilities Required

To achieve this objective, the NRC must maintain an ability to identify the sources and levels of occupational radiation exposure and to determine the technological feasibility and cost of alternative means for reducing the level of exposure, e.g., by design or operational improvements or by decontamination. For those cases in which decontamination is used for exposure reduction, it is also necessary to assess occupational exposure associated with handling and disposal of waste products.

15.6.3 Status of Capabilities

In nuclear power plants, worker exposures are increasing, partly as a result of increasing radiation fields in areas where equipment must be maintained, repaired, and inspected and partly because the extent and frequency of such activities has exceeded earlier expectations. The two principal ways to reduce worker exposure are (1) to reduce the radiation levels at these locations and (2) to reduce the total time workers must be in the radiation fields.

With regard to reducing the radiation levels at the relevant locations, the general causes of the increase in radiation levels in LWRs are known to be the production, transport, and deposition of radionuclides within the primary coolant system. Specifically, the understanding of the corrosion, erosion, transport, and deposition phenomena within the primary coolant system and of the effects on these phenomena of changes in design, materials of construction, quality control, housekeeping practices, and operations is not sufficient to support regulatory requirements for such changes. Also, there are few methods and little data for analyzing the performance and reliability of LWR decontamination systems or for evaluating the net contribution (or reduction) they might make to occupational exposure. A complete assessment of problems associated with processing and disposal of wastes generated by the decontamination process, such as the mobilization of radioactive materials by chelating agents present in low-level waste burial grounds, and the impact of these problems on decontamination choices for the nuclear facility have not been adequately addressed.

With regard to reducing the time of exposure to radiation, the following are possible areas for evaluation and standard-setting by the NRC:

1. Increases in equipment reliability, with a corresponding reduction in routine maintenance requirements;

2. Relocation of equipment requiring inspection and maintenance away from locations where fields of high radiation might be expected; and

3. Training of personnel and development of procedures to minimize the time required for maintenance, repair, and inspection.

4. Development of equipment for remote maintenance and inspection.

Although there is a significant body of data dealing with these topics, it has not been brought together in a single source for easy access and consistent utilization in value/impact assessments of proposed standards and practices. This should be done and, as needed, the data base supplemented to provide the desired levels of exposure reduction.

Another area in which occupational exposures are high and are increasing is in operations with radioactive materials (such as use of radiopharmaceuticals and radiography) and other medical, engineering, and geologic applications of radiation. Much of this increase may be attributed to the growth of these operational uses of radioactive materials. Data are needed to assess the magnitude and causes of these increased exposures and to evaluate possible methods of reducing them.

15.6.4 Research Program Objectives

The specific objectives of this research program are to:

1. Provide verified data on occupational exposure from historic experience and onsite measurements;

2. Provide validated methods for assessing the adequacy of alternative facility and equipment designs to meet occupational exposure standards and reduce radiation field buildup;

3. Provide data on the costs and effectiveness of alternative methods for achieving exposure reduction, including alternative methods for facility and equipment decontamination; and

4. Provide data on the waste treatment and disposal problems associated with decontamination and data on approaches to minimize their impact.

15.6.5 Research Program Plan

DOE, the Electric Power and Research Institute (EPRI) and individual companies have significant research budgets and programs in the field of radiation exposure in nuclear facilities and approaches to reduce these exposures. The NRC research program will be strongly influenced by the research of these organizations to the extent that they develop data or analysis methods that meet NRC needs.

The nature and magnitude of occupational exposures actually encountered in LWRs and other nuclear facilities, their relationship to plant operations, and appropriate methods for reducing this exposure will be evaluated. These programs will give the NRC the necessary information to plan and direct other projects dealing with specific aspects of occupational exposure reduction so as to produce the greatest positive impact.

To the extent possible, data from the ongoing LWR source term measurement program will be evaluated for applicability to the assessment of buildup of radiation fields, crud deposition, and their relationship to design and operation of the plant. This will be followed by programs, initiated in FY 1982 and later, that will be specifically formulated to assess these effects in operating LWRs.

