NUREG-0784

Long Range Research Plan

FY 1984-FY 1988

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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

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RECIPIENTS OF NRC'S LONG-RANGE RESEARCH PLAN

Enclosed for your information is a copy of the Nuclear Regulatory Commission's Long-Range Research Plan (LRRP) for fiscal years 1984 through 1988 (NUREG-0784). This is the NRC's second published 5-year research plan.

The purpose of the plan is to aid the Commission in agency program planning. The LRRP lays out programmatic approaches for research to help resolve regulatory issues.

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

Sincerely,

Frank J Arsenault/for
Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosure: NRC's Long-Range
Research Plan

NUREG-0784

Long Range Research Plan

FY 1984-FY 1988

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1. INTRODUCTION

The Nuclear Regulatory Commission's mission--regulation to ensure that civilian activities involving the use of nuclear materials and facilities are conducted in a manner consistent with 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 for 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 complete information available for this purpose. In many cases, the information and data needed must be obtained by research.

Most of NRC's regulatory responsibility involves large facilities such as reactors and fuel cycle processing plants for which full-scale safety tests are neither practical nor cost effective. Therefore, the principal tool used by NRC in the regulatory process is complex safety analysis computer codes that are verified through comparative analysis both of planned transients and accidents in experimental facilities and of the normal or upset operation of licensed facilities. The development, verification, and assessment of these codes is a major function of the research program.

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 TMI-2 as a basis for reassessing technical criteria with the goal of facilitating Commission actions to improve the regulatory process.

While the staff should legitimately 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.

The Commission has directed that a long-range research plan (LRRP) be developed to ensure that agency resources are being properly directed toward areas of importance to the licensing and inspection processes. The research plan is to be revised and updated annually and subjected to agencywide review. It is intended that this plan should 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 is being updated annually as tasks are completed and new requirements and guidance are provided by the Commission, either through their Policy and Planning Guidance (PPG) documents, by direct comment on the LRRP, or by other means.

RES, in preparing this plan, has received its primary direction from the Commission via the 1982 Policy and Planning Guidance document (NUREG-0885). That document states that:

> "The research program will continue to emphasize support of the safety of operating reactors and other operating facilities. The purpose of the research program is to assist in establishing regulations for existing and future facilities." It also states that "the first priority for NRC research efforts will be light water reactor safety."

These concepts have been the basis and foundation for the research program that is described in this plan.

The major objective of the NRC research program is to provide the understanding of phenomenology and the verified analytical methods to permit identification and well-founded realistic (or best-estimate) analysis of important accident sequences and their consequences. To this end, the research program consists of a mixture of experimental work and code development work that is aimed at understanding complex system transients. The basis for planning future research, in order to obtain cost effectiveness, is smaller-scale experiments providing data that are applied to nuclear plant safety through carefully scrutinized analyses using throughly checked out codes. Big complex inherently atypical facilities tend to yield few data points, and those of doubtful applicability. Other major objectives are to provide the methodology to make more effective use of probabilistic risk assessment in the regulatory process and to improve confidence in the data base for risk assessment. This combination of experiments, code work, and risk analyses puts all of this together in terms of deciding what are the significant things that should be taken into account in the regulatory process.

The pre-TMI regulatory practice of attempting to characterize a wide range of possible accidents with one or several "limiting case" sets of accident assumptions not only may lead to unwarranted conservatism but may result in design requirements that inhibit the ability to cope with actual accidents or precursors of potential accidents.

This effort is directed toward several end uses:

- tter understanding of system response and safety margins provided by plants in real-world accidents and transients.
- Analysis and appreciation of needed operational capabilities, operator training requirements, and human factors.
- Knowledge of fuel behavior during transients and severe accidents.
- Knowledge of fission product pathways, deposition mechanisms, and ultimate destinations.
- Better use of probabilistic risk analysis with better ability to differentiate relative safety significance of regulatory issues or prospective requirements.
- Understanding of a sound basis for the design of mitigating systems.

The Commission has indicated in the PPG paragraph quoted above that RES should emphasize support of the safety of operating reactors and also assist in establishing regulations for future facilities. We believe that we have established the proper balance in achieving this goal. The program that is described in this plan covers research that affects plants now operating, those being built, and those under review. Also, there are some elements of the research that pertain to features that could only be incorporated in new plants not yet designed. These constitute a relatively small part of the program, however.

Many factors in addition to the Commission guidance mentioned above are involved in establishing the final selection of programs and projects to be submitted for Commission, OMB, and Congressional approval. Most prominent among these factors are user needs as submitted to RES by NRR, NMSS, and IE. The ACRS in its annual review of the RES budget provides technical insights on areas where they believe more or less emphasis is needed. Within RES we have established an ordering of research emphasis that is used as a basis for establishing general program priorities. This is:

<u>Areas of Particular Emphasis</u>

Identification and understanding of complex system transients as a basis for application of reliability and risk analysis, improved man-machine interface, improved control system design, better operator training, design of improved safety systems, and defining conditions that may lead to fuel damage.

- Fuel damage and fission product behavior over a wide range of transients and accidents.

- Small-scale specific tests of particular phenomena.

- Human factors, operator training, man-machine interface.

- Applications of codes to regulatory problems.

- Improved safety systems.

- Pressure boundary safety, operability of equipment, integrity of electrical connections, and structural integrity of aging plants.

- Waste management.

- Development of nondestructive examination techniques.

- Application of risk-assessment techniques to better focus the regulatory process on safety issues.

- Approaches to decay heat removal and improved containment.

- Development of basic data and technical information as a base for Commission policy decision (rulemaking, etc.).

<u>Areas In Which Program Goals Have Generally Been Achieved - Being Reduced</u>

- Large-break LOCA/ECCS issues, Appendix K.

- Safeguards research.

- Large integral facilities.

- Open-ended code development.

- Fuel behavior under normal operating transients.

- Research to confirm margins inherent in existing regulations.

In the way of fiscal guidance the Commission states:

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

RES is following this guidance. The overall trend of planned expenditures is down as a result of a concentrated effort on the part of RES to phase down LOCA and related thermal-hydraulic investigations. The termination of LOFT testing in early FY 1983 reduces the resource requirement for that program from over

$40 million per year for FY 1982 and prior years to $15 million in FY 1983 and decreasing amounts in the succeeding few years until the NRC's obligation to return the site to a safe condition is satisfied either by decommissioning or transferring to others. The net result of reductions in LOCA and transient and LOFT research, plus holding most other research programs to constant expenditures, is a projected RES budget that is projected to decline at an average rate of over 8 percent per year from FY 1984 through FY 1988.

A significant effort is planned during the next few years to support the reassessment of the regulatory treatment of severe accidents. This work includes studies of transients leading to fuel or cladding damage, the behavior of damaged fuel, fuel melt, fission product release and transport, and severe accident mitigation concepts. This broad program is being coordinated with NRR and ACRS, and with Commission guidance, to keep pace with the current thinking regarding rulemaking. A severe accident research plan, separate and more detailed than this plan, is being reviewed and modified on a schedule roughly parallel to this plan.

Increased emphasis during the planning period will also be given to developing further improvements and probabilistic risk assessment techniques that RES will apply to research planning efforts and to assessing relative priorities of regulatory activities. Use of these methods will also be increased by the NRC in the licensing review process. This effort will be augmented by increased use of systems reliability analyses and evaluation, including human error rate data analysis.

NRC's current breeder research program funding amounts of $5 million in FY 1982 and $10.5 million in FY 1983 are directed at providing information needed for the CRBR construction permit and operating license reviews. The RES and NRR staffs have had numerous discussions with the Department of Energy (DOE) to ensure that our programs are coordinated and that all the information necessary for licensing will be available. The task of developing the supporting base technology for the LMFBR and specifically for CRBR is the responsibility of DOE. The NRC expects that the DOE base technology program will provide the body of research necessary to support the applicant's case. If commercialization of a broad LMFBR program and supporting fuel cycle were to proceed, there will be a need for both additional and more expanded programs.

The Commission has decided to develop a safety goal and related safety guidance with initial emphasis on individual and societal risks that might arise from reactor accidents. The purpose of this project is to develop a general approach to risk acceptability and safety-cost tradeoffs and, to the extent possible, to specify qualitative safety goals and quantitative safety guidance and standards for review of rules and practices. Direct assistance to this Commission project is being provided by the RES risk analysis staff as described in Chapter 10, "Systems and Reliability Analysis," of this plan.

The Commission has asked that the radioactive source term should be reassessed by early 1983 to provide a basis for siting policy improvement. RES is conducting studies that are expected to provide a reassessed source term on that schedule. During the period of this plan, research to confirm and support or indicate a further reassessment will be conducted.

Regarding the pressurized thermal shock problem about which the Commission has expressed concern, RES is performing analyses and conducting experiments as described in Chapters 2, "LOCA and Transient Research," and 6, "Reactor and Facility Engineering," of the plan. The pressurized thermal shock problem is just a part of the overall primary-system-integrity area of concern. The research program is constantly analyzing other parts such as the conduct of an extensive steam generator research program and an evaluation of nondestructive examination techniques.

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, rulemaking, 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 among 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.

The plan is intended to constitute the initial broad basis for the endorsement of the RES program by the user offices. As the user offices participate in the budget review process, the RES program for that fiscal year (in this case, FY 1984) is reviewed at a more detailed program level; and finally the user offices are given a chance to react to the individual projects as the work statements and contracts are sent to them, just prior to project commitment. RES will confer with the appropriate user offices whenever it intends to make any significant changes from the programs as described in this plan.

Finally, a few words should be included about the organization of the LRRP. This year's plan reflects the new decision units resulting from the consolidation of RES and the Office of Standards Development. A crosscut is provided (Figure 1.1) to tie together major subject areas. To the extent practical, the standards activities have been included in this year's plan.

There is an introduction to each decision unit summarizing the contents of the decision unit. The sections of the decision units contain a statement of the issue that is being addressed by the research or standards effort; the objective of the program is summarized; the relationship to other programs within RES, to those being conducted by other U.S. Government agencies and U.S. industry, and to programs being conducted in other countries is discussed; the FY 1982 and FY 1983 programs and any necessary history are discussed under "Background and Status"; and the planned research and standards effort for FY 1984 through FY 1988 is discussed in the "Research Program Plan" section.

LRRP MAJOR REGULATORY/RESEARCH TOPICS
(PERTINENT CROSS SECTIONS)

CHAPTERS	2 LOCA & TRANS.	3 LOFT	4 ACCIDENT EVAL. & MITIGATION	5 ADVANCED REACTORS	6 REACTOR & FACILITY ENG.	7 FACILITY OPS. & SAFEGDS.	8 WASTE MANAGE.	9 SITING & ENVIR.	10 SYS. & RELIABILITY ANALYSIS
AGING					6.1-6.6				
HUMAN FACTORS						7.1			10.2
INSTRUMENTATION					Sec.6.5.4 6.5.5	Sec.7.2			
LOCA CODE ASSESS.	2.1-2.4	All							
PRES. THERMAL SHOCK	2.1,2.2 2.4				6.2				10.1
SEV. ACCD/CORE DMG.	2.4		All	5.1		7.1,7.2 7.4		9.2	10.1
STEAM GENERATOR	2.1, 2.4				6.2-6.3				
FUEL CYCLE LICENS.					6.9	7.3		9.1	10.3
MATERIALS LICENS.						7.3		9.3	10.3
TRANSPORTATION					6.4				10.3
WASTE MGMT.					6.11		All		
DECOMMISSIONING		3.4			6.10	7.3	8.2		
EMERG. PREPAREDNESS						7.4		9.1	10.1
FIRE, FLOOD & EARTHQK.					6.5,6.8			9.2	10.2
RADIATION PROTECT.						7.3		9.3	
SFGDS/SABOTAGE						7.5			10.2

FIGURE 1.1

An effort was made to accommodate the comments of others on last year's plan (ACRS, the United Kingdom Safety and Reliability Directorate, INEL, etc.) where possible.

2. LOCA AND TRANSIENT RESEARCH

This decision unit comprises the research devoted to providing data and predictive models and codes to understand the response of the reactor system and key components to loss-of-coolant accidents (LOCAs) and other transients that are characterized largely by significant changes in coolant behavior and transport.

The five elements of this decision unit include:

1. The integral systems program, which consists of the Semiscale facility at the Idaho National Engineering Laboratory (INEL) and the Full Integral Simulation Test (FIST) facility at the General Electric Company in San Jose, California. Semiscale was designed to study systems response to transients by using a small-scale, electrically heated replica of the major elements of a nuclear steam supply system. The FIST facility will be used to investigate anticipated transients without scram (ATWSs) and system transients in a BWR configuration.

2. The separate effects and model development program, consisting of numerous elements at diverse laboratories to investigate key effects that occur when the coolant flow in a reactor undergoes a transient. Mathematical models of these effects are created for use in predictive codes so that they may be used to analyze full-scale plants.

3. The 2D/3D program, which is a tripartite international program coordinated among the U.S., Japan, and the Federal Republic of Germany (FRG), to test the capability of predicting the behavior of coolant in full-scale replicas of key parts of the core and primary system under LOCA conditions. The U.S. program is concerned with developing and providing instruments to measure the flows in the tests and with performing complex test prediction and analysis.

4. The code development and application effort, which is a program to develop codes capable of predicting the behavior of full-sized nuclear plants under the accident conditions characterized as LOCA and other transients and to carefully assess the actual accuracy of the codes over the intended range of application.

5. The fuel behavior under operational transients task, which is a program to develop the needed understanding of how the fuel behaves during the prescribed transients so as to be able to understand the flow of heat from the fuel to the coolant, the resistance of the fuel bundles to coolant flow, and the mechanical response of the fuel to transient power and cooling effects. This program will be completed in FY 1985.

2.1 Integral Systems

2.1.1 Issue

Regulatory review and Commission rulemaking require experimental data and analytical models that address LWR transients. Many specific areas of research need are identified in the TMI Action Plan (NUREG-0660 and -0737) and periodic ACRS reviews. Others are covered in numerous NRR user need letters to RES.

2.1.2 Research Program Objective

The objective of the integral systems program is to provide information for:

1. Understanding adverse LWR transients, potential operator reactions, recovery modes, and special features,

2. Rulemaking support derived from the test results that provide an insight into degraded core cooling conditions,

3. Assessing plant monitoring systems, such as liquid-level detectors, thereby providing a means to assess new plant safety features, operator interpretation, and operator guidelines, and

4. Developing LWR models for analyzing transients and assessment of LWR systems codes.

These objectives will be achieved through:

1. Continued use of the Semiscale facility to examine PWR accident scenarios,

2. Use of the FIST facility to investigate BWR transients,

3. Continued application of the RELAP 5 code to Semiscale to support experimental operation and application of experimental results, and

4. Similar application of TRAC-BWR to FIST.

2.1.3 Relationship to Other Programs

The Semiscale and the FIST experimental facilities will be the only integral thermal-hydraulic facilities available to NRC during this 5-year period. There are integral facilities in other countries that are or will be in operation. These include PKL (in Germany), LOBI (in Italy), and ROSA III and ROSA IV (in Japan). The tests, which are coordinated among facilities, produce data generally complementary to the Semiscale and FIST data (rather than duplicative). Further, NRC planning includes consideration of the test plans in those facilities.

The FIST program is a cooperative program jointly funded by NRC, the Electric Power Research Institute (EPRI), and the General Electric Company. In addition to the experimental simulations, a strong analytical effort supporting the TRAC-BWR computer development is included. TRAC-BWR will be used to aid in test planning and will be assessed using data from the FIST facility.

The Semiscale program is funded by the NRC. Included in the effort is development of the RELAP 5 computer program, which is used in a manner similar to TRAC-BWR (TRAC-BWR is separately funded).

Both programs are closely related to many NRC programs, as outlined in other sections.

2.1.4 Background and Status

The FIST is designed to simulate BWR behavior. It is an upgrade of the Two-Loop Test Apparatus (TLTA) facility previously used for large-break LOCA simulations and limited small-break LOCA tests. The upgrade in FY 1982 includes the capability for a wide range of BWR transient simulations. In FY 1983, the first phase of testing will have been completed. This test phase will include a limited number of small-break LOCA and ATWS simulations.

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 so that a wide range of experiments can be performed, and capable of assessing accident-monitoring instrumentation proposed for use in nuclear power plants.

The Semiscale program has provided data and analyses since 1965 for many PWR configurations. Many LOCAs have been simulated, principally in support of code development and assessment, rulemaking, and licensing concerns. Recent attention has been focused on realistic evaluation of issues and response to NRC needs, including steam generator tube breaks (1977), TMI (1979), support to NRR audits (1979-1980), pumps on/off (1980), station blackout (1980), small LOCA with and without upper-head injection (UHI) (1981), and natural convection (1981). The FY 1982 plan includes a three-test series covering intermediate breaks, a six-test series involving steam generator secondary-side upsets (steamline breaks, feedline breaks, loss of feedwater control), and a seven-test series of events initiated by loss of offsite power. A series of ten tests studying transients that involve steam generator tube breaks and another test series of selected transients involving secondary-side upsets will be conducted in FY 1983.

Testing of the Westinghouse vessel liquid-level instrument will have been completed during FY 1982, and tests of other vendor devices will be run as they can be "piggy-backed" onto the test schedule. (Babcock and Wilcox (B&W) is presently studying the possibility of using Semiscale to test some of their instrumentation.)

2.1.5 Research Program Plan

2.1.5.1 FIST Facility

The FIST facility is a volume and power-scaled thermal-hydraulic system test. The major components are illustrated in Figure 2.1. All major components are of full height compared to the BWR. The full height of the FIST facility represents a major improvement over TLTA because of the importance of both elevation heads and level in determining BWR transient response.

Plans for FY 1984 include continuing evaluation of data from the first phase of testing and use of the data to assess TRAC-BWR. The facility capability and

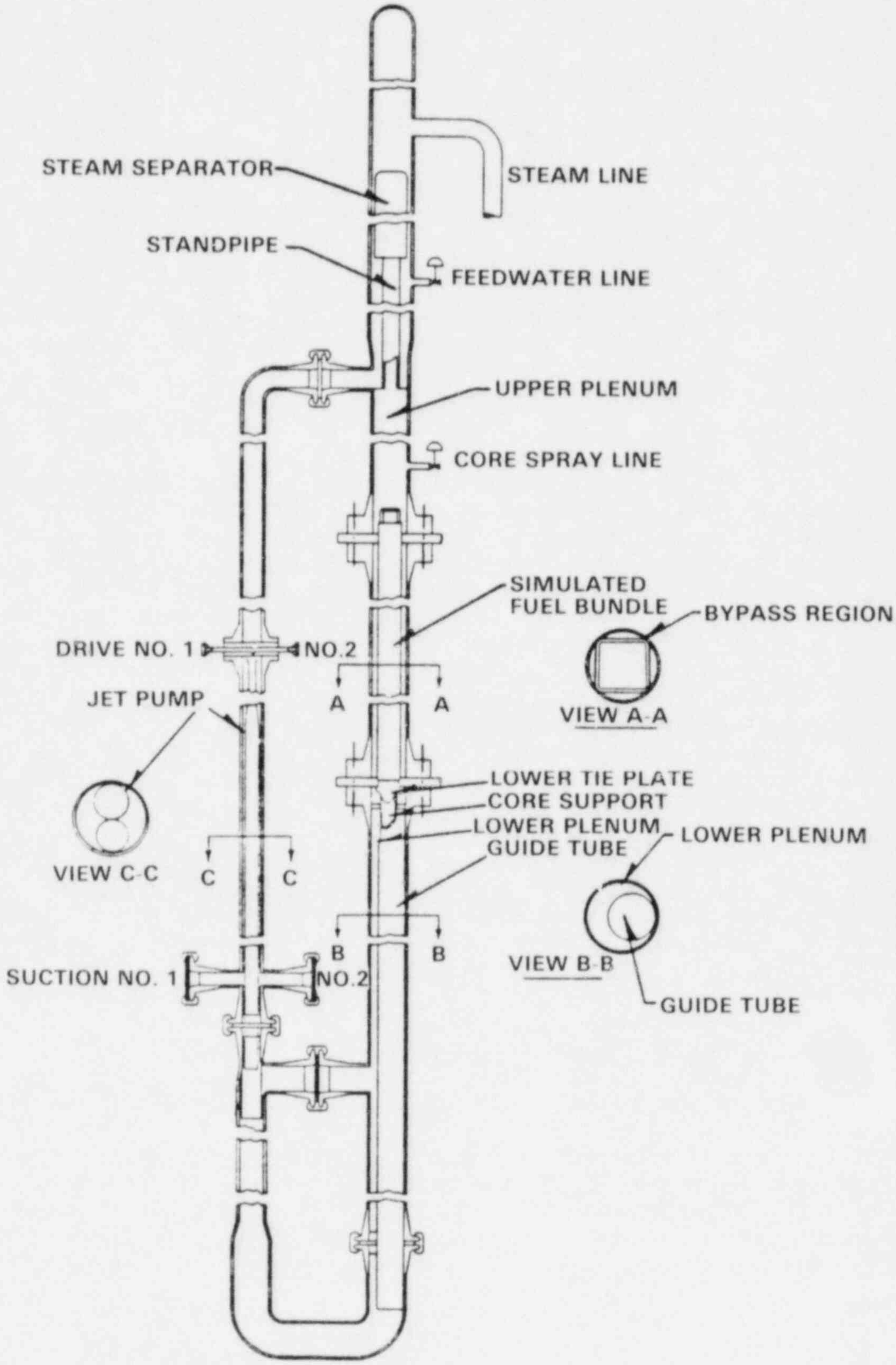

Figure 2.1 BWR Full Integral Simulation Test (FIST) Facility

transient test plans will be evaluated based on the first phase tests, and modifications will be made as required. A second phase of testing will start in mid-FY 1984 and continue through early FY 1985.

Tests typically will involve ATWS, feedwater transients, small LOCA, steamline break, natural circulation/core uncovery, and BWR instrumentation (level detector) response. All testing and analysis efforts are scheduled to be completed in FY 1985.

2.1.5.2 Semiscale

The Semiscale system is a volume and power-scaled thermal-hydraulic test facility designed to study transient and LOCA events typical of 4-loop PWRs. The main facility components, which are equal in height to PWRs, are illustrated in Figure 2.2.

The Semiscale program is, by design, a flexible program capable of quick response to NRC problems. To provide this response, test series are arranged in order of decreasing priority. Since later series may change, based on knowledge gained in prior tests, other information (from foreign tests, for example), and recent priorities, test series are planned in detail for approximately 1 year in advance and in general for series further in the future. High-priority recent requests are then worked into the program as needed with little effort wasted in replanning.

FY 1983 work will encompass completion of the loss of electrical power test series, study of the steam generator tube breaks, and initiation of an additional series concerned with transients involving secondary-side upsets. The longer-range plan includes:

FY 1984	Complete secondary-side upset series; conduct pump suction breaks and off-design condition breaks, ATWS; continue NRR support.
FY 1985	Conduct intermediate- and large-break UHI series; perform tests (where suitable) originally proposed for LOFT; continue NRR support.
FY 1986	Complete LOFT substitution series; study postaccident recovery; continue NRR support.
FY 1987	Study severe transients with significant potential of leading to core damage; evaluate vessel breaks; continue NRR support.
FY 1988	Conclude test program.