Decontamination studies will be funded in FY 1983 and beyond to evaluate (1) the effectiveness of state-of-the-art decontamination methods applied to routine LWR operations and (2) the requirements for and effectiveness of waste solidification systems operating on decontamination wastes, especially as they satisfy waste disposal requirements. Another related program planned for initiation in FY 1983 will assess postaccident decontamination as may be defined by the TMI-2 interagency study groups.

The nature and magnitude of the occupational exposure problem associated with operations at facilities manufacturing or using devices containing regulated radioactive materials will also be investigated. Studies of the relationships among exposure, operations, and quantities of materials handled at operating facilities will be initiated in FY 1984. Similar studies applicable to fuel cycle facilities will be initiated in FY 1986.

16. ENVIRONMENTAL IMPACT ASSESSMENT RESEARCH

The principal objective of the National Environmental Policy Act (NEPA) of 1969 is to build into the Federal decisionmaking process an appropriate consideration of environmental aspects of proposed actions. This objective was clarified by the Calvert Cliffs decision in 1971 which made clear the requirement for a comprehensive, in-depth consideration of the environmental consequences of Federal actions. The opportunity for public debate afforded by this process has resulted in administrative, judicial, and legislative decisions that require increasingly rigorous implementation of the NEPA. This trend requires that information and methods be developed to provide for more comprehensive assessment of potential impacts of the proposed and alternative actions. When the Commission has decided that an environmental appraisal must be prepared for a particular action, it is important that the staff develop environmental appraisals and statements that display for public scrutiny and comment the information and methods used in reaching conclusions.

16.1 Aquatic Resources and Ecosystem Impacts

16.1.1 Regulatory Objective

The objective of assessing the environmental impact of the construction and operation of nuclear facilities, intake and discharge structures, routine effluents, and accidental releases of radioactive material on marine and fresh water plants and animals is to provide a basis for protecting the quality of the environment in accordance with the requirements of the NEPA.

16.1.2 Technical Capabilities Required

To meet this objective, the NRC must have the ability to:

1. Predict the physical transport and dispersion of effluents in marine and fresh water systems from the point of release to the receiving surface and ground waters;

2. Assess licensees' predictive models of the ecological effects, the losses that result from the entrainment or impingement of fish, and the impacts on plants and animals from effluents from nuclear facilities discharged to marine and fresh water systems;

3. Predict and evaluate the impacts of nuclear facility operations on the life cycle of marine and fresh water plants and animals, with emphasis on areas near nuclear facilities; and

4. Assess the impacts of nuclear facilities on water quality.

16.1.3 Status of Capabilities

The NRC requires applicants for a nuclear station construction permit to describe, in quantitative terms, the physical, chemical, biological, and hydrological characteristics, seasonal ranges and averages, and the historical extremes for bodies of surface water and ground water that may be affected by the construction or operation of the station. These data and the design features of proposed stations are then evaluated by the NRC staff. Predictive models simulating the release, dispersion, transport, and deposition of effluents are used to assess the potential environmental and human health effects of construction and operation of the facility. Many of the present models are powerful mathematical tools lacking only verification by field data. Reliable qualitative and quantitative data on the impacts of aquatic effluents from operating nuclear facilities are sparse; they lack detail concerning the cumulative releases of soluble and insoluble salts, suspended solids, radionuclides, chemical elements, organic compounds, and disease-causing entities. There are inadequate data concerning the pathways and fate of effluents and their effects on biota.

Most predictive models based on available biological data and used for environmental impact assessments are unverified mathematical simulations. Field data available for areas in the vicinity of operating systems are uncertain bases for assessing ecological impacts. Quantitative methods are needed for estimating entrainment and impingement mortalities, for determining fisheries stock/recruitment relationships, for estimating relative contributions of fish species while identifying compensating reserve mechanisms, and for estimating natural populations of marine and fresh water organisms.

16.1.4 Research Program Objectives

The broad objectives of this research program are to ensure (1) that the NRC can make acceptably accurate, independent assessments of the projected environmental impacts of radiological and nonradiological aquatic effluents from proposed nuclear facilities and (2) that there is an acceptable body of knowledge upon which to base environmental technical specifications. An immediate, specific objective is to provide verified data, valid predictive models, and follow-on confirmation of predictions from operating experience that will facilitate NRC compliance with the NEPA.

16.1.5 Research Program Plan

Ongoing research to verify the effects of the transport of radionuclides on sediments in rivers will continue through FY 1982. The program of study on transport and fate of pollutants in lakes, estuaries, and oceans will be continued through FY 1987.

By FY 1985, simulation studies will make available verified mathematical models of aquatic effluent dispersion and diffusion, of the transport of effluents on sediments in rivers and in coastal zones, and of the deposition and resuspension of effluent radionuclides in these aquatic systems.

The results of a comprehensive evaluation of the mathematical models used by construction permit applicants for predicting the impacts of intake structures on fisheries will be available by FY 1984. This study will correlate environmental data from operating nuclear power plants with predictions calculated by the model to measure the reliability of the method and to improve the NRC basis for judging the acceptability of predictive models.

Biological pathways through aquatic ecosystems and food chains that lead to man will be verified beginning in FY 1982; this work will continue through FY 1987. These modeling efforts will be based on research in surface-water and ground-water systems near operating nuclear facilities and will focus initially on the sediment-carried radionuclides and how these interact with aquatic biota.

Biometric data from aquatic ecosystems research will be developed in FY 1987; it will complement and supplement the existing literature and will provide bases for verified models to be used by the staff in assessing the effectiveness of regulatory guides and technical specifications at operating nuclear power stations.

16.2 Impacts of Airborne Effluents

16.2.1 Regulatory Objective

The objective of assessing the environmental impact of airborne effluent from normal operation of nuclear facilities is to provide a basis for making licensing decisions in accordance with the requirements of NEPA.

16.2.2 Technical Capabilities Required

To meet this objective, the NRC must be able to:

1. Predict dispersion and deposition of materials in the environment, based on meteorological phenomena, including physical transport of gaseous, particulate, and volatile solid emissions;

2. Formulate, audit, and enforce effective environmental technical specifications to regulate airborne emissions from normal operations of nuclear facilities;

3. Predict and measure vegetative stress, based on verified methods for monitoring the terrestrial environment;

4. Detect trends in terrestrial environmental impacts attributable to nuclear facilities and ancillary systems; and

5. Formulate and develop effective terrestrial requirements for environmental protection plans.

16.2.3 Status of Capabilities

Present NRC practice is to use the meteorology data that the applicant has provided for a site in NRC predictive models to calculate dispersion parameters at designated distances and directions from the point of release and at points of potential maximum concentration of effluents to a distance of 80 kilometers (50 mi). (The meteorological conditions used are based on a minimum of 1 year of onsite data collection.) More lengthy records of weather data from National Weather Service stations near the plant are used to determine if the onsite data are representative of the climate in the area.

The effects of releases are calculated by projecting dispersion and the resulting radiation dose to the population out to a distance of 80 kilometers (50 mi.) from the nuclear facility. The analytical method currently used by the NRC staff is basically a straight-line trajectory dispersion model, which tends to be increasingly conservative at longer distances from the site of the release. Aside from a single case where one Federal laboratory has made a long-term effort to verify a model with specific application to that site, no verified models of dispersion beyond about 10 kilometers (6 mi) are available. Straight-line trajectory models are usually used for environmental impact assessments; however, variable trajectory models are used for accident consequence assessments, taking into account effects of wind meander. Dispersion of accidental releases is covered in Section 6.2.