NRR has requested (NRR User Need Letter 81-16) RES to conduct a design and option study for a B&W and Combustion Engineering (CE) plant simulation, Semiscale MOD 5. This study will include development of detailed objectives and a preliminary test matrix. Funding for the construction of Semiscale MOD 5

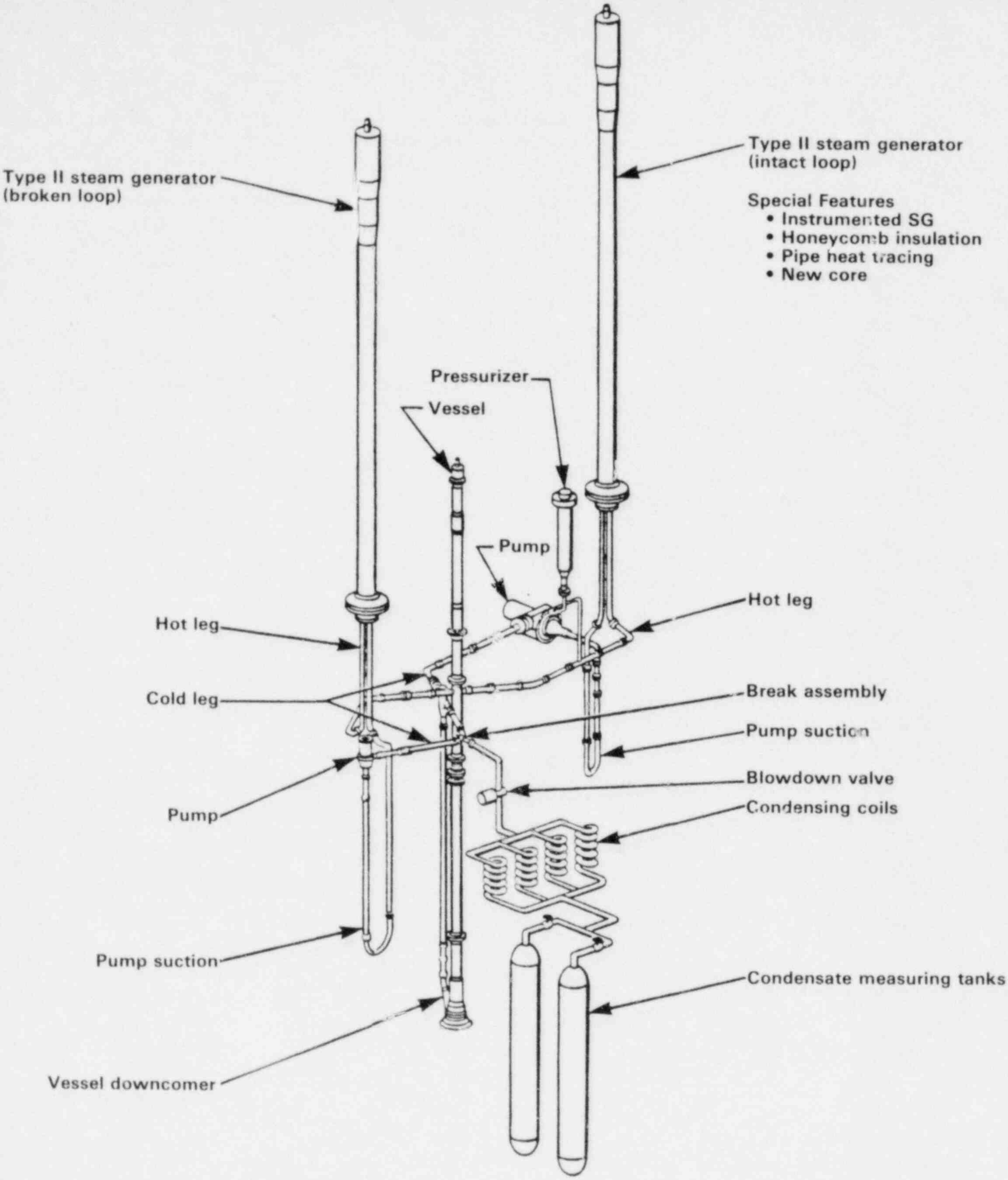

Figure 2.2 Semiscale Mod 2A

cannot be decided until the design and option study has been completed. Consequently, the 5-year plan does not include detailed plans for MOD 5. The expected date for the completion of the design and option study is June of FY 1982 and the funding decision should be made prior to the end of FY 1982.

As previously mentioned, development of the RELAP 5 code is included in the Semiscale program since development has been closely associated with experimental programs. This association is being continued since it has proved valuable in the resultant code capability. However, RELAP 5 code usefulness and application broadened rapidly during 1982 to include a number of NRR- and RES-funded PWR and BWR analyses. RELAP 5 emphasis was consequently shifted to provide high priority to user support in these areas, and development of features needed by these NRC users was emphasized. This application is expected to continue in 1983 as the PWR transient version of the code is completed and released to the public. The future schedule includes:

FY 1984	Continue support activities; maintain a validation program and incorporate new research information as needed; continue working with NRR and other users to ensure user convenience and computer compatibility; incorporate plant analyses capability; and publicly release LWR transient version of code.
FY 1985-1988	Continue activities identified above (with the exception of code releases, this task having been completed).

2.2 Separate Effects and Model Development

2.2.1 Issue

1. An understanding is required of reactor operational transients and accident sequences, including the thermal-hydraulic aspects of pressurized thermal shock, natural circulation, and the behavior of key components such as steam generators, in order to evaluate the adequacy of operational procedures, safety systems, and system designs.

2. Evaluation of the heat transfer and hydraulics of degraded core cooling is needed in support of rulemaking hearings and other Commission decisions concerning degraded core cooling and plant safety features.

3. Confirmatory data are required to evaluate plant monitoring instrumentation proposed by vendors.

4. An on-call capability is needed for NRR so that licensing issues in existing facilities may be investigated with a short turnaround time.

2.2.2 Research Program Objective

Small separate effects experiments will be conducted in order to provide a better understanding of important phenomena in the area of degraded core cooling and operational transients. These results will then be used to assist in understanding or to plan new integral tests or larger separate effects tests. Several areas requiring large separate effects tests have already been identified, and these tests will be conducted.

The modeling task objective is to evaluate both separate effects and integral experiments to support the improvement of computer codes in these areas. The overall objective of the separate effects and model development research is to provide the necessary link between the integral tests and the computer codes.

Another objective of this research is to evaluate the adequacy of different instrumentation concepts proposed to monitor plant conditions.

2.2.3 Relationship to Other Programs

Separate effects and model development programs provide a key link between the integral experiments and codes. Thus, this program is closely related to the "Integral Systems" tests (Section 2.1) and the "Code Development and Application" effort (Section 2.4). These programs provide the codes with more detailed information than can be obtained in integral tests and with research programs to investigate phenomena at a larger size. Small experiments also provide a capability to quickly and rather inexpensively scope out phenomena. Thus they can be used to evaluate the need for and to plan larger experiments.

This research involving the thermal-hydraulic aspects of degraded core cooling and operational transients is related to much of the work being performed under the Accident Evaluation and Mitigation decision unit (Chapter 4). A better understanding of these areas also contributes toward the risk evaluations performed under the System and Reliability Analysis decision unit (Chapter 10).

The separate effects experimental program has been coordinated with other decision units within RES, with other research agencies such as DOE and EPRI, and with foreign countries (i.e., Japan, Germany, and France).

2.2.4 Background and Status

During FY 1982 and FY 1983, several large facilities involved in large- and small-break LOCA research will have completed testing and thus are no longer shown in the LRRP. These facilities include the 30° Steam Sector Test Facility (SSTF), FLECHT-SEASET, and the Thermal-Hydraulic Test Facility (THTF). The TLTA facility will have been upgraded to an integral facility to study operational transients (see BWR FIST in Section 2.1). The small university programs will also have completed most of their work in the area of large- and small-break LOCAs. Work will continue, however, in evaluating the large amounts of data obtained under these programs.

2.2.5 Research Program Plan

Most of the work planned for FY 1984-1988 will be new work as the issues of degraded core heat transfer and operational transients are to be addressed as previously summarized.

1. Data Bank

The data bank at INEL will continue during the period FY 1984-1988 and will be used to store and make available to researchers previously obtained data and new data. Data from integral facilities such as FIST, Semiscale, and LOFT, separate effects tests such as those under the 2D/3D program, and other new experiments described below will be made available and analyzed through the data bank.

2. Data Analysis Model Development

A major effort to analyze reflood and blocked bundle data from the FLECHT-SEASET and 2D/3D programs will take place in the period FY 1984-1986 as these data become available. A large analytical effort is also planned to analyze both existing and new operational transient data during FY 1984-1988. Table 2-1 provides a summary of this planned activity. This effort will provide input to the computer codes to maintain and improve our calculational capability in the area of operational transients.

3. Degraded Core Cooling

Based on degraded core definition from fuel behavior studies, small-scale facilities will be designed and constructed during FY 1984. Experiments in FY 1985-FY 1986 will investigate thermal-hydraulic and heat transfer behavior in distorted core geometries. Data analyses and model development will continue through FY 1987. Starting in FY 1984, development of instrumentation and evaluation of proposed or existing plant instrumentation will be conducted to aid in assessing degraded core conditions.

4. Operational Transient Separate Effects Experiments

In addition to the small experimental programs and analytical work identified above, data needs requiring larger separate effects facilities have been identified. The steam generator plays an important role in the course of many PWR transients. A need has been identified to investigate steam generator behavior in a large-scale separate effects facility. Another area requiring large-scale separate effects tests is pressurized thermal shock. Industry has conducted a number of small-scale scoping tests to understand mixing of cold injected emergency core cooling (ECC) water with the primary coolant and the thermal transient imposed on the vessel walls. Larger-scale tests are needed to complete this work. We plan to evaluate experimental facilities to study both steam generator response (FY 1984-1986) and pressurized thermal shock (FY 1984). Since the industry is also interested and involved in these research areas, cooperative research efforts with industry may be possible.

Table 2-1. Overview of Model Development and Correlation Verification
FY 1984-1988

NRR User Need		RES Results and Schedule		
		Data Sources		
Request from NRR	Information Needed	Completion of Exp. (FY)	Evaluation of Data (FY)	Model Verification (FY)
Heat Transfer NRR-79-20	Blowdown Heat Transfer			
	Post-CHF Data and Correlation	Lehigh University (1983)	1983	1984
	Refill/Reflood			
	Thermal-Hydraulic Model for Blocked Bundle	FLECHT-SEASET (1981) SCTF (1985)	1982 1983-1986	1984
	2D/3D Effect on Reflood	CCTF/SCTF/UPTF (1985)	1985-1986	1986

Table 2-1 (Continued)
FY 1984-1988

NRR User Need		RES Results and Schedule		
		Data Sources		
Request from NRR	Information Needed	Completion of Exp. (FY)	Evaluation of Data (FY)	Model Verification
Operational Transient	Station Blackout			
	Data			
NRR-79-30		Semiscale (1983)		
	Pumps on/off model			1983
	Model for thermal-hydraulic response to transient			
	Loss of Feedwater			
	Data	LOFT (1981)	1982	
	Coolant - solid interaction (thermal/dynamic shocks)			1986
	Alternative ECC			
		CCTF (1984)	1985	
	Reactor Overcooling			
	Simultaneous Condensation and Thermal Shock	INEL (1985)	1986	1988
	Reactor Monitoring with Simulator Modeling	MIT (1984)	1985	1985
	Core Uncovering			
	Data on Level Swelling and Steam Heat Transfer	Semiscale (1981-1982)		
		THTF (1981)	1982	
		LOFT (1983)	1980-1985	

Table 2-1 (Continued)
FY 1984-1988

NRR User 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>Steam Generator and Natural Circulation</u>			
	Data	CCTF (1984)	1985	
Degraded Core Cooling	<u>Degraded Core Cooling</u>	LOFT (1983)	1982-1985	
	Data	Programs from Fuel Branch		
		FLECHT-SEASET blocked bundle (1982)		
		BDHT Steam Cooling (1981)		1982 preliminary
	Model			1987 final

We also plan to continue to closely follow research efforts of other countries. In particular, we will continue to coordinate with and provide support to the ROSA IV program in Japan in investigating operational transients.

5. Thermal-Hydraulic Facilities at INEL

Several thermal-hydraulic experiment facilities have been used at INEL to provide support to the LOFT program. The NRC plan includes keeping two of the facilities with sufficient manpower to operate one of them at any particular time. Peak manpower and other special needs will share personnel from Semiscale. This operating mode is enhanced by the test facility locations and the INEL organization. The facilities are in the same building and adjacent to the building housing Semiscale, and the same personnel supervise the two facilities and Semiscale. The test facilities are the Blowdown Loop and the Two-Phase Flow Loop (TPFL).

The Blowdown Loop was installed in 1975 to provide separate effects test capability for the LOFT program. It has been used to assess and calibrate LOFT instrumentation, to study heat transfer, to qualify the Power Burst Facility (PBF) blowdown valves, and to test loop pumps. It is useful as a driver for various separate effects tests, as well as for the instruments available to determine test conditions. The loop is illustrated in Figure 2.3. It is operational at pressure in excess of 2250 psi and a temperature of 550°F.

The TPFL was designed to test instrumentation over the full range of two-phase flow conditions expected in LOFT. In addition, it has been used to calibrate LOFT secondary relief valves and an instrumented spool piece for the 2D/3D program. The TPFL is shown in Figure 2.4.

The TPFL is a large system, with space for test components 6-m wide, 6-m long, and 5-m high. Operating characteristics are:

Maximum pressure: 6.9 MPa (1000 psi)

Maximum steam mass flow rate: 25 kg/s (55 lbm/s)

Maximum circulating water mass flow rate: 420 kg/s (9000 gpm)

Steam supply vessel volume: 85 m^3 (3000 ft^3)

The test plan for the Thermal Hydraulic Experiment (THE) facility includes FY 1982 work in flow regime testing and for pressurizer and B&W hot leg (candy cane) investigations in support of the pressurized thermal shock (PTS) program. Work in the following years includes:

FY 1983 — Conduct PTS support tests; and conduct large-scale flow regime tests, including influence of components and flow restrictions, and small breaks via top, side, or bottom exiting pipes.

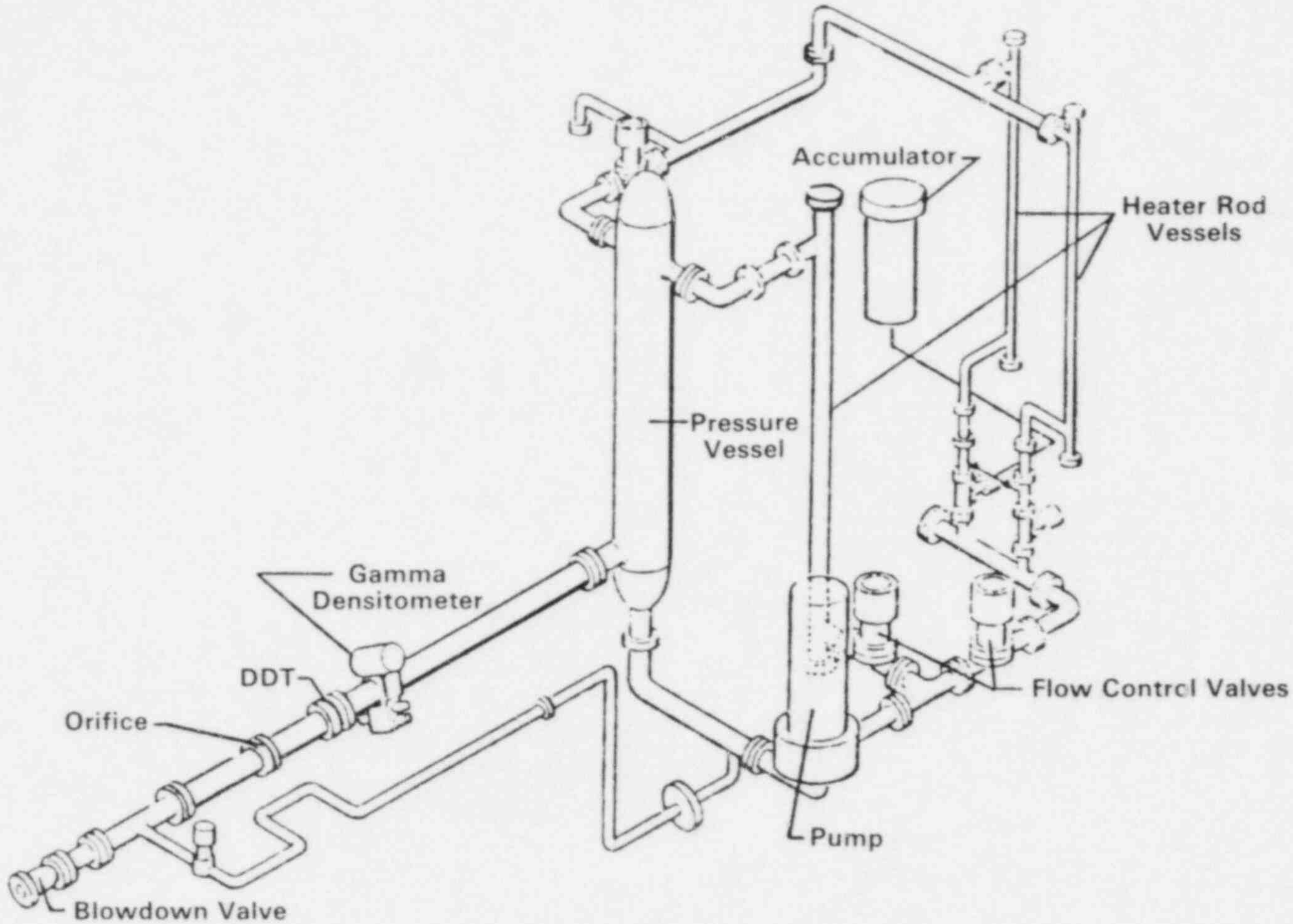

Figure 2.3 Blowdown Facility

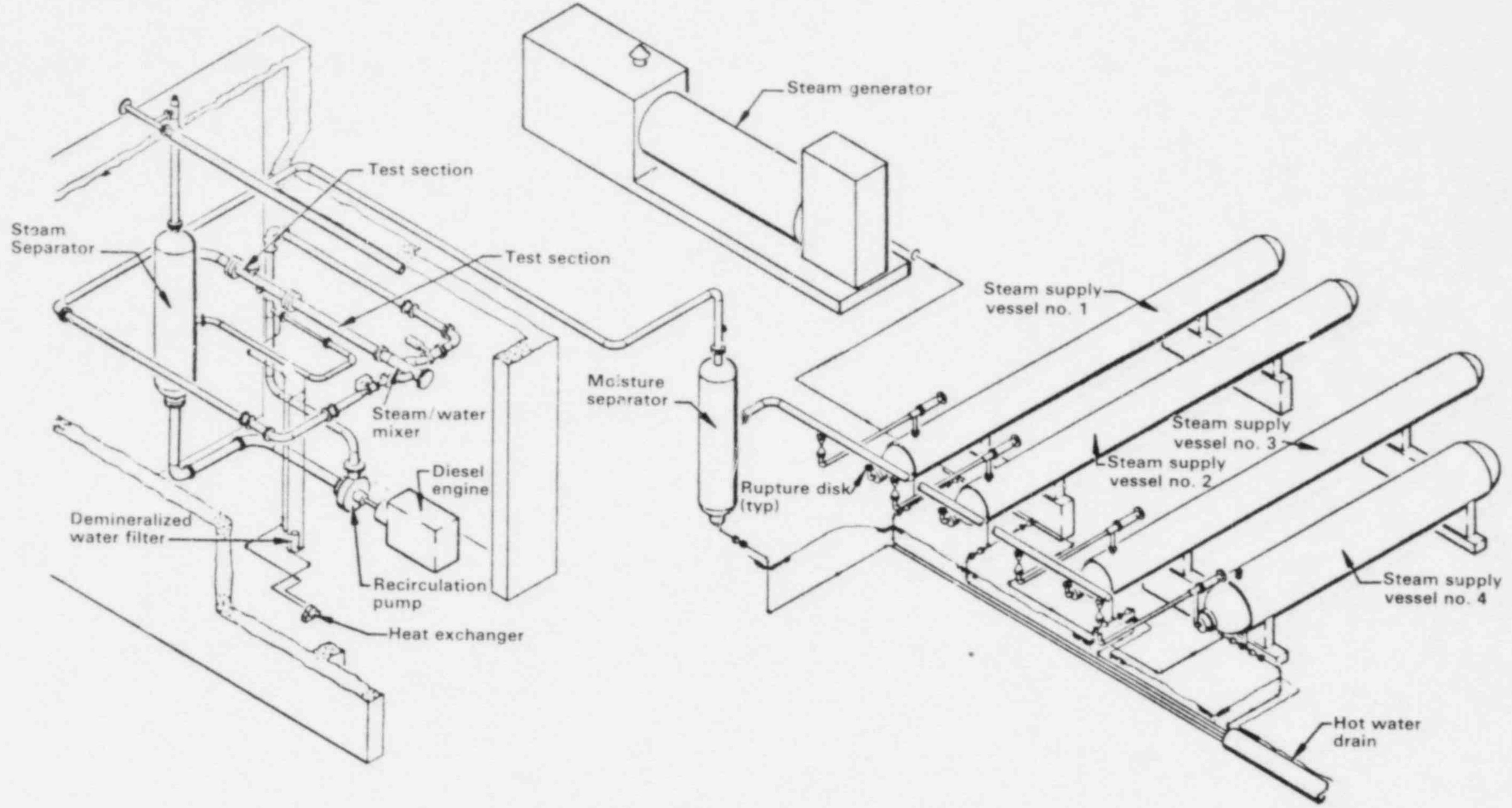

Figure 2.4 Two-Phase Flow Loop

FY 1984 Complete large-scale flow regime tests; study boron tracking; initiate water hammer effects investigation and evaluation of new instrumentation; and study thermal shock.

FY 1985 Complete above activities.

2.3 2D/3D Program

2.3.1 Issue

The 2D/3D program has been formulated to address evaluation of the conservatisms in five areas of current licensing practice and provide three-dimensional data for code improvements and code assessment. The four areas in which conservatisms are being evaluated are the ECC bypass phenomena, requirement to subtract water injected during the ECC bypass period, three-dimensional effect on the reflood process, and effect of core blockage during reflood. Another major issue being addressed by the 2D/3D is providing two/three-dimensional data for code improvement and code assessment for the TRAC and TRAC/COBRA codes.

The 2D/3D program provides the only large three-dimensional separate effects data for the refill and reflood portions of a large-break LOCA.

2.3.2 Research Program Objective

The 2D/3D program was initiated jointly with the Ministry for Research and Technology (Bundesminister fuer Forschung und Technologie (BMFT)) of the FRG and the Japan Atomic Energy Research Institute (JAERI) to minimize the cost in resolving the issues raised above. To reduce the cost further, the 2D/3D program was limited to the PWR LOCA and, for further cost reduction, to the refill and reflood phases of a large-break LOCA and the natural circulation phase of a small-break LOCA. These areas were believed to represent the greatest uncertainty, and hence the greatest risk, in the LOCA-initiated event. In particular, the 2D/3D program has the following objectives:

1. Assess in a large-scale experimental facility the extent to which ECC bypass occurs, and provide a data base to confirm results obtained in subscaled experimental facilities;

2. Address the three-dimensional effects that occur during reflood and their effect on improving reflood on simulated core;

3. Assess the steam-binding effect on reflood;

4. Provide an experimental data base for the evaluation of reflood heat transfer in a simulated core containing flow blockage;

5. Provide data and evaluation of phenomena for the deentrainment of liquid carryover from the core during reflood;

6. Give 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;

7. Study the effectiveness of various ECC injection modes such as hot leg, cold leg, lower plenum, and combined injection. The B&W vent valve concept will also be studied. This will provide data and guidance for reviewing the alternative ECCS design; and

8. Determine and provide data for core cooling by natural circulation and reflux boiling during small-break LOCAs.

2.3.3 Relationship to Other Programs

As mentioned above, one of the reasons for initiating the 2D/3D program was to improve one-dimensional test data by providing large-scale, two/three-dimensional test data. Such one-dimensional test facilities include Semiscale, LOFT, FLECHT-SEASET, and FRG PKL and LOBI. Since most of these facilities can be run at higher pressures and/or higher temperatures than the 2D/3D facilities, these data will complement the missing parts of the 2D/3D program data. The LOFT is, of course, a unique nuclear heater facility, and thereby any difference between electrical and nuclear heating may be assessed by comparing LOFT with 2D/3D data.