There is not sufficient information to allow reliable, realistic, quantitative prediction of the dispersion and long-term effects of radioactive material releases from all modes of operation during the operating life of a nuclear facility, from startup through decommissioning. Predictive models used to describe dispersion, diffusion, and deposition of atmospheric effluents to ranges of 10 kilometers (6 mi.) have only been verified over flat terrain. Little is reliably known beyond about 10 kilometers and the area of current interest extends to 80 kilometers (50 mi.). Coastal, valley, and mountain site dispersion models usually require site-specific field measurements for adequate evaluation and verification. Field measurement data are required to verify biometric models for other radionuclide pathways leading to man.

A significant body of environmental monitoring data has been collected at operating nuclear facilities. These data need to be collated, interpreted and analyzed to evaluate the predictive models used for environmental impact assessments. The predictive models describing terrestrial ecological impacts resulting from nuclear plant effluents discharged to the atmosphere need to be evaluated by collecting field data at operating plants. The relatively few years of operating history have allowed little opportunity to detect or measure either changes in ecological conditions that occur slowly or long-term trends that are developing. There is a continuing need for operating reactor impact assessments to weigh the effectiveness of technical specifications.

16.2.4 Research Program Objectives

The broad research objectives related to airborne effluents from normal operations at nuclear facilities are (1) to provide verified data and predictive models to overcome deficiencies in information and methods, (2) to improve the realism of the NRC estimates of environmental impacts required by NEPA, and (3) to verify the effectiveness of NRC technical specifications in this area.

16.2.5 Research Program Plan

Research will be initiated in FY 1982 to incorporate laboratory and field measurements to verify mathematical models of atmospheric transport, dispersion, diffusion, and deposition of effluents resulting from normal operation of nuclear facilities to a range of 80 kilometers (50 mi.) from the site, over terrains representative of U.S. mountain and coastal features; this research will reduce the amount of site-specific information required. These studies will continue through FY 1987. Beginning in FY 1982 and continuing for 3 years, data describing the movement of nonradioactive effluents released to the atmosphere from operating facilities will be compiled, collated, and analyzed to provide data for assessing synergistic effects and to verify impact assessment models. Laboratory and field studies to evaluate projections of environmental effects made in environmental statements and the effectiveness of environmental technical specifications will be initiated at operating nuclear facilities in FY 1983 and will continue through FY 1986. This research will be closely coordinated with the meteorological research described in Section 6.2.5.

16.3 Socioeconomic Impacts

16.3.1 Regulatory Objective

The objective of assessing the socioeconomic impacts of nuclear facility construction, operation, accident consequences, and decommissioning is to provide an explicit, comparable basis for judgments of costs and benefits to be derived from the proposed licensing action.

16.3.2 Technical Capabilities Required

The interpretation of the requirements for socioeconomic assessments that has evolved through public hearing and court actions requires that the NRC have the capability to predict:

1. Economic effects on local communities such as changes in land use, housing costs and availability, labor costs and availability, level of commercial activity, the tax base, and the demand for tax-supported services, including emergency response services;

2. Social effects such as changes in the quality of publicly or privately supported services, community social structure, population, policies of social institutions, and changes in the esthetic or psychological environment; and

3. (For reactors) need for power in a given service area and the relative economics of alternative means of providing the electricity-generating capacity, if it is needed.

16.3.3 Status of Capabilities

Current methods for predicting economic effects generally apply to gradual changes rather than to a single major impact on a locality, such as the establishment of a nuclear facility. In addition, the phenomena associated with these economic effects are not well understood, and large, unevaluated uncertainties exist in the predictions of economic as well as social impacts. A major problem is the lack of a generic structure for impact assessment that would ensure comparable, equitable judgments. Although NRC socioeconomic impact assessments should be improved in almost all areas of local economics, those areas in which public criticism is strongest are land use, fiscal impacts on local governments, and economic effects of nuclear accidents. Aspects of community environments other than economics are more difficult to analyze quantitatively. In these noneconomic areas, the ability to forecast and evaluate impacts is less well developed for the purposes of regulatory decisionmaking. Despite the difficulty in placing impacts into a cost-benefit framework, rapid progress is being made in this field. A major problem is to provide for timely application of research techniques to environmental impact assessments by the NRC. Examples of specific areas of socioeconomic impact analyses that have weak methodological or data bases are:

1. Impacts on land-use patterns and urban systems, including changes in traffic patterns;

2. Visual impacts of closed-cycle cooling systems;

3. Nonradiological consequences of accidents such as adverse impacts on the local economy and community institutions;

4. Impact on recreational use of nearby land and water resources; and

5. Impact of fogging and icing on local agriculture and other types of land use.

Rapid increases in the price of electricity and changes in the economics of nuclear versus coal as a means of producing electrical power have resulted in a much wider range of forecasts of the need for new electric-power-generating capacity. The NRC's capability to judge the adequacy of these forecasts--whether they are provided by license applicants or developed independently by contractors--must be kept current through access to the most advanced modeling systems.

16.3.4 Research Program Objectives

The broad objectives of this program are to:

1. Provide more quantitative methods for predicting socioeconomic changes such as labor force migration, impacts on community and institutional structures, land-use and value changes (consistent with the NEPA requirements for environmental impact assessments);

2. Provide measured data for testing, verifying, and improving the predictive methods used by applicants and by the Commission staff;

3. Assess the socioeconomic consequences of a variety of classes of nuclear accidents; and

4. Develop and maintain adequate modeling capability for assessing the need for power at the State and regional level.

16.3.5 Research Program Plan

Presently underway and planned to continue through FY 1982 are postlicensing studies of the socioeconomic impacts of nuclear power station construction and operation. These studies will determine the extent and scope of effects at existing stations and will assess the accuracy of previous impact forecasts. It is also necessary to analyze errors in predictions and to recommend improved forecasting procedures based on these analyses.

Modeling construction-labor force migration patterns (based on independent variables for which data are easily obtainable) and verifying the predictive capabilities of models with data independently developed will continue through FY 1982. This research will enable the NRC to make more accurate forecasts of the local population increases--and associated socioeconomic effects--that result from nuclear power station construction and operation.

Beginning in FY 1982, the research program will assess the impact of the TMI accident on the TMI area--as well as the impact of other possible accident scenarios on other communities in the vicinity of nuclear power stations--to determine the adverse impacts on the local economic, social, and institutional structures.

The program will extend the cost/benefit methodology for quantifying environmental impacts to a wider range of environmental effects of nuclear power stations, specifically air quality impacts on agriculture, changes in recreational use of land and water resources, the visual effects of alternative closed-cycle cooling systems, and the impact of changes in traffic patterns. Common quantitative-measurement data on socioeconomic impacts will be developed to allow comparisons of different types of impacts in comparable units. A pilot study is currently underway; additional work will start in FY 1982 and extend through FY 1985.

Another program will be developing modeling systems to perform independent, confirmatory assessments of the State and regional need for power forecasts, incorporating the latest quantitative techniques into the modeling system. This effort is ongoing and is planned to last through FY 1985. It will develop analytical models to permit NRC staff to assess forecasts of the relative economics of alternative means of providing electricity-generating capacity

Acronyms and Initials

ACI	American Concrete Institute
ACRR	Annular Core Research Reactor
ACRS	Advisory Committee on Reactor Safeguards
AE	Acoustic emission
AEC	Atomic Energy Commission
AEOD	Office of Analysis and Evaluation of Operational Data, NRC
AGNS	Allied-General Nuclear Services
ALARA	As low as reasonably achievable
AOT	Anticipated operational transients
APS	American Physical Society
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
ATWS	Anticipated transient without scram
BCL	Battelle Columbus Laboratory
BE	Best estimate
BEACON	Best Estimate Advanced Containment Code (computer code)
BE-EM	Best estimate-evaluation model
BMFT/FRG/KfK	Kernforschungszentrum, Federal Republic Germany
BNL	Brookhaven National Laboratory, NY
BRH	Bureau of Radiological Health
BWR-TRAC	Boiling Water Reactor-reactor computer code
CAIS	Coded aperture imaging system
CCTF	Cylindrical Core Test Facility
CDS	Conceptual design study
CE	Combustion Engineering
CEA	Commissareat a l'Energie Atomique, France
CEP	Comprehensive evaluation program - an onsite review of safeguards performance vulnerabilities at each major licensed fuel cycle facility