Recently, JAERI initiated another large-scale test program, ROSA IV, which is slightly larger than the LOFT facility, but about half as big as the Cylindrical Core Test Facility (CCTF) of the 2D/3D program. However, this facility can be run at the full pressure of a power reactor, 16 MPa, compared to about 0.6 MPa for the CCTF.

2.3.4 Background and Status

The 2D/3D program was initiated in 1978, and the agreement was formally signed in 1980 by the U.S., FRG, and Japan.

In 1978 JAERI completed the construction of a 2,000-heater-rod full-length CCTF with full-height primary loops and steam generators simulating a PWR. A series of 24 tests, including both the refill and reflood phases of a large-break LOCA, was conducted in 1979-1981. Another series of 20 tests will be conducted in 1982-1986 with new vessel internals (Core II) and added instruments.

In addition, JAERI completed in 1980 the construction of a 2,000-heater-rod full-length Slab Core Test Facility (SCTF) with associated simulated loops and steam generator that uses a configuration to represent a radial segment of a PWR. The first 10 of the planned 20 blocked bundle refill and reflood tests have been run, and the rest will be run in FY 1982-1983.

FRG completed in 1981 the design of a full-scale Upper Plenum Test Facility (UPTF) with simulated core, steam generators, and loops. The construction has just started and will be completed by the end of 1984. This facility will be used for, among other tests, upper plenum entrainment and deentrainment tests, combined injection tests, and the downcomer bypass tests.

The NRC provided the advanced two-phase flow instrumentation to the CCTF Cores I and II and SCTF Core I and will do so for the UPTF and the SCTF Cores II and III. In addition, NRC has been providing analytical support for all facilities by performing facility design calculations and pretest and posttest analyses of experiments using the advanced multidimensional two-fluid transient analysis code (TRAC).

In FY 1983, the analyses of CCTF Core I test data will be completed and significant results will be reported in a Research Information Letter (RIL) on CCTF I tests. Steam-binding effects on reflood will be examined, and the margin of conservatism in evaluation models will be assessed. In addition, SCTF Core I test series will be completed. NRC instruments for UPTF will be designed and fabricated.

2.3.5 Research Program Plan

The 2D/3D program plans for the next 5 years are shown below on a yearly basis:

1984 The SCTF Core II will be constructed. The NRC instrumentation for Core I will be refurbished for use in Core II. The TRAC analysis for SCTF I data will be completed. The effect of flow blockage in the bundle will be examined. Significant results will be reported in a RIL on SCTF I tests. NRC instruments for UPTF will be delivered, and the construction of UPTF will be completed. The PKL II test series will be completed.

1985 SCTF Core II test series will be completed. The PKL II data will be analyzed, and the flow distribution in the vessel at the end of blowdown phase of a LOCA will be examined. Significant results will be reported in a RIL on PKL Core II tests.

1986 The NRC instruments for SCTF Core II will be refurbished for use in Core III. New NRC instruments needed for coupling between SCTF III and UPTF will be delivered to SCTF Core III, which will be constructed by early 1986. The CCTF Core II test series will be completed. The ECC bypass and other refill phenomena will be examined. Effectiveness of various ECC injection modes will also be examined. Significant results will be reported in a RIL on CCTF II tests. Shakedown tests in UPTF will be completed, and the main tests will be started.

1987 The SCTF Core II test data will be analyzed and two-dimensional flow effect in core will be examined. Significant results will be reported in a RIL on SCTF II tests. The SCTF Core III test series will be completed. The UPTF test series will also be completed.

1988 The SCTF Core III and UPTF test data will be analyzed. The ECC bypass and other refill phenomena will be reviewed against licensing criteria with the full-scale UPTF data. The earlier smaller-scale CCTF II data will be compared with the UPTF data. The entrainment and deentrainment of liquid in the upper plenum and hot legs will be examined. A RIL will be issued to summarize all the significant results obtained under the 2D/3D program.

2.4 Code Development and Application

2.4.1 Issue

1. Identifying the margin of safety in licensing assessments requires codes capable of providing a BE simulation of system behavior under the following accident scenarios including the effect of multiple failures, partially operable safety systems, and operator errors:

   a. Large-break LOCA,
   b. Intermediate-break LOCA (IBLOCA),
   c. Small-break LOCA (SBLOCA),
   d. Main steamline break (MSLB),
   e. Runaway feedwater transient (RFT) (Scenarios c, d, and e are the main ones being investigated for the PTS issue),
   f. Steam generator tube rupture (SGTR),
   g. Operational transient (OT), which is dominated by balance-of-plant (BOP) behavior,
   h. Combination primary/secondary LOCA,
   i. Anticipated transients without scram (ATWS),
   j. Reactivity insertion accidents (RIAs),
   k. System behavior under degraded core cooling (DCC) conditions,
   l. Behavior of plants with upper-head injection (UHI), and
   m. Stability of both BWRs and PWRs under two-phase flow conditions.

2. Before the code is used in licensing applications, it must be independently assessed to quantify the accuracy and enhance the understanding of the code's capabilities.

3. Analysis of accident prevention and mitigation as well as the development of accident management tactics requires methods that can provide realistic estimates of plant behavior.

4. Application of codes developed in the code improvement and maintenance program is required to support resolution of unresolved safety issues and plant licensing issues, as well as to assist in the assessment of operator training procedures.

5. NRC-developed codes are applied to the analytic support of several test facilities, both domestic and foreign. Pretest and posttest predictions supplied as part of this support also contribute to the code assessment process.

2.4.2 Research Program Objective

1. Develop, assess, and maintain BE computer codes to analyze accidents in full-scale LWRs.

2. Provide research support that aids in understanding the basic physical processes underlying LWR system behavior under accident conditions.

3. Gather and systematically organize LWR plant data so that input decks for computer analysis can be easily generated.

4. Develop a fast-running, user-oriented plant analyzer capability for both PWRs and BWRs. The plant analyzer will be used to investigate the proper design of safety and control systems.

5. Assess completed codes against test data to ensure that reliable BE codes of known and acceptable accuracy are available to the NRC in the licensing audit and safety evaluation activities.

6. Communicate code weaknesses uncovered in the course of code assessment to code developers in timely fashion to expedite the required code improvements.

7. Use best available codes (1) per NRR requests, to provide plant transient analysis; (2) to determine the signature of transients or accidents perceivable by the plant operator; and (3) to evaluate and recommend the most desirable operator actions that will bring the plant to safe shutdown.

2.4.3 Relationship to Other Programs

Codes developed and assessed under this decision unit are used as analytic tools to assist several other programs in the understanding of LWR system behavior during transient and accident conditions. The accurate simulation of LWR system behavior is necessary in order to develop guidelines for operator action and safety system design so as to prevent core damage. At a minimum, information should be provided to allow management of the accident to limit its progression. These BE codes are also used to define the margins of conservatism in current vendor and utility analyses.

The SCDAP (Severe Core Damage Analysis Package) code developed under another program will be incorporated into a simplified version of 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.

System codes developed to analyze plant transients and small breaks for both PWRs and BWRs are available to perform analyses of plant transients that lead to degraded core conditions. Alternative modes of plant recovery and operator action during these accident conditions are being investigated under other programs, using these codes. Codes developed will also be used to support rulemaking procedures and the resolution of licensing issues.

Under the 2D/3D program agreement, copies of the TRAC-PWR code and TRAC-COBRA code have been provided to the Germans and Japanese, respectively. They are using these codes to perform their own analyses of system behavior during the refill/reflood periods.

The TRAC-PWR code is actively being used in the United Kingdom to help in their understanding of PWR LOCA behavior.

The RELAP 5 code is being used in Japan, England, and other countries for LWR analyses, as well as by vendors and utilities in the United States.

There are two Standard Problem programs in which NRC is participating. The first is the U.S. Standard Problem (USSP) program involving only U.S. participants. The second is the International Standard Problem (ISP) program where the participation is international and managed by OECD/CSNI. It is anticipated that the Standard Problem program will continue as long as important test data are being generated in domestic and foreign test facilities, i.e., at least through FY 1987.

Both system and component codes are being applied to the analytic support of experimental test facilities. The RELAP 5 code provides analytic support to the INEL test facilities, LOFT and Semiscale. The TRAC-PWR code provides analytic support to the various facilities under the 2D/3D program. These include the full-scale UPTF in Germany, the 2000-rod CCTF in Japan, and the 8-bundle-by-1-bundle SCTF, also in Japan. The TRAC-BWR code is being used to analyze data from BWR test facilities, which include TLTA and the 30° SSTF. This code will also be used to analyze data under the new FIST program.

The K-FIX code is analyzing fluid-structure interaction data from the HDR facility in Germany, while the SOLA-3D and COBRA/CONTAINMENT codes are analyzing hydrogen-transport data from EPRI-sponsored containment facilities.

In addition, system codes are being used under the severe accident sequence analysis (SASA) program to investigate the consequences of accident sequences involving multiple failures and/or operator actions.

2.4.4 Background and Status

2.4.4.1 Code Improvement and Maintenance

Systems Codes - Both TRAC-PF1/MOD1, for PWRs, and TRAC-BD1/MOD1, for BWRs, will be released to the NESC (National Energy Software Center) in mid-FY 1983. The modeling improvements in both these codes will concentrate on BOP components and control systems.

The TRAC-COBRA code for analysis of system behavior in PWRs with UHI injection will be released to the NESC at the end of FY 1982.

RELAP 5/MOD 1 was released to NESC in early 1981. It will be assessed early in FY 1983. RELAP 5/MOD 2 will be released to NESC in 1983. The completed LWR code will be released in 1984.

All the above codes will have gone through an extensive period of developmental assessment against test data before being released.

Work will begin in FY 1983 on incorporation of the first version of the SCDAP code into a simplified systems code for degraded core cooling analysis.

Improvement and maintenance will continue on the RAMONA-3B and RELAP 4 codes, as required. RAMONA-3B is the only BWR system code available that has the three-dimensional nuclear kinetics capability required for partial ATWS calculations.

Modeling improvements are being incorporated into TRAC-PF1 and RELAP 5 to handle phenomena expected to occur during PTS transients.

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

Subchannel codes have always been used by the industry to determine the local "worst condition." For this purpose, the work for FY 1982 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 1983. This COBRA code version is also being updated to analyze containment problems of interest to NRC.

The K-FIX code is being maintained for use by the NRC, especially for the analysis of fluid-structure interactions on the core barrel after a large-break LOCA.

The SOLA-3D code is being updated for NRC use in two issues: pressurized thermal shock and hydrogen transport in containment. For the former issue, simple models for turbulent mixing of two fluid streams are being included and will be assessed against test data.

The NUFREQ code for analysis of stability in BWRs is also being improved and maintained.

Plant Analyzer and Data Bank - As the development phase of NRC codes is nearing completion, more emphasis is being placed on making them available in a form for use by NRC personnel and others. This user-oriented development is focused in three areas: (1) use of the latest available computer hardware and improved software to allow computation time of up to ten times faster than real time for LWR system transients; (2) display of the computed transient on terminal consoles so that the user can easily understand the calculated results and interact with the calculation, if desired; and (3) incorporation of LWR plant data into a data bank that is easily accessible for the development of input decks for computer codes and the plant analyzer.

RAMONA-3B is the first code being converted to improved hardware (the AD-10 computer) in order to demonstrate its speed and convenience to NRR users. Proposals have been requested from both Los Alamos National Laboratory (LANL) and INEL to perform similar conversions for the computer codes they have developed. These proposals will focus on developing plant analyzers using existing NRC-developed codes by packaging the software with the most advanced hardware for fast-running simulation work to achieve faster-than-real-time and user-oriented capabilities. With this approach, only a minimum of software will be required. The capabilities of the plant analyzer are expected to be equivalent to our codes that have already been developed and assessed and yet to provide rapid answers.

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 is planned to compile in advance and store the required plant information in a plant data bank.

The data bank will 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 and will be made accessible via remote terminals. EPRI and utility cooperation is being sought to ensure the maintenance of up-to-date data. Demonstration of the plant data bank capability and user convenience is to be completed in FY 1982. Complete data for only one plant will be entered for this demonstration.

2.4.4.2 Independent Assessment

To assess capabilities of TRAC and RELAP 5 code versions, RES has scoped out a comprehensive program involving four national laboratories: Brookhaven National Laboratory (BNL), INEL, LANL, and Sandia.

The assessment effort at BNL emphasizes 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 also involves testing other correlation and models for the basic processes and appropriate sensitivity studies.

Because of the availability of large computational resources, the code assessment effort at INEL, LANL, and Sandia emphasizes examination of code performance through comparison of computed results with test data obtained from PWR experimental facilities (LOFT, Semiscale, 2D/3D, and Separate Effects) and

BWR experimental facilities (TLTA and SSTF), as well as applicable foreign data.

During FY 1982-1983, independent assessment centers on the LWR systems codes for both PWRs and BWRs so that their application to licensing issues can proceed with more understanding of their reliability. Assessment of TRAC-PD2 was completed in mid-FY 1982, and assessments of TRAC-BD1, TRAC-PF1, and RELAP 5/MOD1 codes will be completed in FY 1983.

2.4.4.3 Code Applications

The main licensing issue currently being investigated is the PTS issue. A cooperative program is under way (involving NRC/RES, NRC/NRR, Oak Ridge National Laboratory (ORNL), LANL, INEL, and BNL) to improve modeling for PTS phenomena and correctly simulate the secondary-side feedwater train for Westinghouse, B&W, and CE plants. Three main scenarios are being investigated: small-break LOCA, main steamline break, and runaway feedwater transient.

Other issues being investigated include whether and when to turn the main coolant pumps off after a SBLOCA, and what the accident signature is for a combination primary and secondary LOCA. A partial ATWS in BWRs is also being analyzed.

NRR requests for analysis of pressure drops during accumulator injection and pressurizer emptying are being addressed, along with their questions on behavior of B&W plants during various SBLOCA scenarios.

Stability behavior of BWRs and PWRs is being investigated. Input decks will be generated for selected plants, using the data bank when available.

2.4.5 Research Program Plan

The coordinated plan linking up the three subelements of this task is shown in Figure 2.5. The planned accomplishments, by fiscal year, for code development are listed below.

FY 1984 — The final planned versions for the TRAC code will be released to the NESC: TRAC-PD3 for PWR analysis and TRAC-BD2 for BWR analysis. Both code versions will emphasize modeling of reactor kinetics, with feedback, primarily for ATWS and RIA simulation. The SCDAP code will be incorporated into a simplified version of a systems code, for analysis of LWR behavior under degraded core conditions. The data for several plants will be added to the data bank.

FY 1985 — Periodic updates will be issued for the TRAC-PWR, TRAC-BWR, TRAC-COBRA, COBRA-TF, and RELAP 5 codes. The first version of the PWR system plant analyzer will be completed and

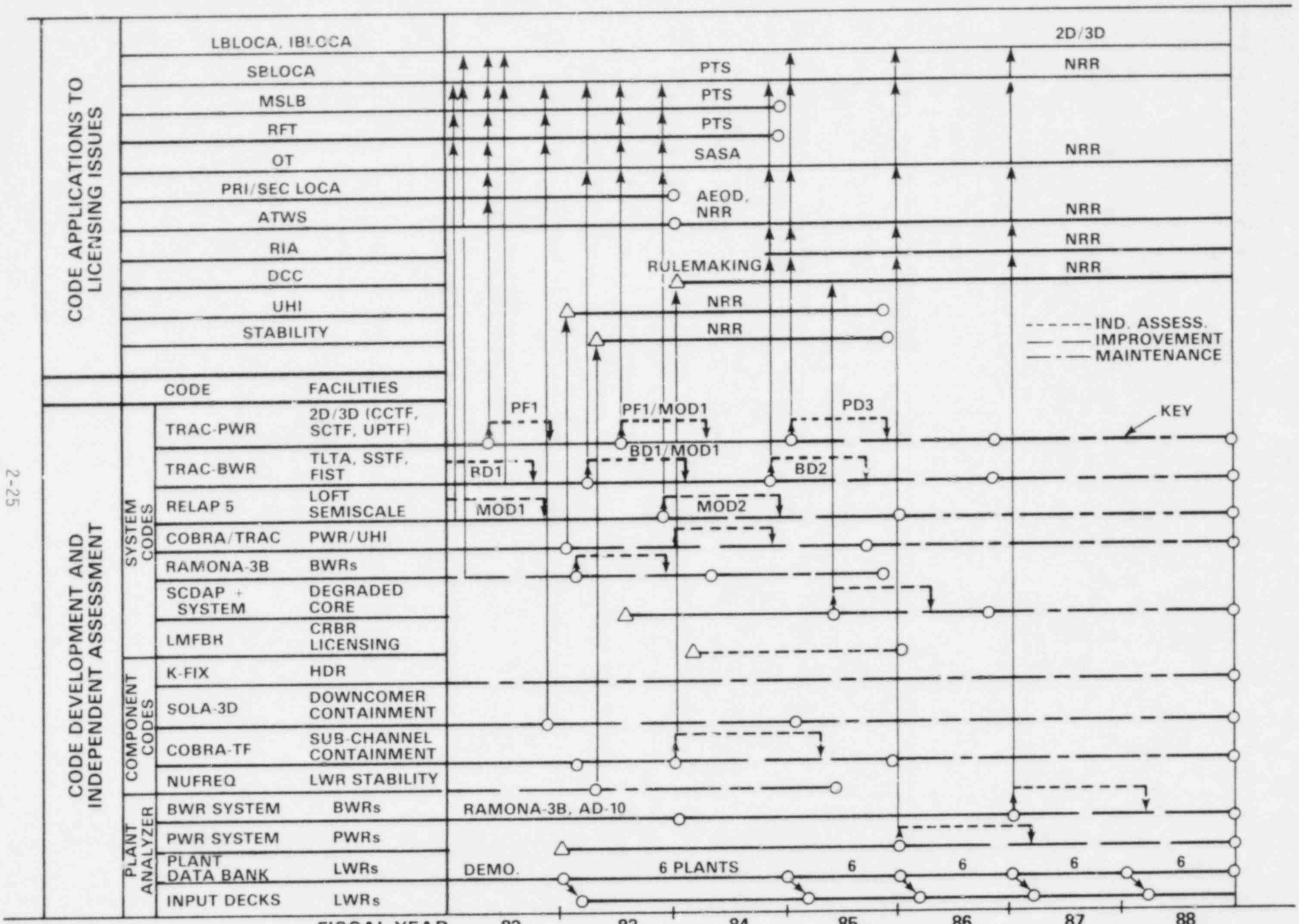

Figure 2.5 Relationships Among Code Development, Independent Assessment, and Code Application

demonstrated for use by NRC personnel. The data for six more plants will be added to the LWR plant data bank.

FY 1986 — The first version of the BWR system plant analyzer will be completed and demonstrated for use by NRC personnel. Maintenance will continue for all NRC codes being used for analysis. The PWR plant analyzer and the associated LWR-plant data bank will be improved and maintained. The data for six more plants will be added to the LWR-plant data bank.

FY 1987-1988 — The PWR and BWR plant analyzers will be improved and maintained, along with necessary maintenance of the LWR-plant data bank. Maintenance will continue for all NRC codes being used in analysis.

Essentially all independent assessment will be completed in early FY 1986. This coincides with the completion of all major code development in FY 1984 (see Figure 2.5). After FY 1986, there will be a small but important program for independent assessment of the PWR and BWR plant analyzers. The effort in code application is planned to increase continually over the next several fiscal years as resources are shifted away from code development and independent assessment. Although it is not possible now to predict what licensing issues will be important in FY 1984-FY 1988, the plan presented here, and illustrated in Figure 2.5, is based on what licensing concerns we now know are important.

The planned accomplishments, by fiscal year, for independent assessment and code application are listed below.

FY 1984 — The following assessments will be completed: TRAC-PF1/MOD1 and TRAC-BD1, for analysis of transients involving the balance-of-plant in PWRs and BWRs; RELAP 5/MOD2, for analysis of large-break LOCAs; RAMONA-3B, for analysis of BWR transients with reactivity feedback; and TRAC-COBRA code, for analysis of UHI plants. Analytic work will continue on the PTS issue. Investigation of the consequences of an intermediate-break LOCA will be reported. NRR user requests will be answered. New plant input decks will be prepared and checked out.

FY 1985 — Assessments of COBRA-TF, for subchannel and containment analysis, and of the NUFREQ code, for stability analysis of BWRs and PWRs, will be completed. Investigation of stability in both BWRs and PWRs will be completed and reported to NRR. UHI plant analysis for NRR will be reported. NRR user requests will be answered. New plant input decks will be prepared and checked out.

FY 1986	Assessment of TRAC-PD3 and TRAC-BD2, for analysis of transients with reactivity feedback in PWRs and BWRs, will be completed in early FY 1986. Assessment of systems codes for degraded core cooling in LWRs and system transients in LMFBRs and assessment of the PWR plant analyzer will be completed. Calculations for the severe accident rulemaking will be completed. NRR user requests will be answered. New plant input decks will be prepared and checked out.
FY 1987-1988	NRR user requests will be answered, and new plant input decks will be prepared and checked out.

2.5 Fuel Behavior Under Operational Transients

2.5.1 Issue

A need exists to analyze the fuel behavior in a licensee's plant design during normal operation and during any postulated accident to determine if fuel rod failures and fuel-cladding temperatures stay below specified safety limits and to be sure that the specified limits are appropriate for the protection of public health and safety. If an expected operational transient occurs, the licensee submits analyses showing the fuel rod failure potential and may request permission to continue operation. NRR requires models and data to audit and assess the licensee's analysis. Accordingly, NRR has requested, and RES has consented to establish, a research program to provide NRR with the models necessary (backed by well-characterized experimental data) to assess licensee and/or vendor fuel damage analyses.

A new model to predict pellet/cladding interaction (PCI) type fuel failure will be developed (and linked to the FRAPCON code) for use by NRR to audit analyses of transients and analyses for advanced design (e.g., higher-burnup and PCI-resistant) core reloads. The NRR staff also plans to use the FRAP-T code in their efforts to review specified acceptable fuel design limits.

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

2.5.2 Research Program Objective

The objective of this element of the decision unit is to produce (1) a user-convenient, best-estimate steady-state computer code (FRAPCON) that has been assessed against a large quantity of high-quality in-pile experiments and against LWR fuel performance data, (2) a user-convenient, best-estimate transient code (FRAP-T) (containing all models necessary to calculate the current design basis accident (DBA) type LOCA, ATWS, power cooling mismatch (PCM), and RIA scenarios) that has been assessed against in-pile and out-of-pile experiments, and (3) a base of experience and assessed models on Zircaloy mechanical behavior that can be used to independently assess the

reusability of reactor cores after anticipated operational transients (AOOs).*
This element of research ends in FY 1985.

2.5.3 Relationship to Other Programs

There is a substantial effort being conducted outside the NRC on related programs. Many of these programs are viewed as solving the problem because of its economic impacts. The very small NRC program is focused on safety impacts, e.g., assessment of the radioactive release potential to the reactor coolant system (RCS).

The EPRI is funding development and assessment of codes similar to FRAPCON (COMETHE and SPEAR) and FRAP-T (STEALTH) as is the FRG (SSYST - a combined steady-state and transient code). The United Kingdom (U.K.) and Japan have also developed codes similar to FRAP-T. The Halden Project has a number of steady-state and transient in-pile experiments under way and others planned, including analytical efforts, one of which is use of FRAPCON-2. Both DOE and EPRI are funding major efforts to develop PCI-resistant fuel rods. The DOE is participating in a multinational high-burnup fuel testing and examination program (TRIBULATION Program). A recent DOE report (CEND-381) summarizes and assesses research and development work in this program area.

2.5.4 Background and Status

In the Commission action after the 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, cladding oxidation, fuel-stored energy, gap conductance, fuel/cladding mechanical interaction, and the overall mechanical properties of the cladding, a comprehensive research program to study these phenomena was begun. All this work has been completed.