CFR	Commercial fast reactor, United Kingdom
CHF	Critical heat flux
COBRA	Computer code aimed at analyses of PWR plants featuring upper head injection
COCORP	Consortium for Continental Reflection Profiling
COMMIX	Transient, three-dimensional thermal hydraulics code
CONTAIN	Containment analyses code
CONTEMPT	Containment Temperature and Pressure Computer Code
CORCON	Advanced computer code which models the phenomena which would occur if molten core material penetrates a reactor vessel and contacts concrete
CORRAL	A computer code used to model the behavior of fission products in the containment atmosphere
CRAC	Calculation of Reactor Accident Consequence (code)
CRBR	Clinch River Breeder Reactor
CSTF	Containment Systems Test facility
DOD	Department of Defense
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor, Idaho
ECCS	Emergency core cooling system
EG&G	Electronics Firm - Operating Contractor for INEL
EIS	Environmental impact statement
EM	Evaluation model
EMP	Electromagnetic pulse
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ESMERALDA	Gas cooled reactor computer code
ESSOR	EURATOM's 50 mw(th), organic cooled, heavy-water moderated, experimental power reactor at Ispra, Italy
EURATOM/ISPRA	Joint Research Center, Commission of the European Communities, Italy
EXMEL	A model for fuel and clad melting developed at Stuttgart

FAST	Fuel aerosol simulant test
FBR	Fast breeder reactor
FCI	Fuel coolant interaction
FD	Fuel disruption
FEMA	Federal Emergency Mobilization Agency
FERC	Federal Energy Regulatory Commission
FFTF	Fast Flux Test Facility
FITS	Fully instrumented test series
FLECHT-SEASET	Full length emergency cooling heat transfer-separate effects and systems effects tests
FMEA	Failure mode effects analysis
FRAP	Fuel rod analysis program
FRAP-T	Fuel rod analysis program transient
FRAPCON	Computer code used for steady-state analysis of fuel rod response during normal reactor operation
FRG	Federal Republic of Germany
GALE	Gaseous and liquid effluent (evaluation code)
GDC	General design criteria
GEIS	Generic environmental impact statement
GRASS-SST	mechanistic model used to predict the release of fission products from high-temperature fuel
GRSR	General Reactor Safety Research, RES, NRC
HAARM	A mechanistic aerosol behavior code
HAZ	heat affected zone
HCDA	Hypothetical Core Disruptive Accident
HDR	Heissdampfreaktor - a decommissioned steam reactor in West Germany used to conduct reactor safety experiments
HELB	High energy line break
HPIS	High pressure injection system
HRR	High-reactivity-ramp rate

ICS	Integrated control system
IE	Office of Inspection and Enforcement, NRC
IFD-NDE	Internal-friction-damping nondestructive examination
IGSCC	Intergranular stress-corrosion cracking
INEL	Idaho National Engineering Laboratory
IREP	Interim Reliability Evaulation Program
IRT	Research reactor, natural uranium, graphite, 5.5 MW(th), Kurchatov Institute, Moscow, USSR
ISI	Inservice inspection
ISP	International Standard Problems
ITV	Intermediate heat vessel
IWMG	Interoffice Waste Modeling Group
JAERI	Japanese Atomic Energy Research Institute
JOYO	Japanese Fast Breeder Test Reactor
K-FIX	Thermal Hydraulic Component Computer Code
LAPUR	Reactor Kinetics Computer Code
LASL	Los Alamos Scientific Laboratory
LBT	Long-bundle tests
LER	licensee event report
LMF	Large-melt facility
LMFBR	Liquid metal fast breeder reactor
LOCA	Loss of coolant accident
LOF	Loss of flow
LOFT	Loss of fluid test
LRRP	Long Range Research Plan -- a 5-year projection of research activities planned by the NRC Office of Nuclear Regulatory Research
MARCH	Computer code used to analyze core meltdown phenomena
MARCH/CORRAL	code used for core-melt accident assessment

MC&A	Material control and accounting
MEKIN	MIT 3D Kinetics Computer Code
MF	Molten fuel
MRBT	Multi-rod burst test
MSLB	Main steam line break
MTA	Mobile test assembly
MULTI-AEROS	Mechanistic aerosol behavior code
NBS	National Bureau of Standards
NDE	Nondestructive evaluation
NEPA	National Environmental Policy Act
NESC	National Energy Software Center
NMSS	Office of Nuclear Material Safety and Safeguards, NRC
NOAA	National Oceanographic and Atmospheric Administration
NPRDS	Nuclear Plant Reliability Data System
NREP	National Reliability Evaluation Program
NRU	Canadian natural-uranium, heavy-water moderated and cooled test reactor, Chalk River, Ontario
NSPP	Nuclear Safety Pilot Plant, Oak Ridge TN
OBE	Operating basis earthquake
OECD/CSNI	Organization for Economic Cooperation and Development Committee for the Safety of Nuclear Installations
OPTRAN	Operation transient
ORNL	Oak Ridge National Laboratory, TN
PBE	Prompt burst experiments
PBF	Power Burst Facility
PCA	Pool critical assembly
PCI	Pellet/cladding interaction
PHENIX	French Fast Breeder Power Reactor
PNC	Power Reactor and Nuclear Fuel Development Corporation, Japan
PPPG	Policy, Planning and Program Guidance

PRA	Probabilistic risk assessment
PRDMF	Power reactor dosimetry measurement facility
PSF	Pool side facility
PWR-BDHT	Pressurized water reactor-blowdown heat transfer
QTE	Qualification testing evaluation
QUICK	Mechanistic aerosol behavior code
RAMONA	Best estimate computer code used for analyses of BWR transients involving detailed reactor kinetics effects
RBE	Relative biological effectiveness
RCS	Reactor coolant system
RELAP	Best estimate computer code used to analyze pressurized water reactor accidents and transients
RETRAN	Reactor Transient Analysis Computer Code
RIA	Reactivity initiated accident
RIL	Research Information Letter
RPV	Reactor pressure vessel
RSS	Reactor Safety Study
SAFT-UT	Synthetic aperture focusing technique for ultrasonic testing
SANDY	Sandia fuel dynamics program
SAREF	Fast reactor experimental loop in the Engineering Test Reactor, Idaho
SASA	Severe Accident Sequence Analysis
SCALE	Computer Code
SCDAP	Severe core damage analysis package
SCTF	Slab core test facility
SD	Office of Standards Development, NRC
SDMF	Surveillance dosimetry measurement facility, ORNL
SEP	Systematic evaluation program
SIMMER-II	Fast reactor transient analysis computer code
SMACS	Structural Mechanical Computer Code
SRA	Systems reliability analysis