Steady-state in-pile tests in the Halden reactor were sponsored by the NRC 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. Postirradiation examination work will be completed in FY 1983. Most of the in-pile transient experiments are being conducted in the PBF. The test program continued through FY 1982 and included tests to simulate such operational transients as BWR turbine trip without bypass. Final analyses will be completed in FY 1983.

2.5.5 Research Program Plan

The remaining program consists of experimental and analytical studies in two major areas: (1) PCI and (2) assessment of fuel codes.

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*Referred to as AOOs in the Standard Review Plan (NUREG-0800).

The basic studies are primarily concerned with PCI failures induced during power ramps. The studies are conducted out-of-pile with modeling performed in parallel.

The best-estimate fuel codes being assessed are FRAP-T (fuel rod analysis program - transient) and FRAPCON (steady-state). The FRAP-T code is 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 and cladding based on prior operational conditions before the transient, as input to FRAP-T, which analyzes the actual transient event.

2.5.5.1 Basic Studies of Cladding and PCI

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 Argonne National Laboratory), (2) determine the important parameters required for assessment and completion of a cracked fuel mechanics model for fuel failures during power ramping (begun by NRR and taken over as a research project by RES in FY 1981 at Pacific Northwest Laboratory (PNL)), and (3) quantify and characterize the PCI failure process via in-pile testing (begun in FY 1980 at Studsvik). All this work is in an intermediate stage and will be completed in FY 1985.

2.5.5.2 Assessment of Fuel Codes

The assessment of the transient and steady-state fuel codes (FRAP-T and FRAPCON) began at INEL in FY 1976. Independent assessment of these codes is performed using test reactor data and ex-pile separate effects data. An assessment program using high-burnup commercial rod data will be completed in FY 1984 for the steady-state (FRAPCON) code. 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.

3. LOFT

This decision unit comprises a program of tests at the Loss-of-Fluid-Test (LOFT) reactor at Idaho National Engineering Laboratory (INEL). The remaining tests in this program utilize the unique features of the LOFT facility; it is the only nuclear-heated facility capable of carrying out tests of the response of the primary system of a PWR to loss-of-coolant accidents (LOCAs), to anticipated transients without scram (ATWSs), and to other abnormal and accident conditions.

These tests provide the capability of ensuring that the codes used in full-scale plant analysis combine the effects predicted by various models in an accurate fashion and that no important effects are overlooked.

In addition, performance of these tests offers an opportunity to try out advanced instruments and data-reduction techniques, as well as an opportunity to understand how to enhance the capability of the operator to respond to accident conditions. The test program selected by the NRC is expected to be completed by early 1983. At that time, depending on current negotiations, the facility may be decommissioned or devoted to testing supported by an international consortium.

3.1 Issue

Tests, in which system effects and nuclear feedback effects are expected to predominate, are required to provide the NRR staff test data against which their safety evaluation codes can be tested and improved.

3.2 Research Program Objective

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

The specific goals of this program are:

1. Acquiring 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 the behavior of PWRs under accident conditions and 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. Testing an advanced operator display system at a PWR under actual accident conditions.

3.3 Relationship to Other Programs

As the only operating scaled PWR, LOFT relates to the separate effects research programs as an integral systems test of individual phenomena and to the systems research programs as a large-scale nuclear-powered test. The early LOFT tests were first run on the Semiscale facility as LOFT counterpart tests to gain understanding of the response to be expected in LOFT. The LOFT LOCA tests were simulated with individual fuel rods in the Power Burst Facility (PBF) reactor to give an indication of the response of the fuel in the LOFT core prior to running each test at LOFT. In FY 1981, a simulation of the Arkansas Nuclear One Unit 2 turbine trip transient was performed at LOFT to obtain information on the effects of scaling on transient response.

Through the direct participation by the reactor safety centers of ten other countries, supporting analytical and experimental results have supplemented the LOFT program. Results from the LOFT program, in turn, have been used to address safety issues in these and in other countries.

In general, LOFT provides an essential link from the small-scale system tests of Semiscale and the many large- and small-scale separate effects tests in NRC's research program to full-scale commercial plant behavior.

3.4 Background and Status

The LOFT program began nuclear-powered LOCA testing in December 1978. The first LOCA test with pressurized fuel in the central core assembly will be conducted in fiscal year 1982. This test, designated L2-5, is a double-ended cold leg break with delayed emergency core coolant (ECC) injection. The test is expected to show that a full break with pressurized fuel and operating conditions consistent with the most conservative of Appendix K assumptions does not result in damage to the fuel cladding. Additional tests to be performed in FY 1982 include an ATWS, a series of four-operational transients, and a boron dilution transient from cold shutdown conditions. The ATWS and the boron dilution transient from cold shutdown were specifically requested by NRR for use in assessing vendor computer code capabilities.

NRC-sponsored testing in LOFT will be concluded in FY 1983. Following a second ATWS test, the test program will be concluded with a double-ended cold leg break with end-of-life fuel pressurization and excessively delayed ECC injection. The results of the last test will provide (1) a measure of conservatism in the licensing evaluation models and current ECC systems, (2) severe ballooning and bursting of the cladding in the central fuel assembly, and (3) a measure of the coolability of a ballooned and burst fuel bundle. The central fuel assembly will undergo extensive postirradation examination (PIE) to obtain data on fuel rod cladding burst conditions and the fuel cladding damage distribution pattern in a 15 x 15 fuel rod assembly.

Beginning in FY 1982, NRC and DOE will hold discussions to define deactivation plans* for the LOFT facility, consistent with the DOE/NRC Memorandum of Understanding. Deactivation of the LOFT facility will be started after the last test is completed. Analysis of the LOFT test results and investigations of its application to operating plants will continue through the year.

3.5 Research Program Plan

Deactivation of the LOFT facility is expected to be completed during FY 1984.

The PIE program for the damaged central assembly will provide data on stress levels in the vicinity of the clad burst, the distribution both longitudinally and radially of the ballooning and burst, the conditions present within the fuel rod at the time of burst, and the postburst cladding oxidation. The examination of the fuel is expected to involve staff from additional national labs, including Pacific Northwest Laboratory, Oak Ridge National Laboratory, and Battelle Columbus Laboratory.

Analysis of the data should be concluded during the year and final reports on the results prepared.

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*The U.S. Department of Energy is considering establishing an international LOFT consortium to run a test program for an additional 2-year period. The proposed consortium would involve major participation by foreign countries, DOE, EPRI, and the NRC. If this can be successfully established, the decommissioning phase would be delayed approximately 2 years.

4. ACCIDENT EVALUATION AND MITIGATION

This decision unit comprises the coordinated phenomenological research programs needed to develop a sound technical basis for NRC decisions concerning the ability of existing or planned nuclear power reactors to cope with severe accidents, i.e., those that involve damaged or melted fuel. It is expected that the major application of this program will be to support regulatory decisions on new standardized plants and plants in the early and mid-1980's. Also, some provision may be needed for backfitting to operating plants. This provision must be consistent with safety goal policy yet to be developed. To ensure a sound technical basis for these regulatory decisions, two categories of information will be developed: (1) a manageable analysis process and models to assess benefits in terms of residual risk reduction and the accompanying costs and (2) a base of data related to the behavior of nuclear power plant systems and components under a range of severe-accident conditions. The risk analysis process can therefore be knowledgeably applied. This unit concerns the second category. The plan for category 1 is in the Risk Analysis Decision Unit.

The five elements of this decision unit are:

1. Behavior of Damaged Fuel: A program of in-reactor and laboratory experiments and analyses to obtain information for determining the characteristics of a severely damaged core at different points in severe accident sequences, the coolability of the damaged core by reflooding, hydrogen generation, and fission product release. This is an integrated four-part program of research. The first part consists of integral, multi-effects, in-pile tests, primarily in the Power Burst Facility (PBF) test reactor but also in the Canadian NRU reactor, to provide early scoping data on governing phenomena and later data for proof tests of the severe fuel damage models and codes developed in the program. The second part consists of separate-effects experiments on the governing phenomena, both in the Annular Core Research Reactor (ACRR) and in the laboratory, to furnish a data base for model development. A Severe Core Damage Analysis Package (SCDAP) is the third part of the integrated program, including development of severe fuel damage models from the experimental data base and their integration. There will be continuous active interaction and feedback between the analysis and experimental programs. The fourth part involves the benchmark information to be obtained later from the TMI-2 core examination, which itself will benefit from the early results of the behavior of damaged fuel research.

2. Fuel-Melt Behavior: Research to investigate the interaction of melting fuel with the primary system, interactions that will occur if the damaged fuel cannot be successfully cooled, and also interactions of the mass of molten fuel with the concrete should the primary system be penetrated. Some of the tests in this area are extensions of work done under the behavior of damaged fuel program, but one large facility is unique to this program: the Large Melt Facility (LMF), an inductive furnace at Sandia capable of melting masses up to 500 kg of fuel (UO_2) and testing the reaction with concrete or other structural materials. This program includes work on steam explosions and hydrogen generation and control, as

well as analytic work to model the interactions so that test results can be factored into risk analyses and risk benefit vs. cost assessments.

3. Fission Product Release and Transport: A program to determine the radiological source term in serious accidents. The state of the art for such determinations was summarized in the recent publication, NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents." This program includes the work needed to reduce the uncertainties noted in that report, such as high-temperature chemistry of fission products as they are released from fuel and interact with clad, structure, and coolant; the transport of fission products within the primary system, including processes that tend to either decrease or increase the amount of radioactive material reaching the containment; and transport within the containment. A major existing test facility is the Nuclear Safety Pilot Plant at Oak Ridge National Laboratory (ORNL), where scaled experiments on transport within containment are performed. Other facilities exist at Sandia and Battelle Columbus Laboratory (BCL). The tests are analyzed in codes that are incorporated into risk analysis methods as well as mechanistic accident analysis code systems.

4. Accident Mitigation: Research to determine the environmental conditions for which equipment should be qualified to permit containment to function reliably under severe accident conditions, including such effects as significant hydrogen production and burning, radioactive aerosol and steam dispersal, and the evolution of products from the core-concrete interaction. The research program also develops generic engineering design criteria for such systems as hydrogen burning or absorption systems designed to function under severe accident conditions.

5. Severe Accident Sequence Analysis: A program of systems analysis using best available methods to perform mechanistic analyses of reactor accident scenarios characterized by risk analysis as posing the greatest potential risk. The objective of the analyses is to determine how the operator may intervene in an optimum way to limit the accident progress or to mitigate its consequences, what information is needed to determine the correct procedure, and what alternative strategies may be employed in case of unforeseen equipment failure. This program also responds to specific regulatory information needs for information of a similar nature. The program is currently carried out at four national Laboratories--Idaho National Engineering Laboratory, Los Alamos National Laboratory, Sandia, and ORNL--and at private laboratories.

4.1 Behavior of Damaged Fuel

4.1.1 Issue

Severe core damage resulting in large hydrogen and fission product releases to the containment can occur despite current regulatory procedures and engineered safety systems. However, the TMI-2 event has shown that accidents that result in core temperatures in excess of 2200°F need not result in a massive core melt, pressure vessel failure, containment failure, or large releases of radiologically significant fission products to the environment as has been conservatively assumed in the past.

If the accident is terminated below approximately 3400°F, the issue is the coolability of a core containing fragmented pieces of oxidized and embrittled Zircaloy-clad fuel rods. The major safety issues at temperatures between 3400° and 4700°F are core coolability (i.e., can further core degradation be stopped?) and fission product and hydrogen release from very hot solid fuel rods, liquefied fuel, and fragmented fuel.

4.1.2 Research Program Objective

The formulation of regulatory policies and criteria for operating procedures to manage and mitigate the consequences of such accidents requires the development of a data base and analytical methodology ranging considerably beyond that needed for current design basis accidents. Very little data are currently available on the characteristics of severely damaged LWR cores. Information is required to determine the coolability of the core, the coolability of various types of fuel/clad debris, the nature of the thermochemical reactions that take place at high temperatures, and the extent and nature of the fission products and hydrogen released. Reliable information must be obtained from in-pile tests that closely duplicate reactor conditions such as the nuclear heat source (in liquid and solid phases), fission products, and prototypical fuel/cladding thermal and chemical reactions. In addition, we need an overall understanding of the way in which fuel relocates as cooling is severely degraded or totally lost so as to be able to model the attack of hot or molten fuel on the lower vessel internals. We also need to determine a correlation, if any, between reflood rate and core uncovery time that minimizes further fuel damage from quenching.