SRP	Standard review plan
SSC	Super system code
SSE	Safe shutdown earthquake
SSI	Soil structure interaction
SSMRP	Seismic safety margins research program
SSTF	Steam Sector Test Facility
SUPER-PHENIX	Commercial sized French Fast Breeder Power Reactor
TDC	2-dimensional finite cylinder transport computer code
TFBP	Thermal fuel behavior program
TLD	Thermoluminescence dosimeter
TLTA	Two Loop Test Apparatus
TOP	Transient Over Power
TRAC	Computer code used to model core reflood and quenching
TRAP-MELT	Model used to analyze fission-product behavior within an LWR primary system under accident conditions up to and including fuel meltdown
UHI	Upper head injection
UPTF	Upper Plenum Test Facility
USI	Unresolved safety issues
USINT	Computer code for uranium/sodium/concrete interactions
USSP	US Standard Problems
WRAP	Water reactor analysis package
WRAP-BWR-EM	Water reactor analysis package-boiling water reactor evaluation model (computer code)
WRAP-PWR-EM	Water reactor analysis package-pressurized water reactor evaluation model (computer code)
WRSR	Water Reactor Safety Research, RES
ZONE	Mechanistic aerosol behavior code

Reports and Regulatory Guides

Regulatory Guides*

1.3	Assumptions Used for Evaluating Light Water Cooled Nuclear Power Plant Siting Distances for Purposes of 10 CFR 10
1.4	Assumptions Used for Evaluating Potential Consequences of LOCA for BWR
1.7	Control of Combustible Gas Concentration in Containment Following a LOCA
1.46	Protection Against Pipe Whip Inside Containment
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems
1.89	Qualification of Class IE Equipment for Nuclear Power Plant
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plant and Evirons Conditions During and Following an Accident
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
1.120	Fire Protection Guidelines for Nuclear Power Plants
1.133	Loose Part Detection Program for Primary System of Light Water Cooled Reactor
1.145	Atmospheric Dispersion Models for Potential Accident Consequence

NUREG **

NUREG-75/067	Cracking Incidence in Austentic Steel Piping of BWR Plants
NUREG-0531	Investigation and Evaluation of Stress Corrosion Cracking in Piping of LWR's
NUREG-0578	TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations
NUREG-0585	TMI-2 Lessons Learned Task Force: Final Report
NUREG-0657	Review and Evaluation of the NRC Safety-Research Program for FY 1981
NUREG-0660	NRC Action Plan Developed As a Result of the TMI-2 Accident
NUREG-0669	Fixed Site Physical Protection Upgrade Rule Guidance Compendium
NUREG-0676	LOCA Computer Code Assessment
NUREG-0691	Investigation and Evaluation of Cracking Incidents in PWRs
NUREG-0699	Comments on the NRC Safety Research Program Budget for Fiscal Year 1982
NUREG-0737	Clarification of TMI Action Plan Requirements

NUREG/CRs **

NUREG/CR-1169	Safeguard Vulnerability Analysis Program (SVAP)
NUREG/CR-1233	Structured Assessment Approach (SAA)
NUREG/CR-1245	Safeguards Network Analysis Procedure (SNAP)
NUREG/CR-1246	Safeguards Automated Facility Evaluation (SAFE)
NUREG/CR-1409	Summary of Zion/Indian Point Study
NUREG/CR-1410	Report of Zion/Indian Point Study, Vol. 1
NUREG/CR-1411	Report of Zion/Indian Point Study, Volume 2

Other Reports***

WASH-1400	Reactor Safety Study

* Available for purchase from the NRC/GPO Sales Program, Regulatory Guide Accounts, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

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16. ABSTRACT (200 words or less)

This Long Range Research Plan (LRRP) was prepared by the Office of Nuclear Regulatory Research (RES) to provide a document which will assist the NRC in coordinating its research planning with the budget cycles. The plan would aid the Commission in establishing appropriate priorities and assuring effective utilization of NRC resources. The 5-year Plan (FY 1983-1987) is being developed by RES with recommendations from the user offices (licensing, standards, inspection and enforcement) on their current or perceived further research needs and identifies regulatory issues and lays out programmatic approaches for research to be done as part of the resolution of these issues. The LRRP discusses research in three major areas: Reactor Safety Research, Systems and Reliability Research, and Fuel Cycle and Environmental Research. The plan will be updated at least annually as tasks are completed and new requirements and guidance are provided by the Commission.

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