This information is needed to form a technical basis for:

1. Licensing and rulemaking decisions for accident conditions beyond the design basis,

2. Establishment of performance requirements for engineered safety features to achieve in-vessel termination of accidents,

3. Accident management guidelines,

4. Potential reduction of calculated risk with knowledge of in-vessel severe core damage behavior, and

5. Reduction of uncertainties in probabilistic risk assessment (PRA).

4.1.3 Relationship to Other Programs

Key interfaces of this element with other NRC-sponsored research elements include:

1. Hydrogen Generation and Control - The PBF test program and the resulting SCDAP assessment will provide important integral data on the time-dependent hydrogen release rate from a severely damaged core.

2. Probabilistic Risk Assessment - The results of the severe fuel damage (SFD) research program will furnish a data base and models, which are not currently available, on the in-vessel portion of severe-accident sequences and on the potential for partial recovery from severe accidents. As TMI-2 clearly demonstrated, maximum fuel temperatures exceeding 2200°F do not inevitably lead to full core meltdown, and partial recovery is possible. Adequate information to incorporate the in-vessel recovery potential into PRA does not currently exist, and furnishing such information (data and models) is a major goal of the SFD program. Furnishing information on the conditions at core melt-through of the reactor vessel in unrecovered accident sequences is also a goal of the SFD program. The SDF information may significantly reduce the current numbers for reactor risk and should significantly reduce the uncertainties in the calculated-risk numbers.

   Currently best-estimate state-of-the-art codes such as RELAP, TRAC, MARCH-CORRAL, and CRAC are used in PRA, and a new risk code (MELCOR) is under development. The MARCH code was never intended as a device to predict details of degraded core behavior. It is currently being used for this purpose since other codes do not yet exist and since a reliable data base is not available as an alternative.

3. Severe Accident Sequence Analysis (SASA) - In order for the SASA program to achieve its aims, a strong and reliable data base on the response of all safety-related components of the plant is required. Both SASA and PRA disciplines are examples of logic exercises that produce no new data themselves but rely on a wide-ranging data base of plant physical responses under abnormal conditions. If the data base for these analytical methods does not exist, the postulated cause-effect relationships can break down and the conclusions will either lack sufficient assurance to be useful or even become untenable.

   There are at present only very limited data on severe core degradation, fission product behavior and responses, hydrogen generation, core support structure attack, and large melt behavior. Since accidents involving core degradation and meltdown dominate risk, governing phenomena in such accidents must be better understood for the SASA program to be successful. The behavior of damaged fuel program element is important in formulating recovery and control guidelines, since, without a firm phenomenological base, the conclusions drawn from SASA studies will have an unacceptable degree of uncertainty concerning the best approach to bring the reactor to a safe shutdown from a severely damaged state.

4. Fission Product Release and Transport - The PBF program will provide important integral data on the release and in-vessel transport of fission products and aerosols during severe-accident conditions. The PBF tests will also indicate what phenomenological events significantly influence fission product release such as fuel/cladding motion, fuel fragmentation, and interactions of control material and structural materials with the fuel rods. These latter effects are very difficult to investigate and quantify in out-of-pile separate effects tests.

Invaluable benchmark data in severely damaged fuel will be furnished by the examination of the TMI-2 core. This planned work will be done by the DOE in close coordination with NRC.

Industry is supporting some SFD research through the Electric Power Research Institute (EPRI). The Industry Degraded Core (IDCOR) program is performing analysis on melt progression, in-vessel debris coolability, melt penetration of the reactor vessel, and ex-vessel debris coolability.

4.1.4 Background and Status

The consequences of a severe accident are dependent upon the sequence of damage in the accident, and there are many paths that degraded core cooling accidents can take. The SFD program is designed to map in a rudimentary way the complex response surface defined by the damage phenomena produced by varying certain key parameters, i.e., heating rate, cooling rate, steam flow, peak temperature, fuel rod burnup, bundle size, and the presence of low-meltingpoint control and structural materials. This mapping of the damage phenomena is needed to bound the range of the effects of these various parameters. Depending on the parameter, SFD states and configurations can range from fuel rods with cladding totally oxidized to ZrO_2 and geometry altered only by localized rod ballooning and rupture of the rods during the heatup; to rods with the formation, relocation, and freezing of molten cladding and liquefied fuel; to the formation of rubble beds of fuel pellet fragments, oxidized cladding fragments, solidified molten fuel, solidified liquefied fuel, and solidified spacer grid and control rod materials. Data needed on the amount and timing of the release of fission products and the generation of hydrogen are also obtained along with the information on damage progression. All the data will be used to characterize the resulting core geometry so that realistic coolability studies of such configurations can be made both in-reactor and ex-reactor to answer the important question of how to maintain and manage a damaged core without further degradation and additional fission product release.

Information on the progression and character of damaged fuel and related coolability will provide needed technical bases for developing accident management guidelines and potential refinements to system design. Of particular significance is reflooding of a damaged core to minimize the potential for reduction of the damaged-core coolability by the additional damage produced by quenching. This information need not, in general, be highly detailed. The emphasis is on characterizing the state of damage and on correlating the damage state and hydrogen generation and fission product release with the coolant history and measurable core exit temperatures for the higher-probability severe-accident sequences.

Had portions of the TMI-2 core reached the melt stage, the predictions consistent with PRA scenarios are that the molten fuel would have attacked and failed the reactor vessel. The existing PRA models of the attack are limited because of a lack of detailed knowledge of this process. The vessel attack by molten-core material represents the end-point of severe fuel damage and is included within the scope of the SFD program.

Two major facilities will be used to conduct the experiments that address the program objectives. One is the PBF, in which integral (multi-effect) tests are performed with 32-rod bundles of test fuel in a flowing water/steam loop, and the state of the damaged fuel is determined by posttest examination. The other is the ACRR, in which separate-effects phenomenological experiments for model development are performed on small 9-rod arrays with continuous optical diagnostics of the state of the test fuel. Experiments on the coolability limits of severely damaged fuel and core debris under reflood conditions will also be performed in the ACRR, following a series of successful similar experiments on the coolability of LMFBR core debris in sodium. The capabilities of these two test reactors and the respective experimental programs strongly complement each other. Substantially more information can be obtained in this complementary operation than would be possible using either facility alone.

4.1.5 Research Program Plan

A four-part integrated program of research (conducted in parallel) is proposed to provide the needed information base (data and verified models). The first part consists of integral, multi-effects, in-pile experiments in the PBF and NRU to provide both early scoping data on governing phenomena and later data for proof tests of the SFD models and codes developed in the program. The second part consists of separate-effects experiments on the governing phenomena, both in the ACRR test reactor and out-of-pile, in laboratory bench-scale experiments to furnish a data base for model development. An analysis package is the third part of the integrated program, including development of SFD models from the experimental data base and their integration into the SFD code, SCDAP. The fourth part is the important benchmark data to be obtained from the TMI-2 core examination. The early results of this SFD research will be available to assist in the planning and the execution of the TMI-2 core examination.

4.1.5.1 In-Pile Integral Experiments

The major part of the program of integral in-pile experiments is the SFD series in PBF. Phase 1 of the program, which is now under way, will provide integral scoping data in the temperature range of 2200-2400 K. The tests also will provide data on hydrogen generation and fission product release from the reactor core. The characteristics of the severely damaged fuel will be obtained from posttest examination. This series is the foundation of the SFD research program and will form the necessary base for the in-pile and laboratory separate-effects experiments on governing phenomena as well as for the models in the integral fuel-behavior code SCDAP.

Current plans call for five 32-rod experiments* to be conducted: two at slow heating rates less than 0.5°C/sec (to fully oxidize the cladding and therefore preclude the formation of liquefied fuel), two at faster heating rates of about 4°C/sec, and one approximating the estimated TMI-2 conditions. One of each of the slow and fast rate experiments will be cooled slowly from maximum temperatures of about 2175 K (1900°C, 3460°F) to preserve as much as possible the configuration existing at the maximum temperature, and the others will be quenched with reflood water to produce quench debris. The detailed plans for the fifth experiment simulating TMI-2 conditions have not yet been specified. The early experiments will also verify the adequacy of the design of the test train and the shroud that is required to contain the liquefied fuel in later experiments. The characteristics of the debris in these tests will be used for assessing the coolability of such debris by core reflooding. These characteristics will also be used in guiding and planning separate-effects debris coolability experiments in the ACRR. Finally, data on hydrogen generation and fission product release and transport will be obtained in all these tests and will be used to assess the models in SCDAP as well as current source-term methodology.

There has been preliminary planning for a possible second phase of integral SFD tests in the PBF to explore the effects of irradiation, control rod materials, and fuel element design. This series, to be completed in FY 1986, would also include experiments at higher temperatures. To explore the effects of using actual decay heat rather than fission simulation of decay heat, a 1-week irradiation of previously irradiated fuel to replenish the short-lived fission product inventory will be used.

Starting in FY 1984, integral SFD data will be obtained from experiments in the NRU reactor at Chalk River, Canada, with 21-rod, full-length fuel bundles. The data will supplement the PBF results with 3-foot-long fuel bundles and determine the scaling effect of a 12-foot axial length. These tests may also cover a wider range of accident conditions than the PBF Phase 1 tests, including high pressure and both PWR and BWR conditions.

Starting in about FY 1986, integral SFD data are currently projected to become available from the SUPER-SARA program in the ESSOR reactor at Ispra. Fourteen of the 20 planned tests are to involve severe fuel damage. The data will supplement the 3-foot PBF results and determine the scaling effect of a 6-foot axial length. The SUPER-SARA tests also cover a wider range of accident conditions than the PBF Phase 1 tests, including high pressure and both PWR and BWR conditions.

4.1.5.2 Separate-Effects Experiments

The second major part of the research on severe fuel damage is a program of complementary separate-effects phenomenological experiments on the dominant processes involved in the behavior of severely damaged fuel. A major objective of the separate-effects experiments is to determine the range of core

*Four of these experiments will have been conducted prior to FY 1984.

conditions (if any) for which simple reflood is not sufficient to cool the debris and terminate the accident within the reactor vessel and to determine the forced reflood (pressure and flow velocity) necessary for coolability under these conditions. The dryout coolability limit is reached when liquid cannot penetrate to all points in the bed against the outflowing vapor. Considerable data and rather sophisticated analytical models of the quasi-static dryout coolability limits of beds of decay-heated particulate fuel debris under liquid pools have been developed in fast reactor safety research. A series of seven LWR-specific core-debris coolability experiments* will be performed in the ACRR. These will be extensions of the previous LMFBR safety experiments, and the purpose of the initial experiments will be to validate, for LWR accident conditions, the current fast-reactor debris-coolability models. The LWR-specific conditions that require experimental verification, in addition to the change to water coolant, are high pressure, very deep debris beds, inlet flow, and particularly the characteristics of the LWR core debris. It is known that the characteristics of the core debris are a major determinant of the dryout coolability limit under reflood conditions.

A program of separate-effects phenomenological experiments has also been started in the ACRR on the mechanisms involved in the formation and relocation of fuel debris and on the characterization of the debris. These experiments will provide visual data continuously (in the form of motion pictures) for specified accident sequences, as well as debris characterization for reflood quenching at various times in the accident sequences. Data from these separate-effects experiments will be used to develop phenomenological models of the major processes for incorporation into the SCDAP. These separate-effects experiments effectively complement the larger-scale integral Phase 1 SFD tests in the PBF, and they will substantially broaden the data base for model development.

Laboratory separate-effects experiments are planned (depending on the amount of supplemental work in the FRG) to determine the thermodynamics and kinetics of the reactions between UO_2, Zircaloy, and steam, including steam starvation effects. Experiments are also planned on the candling process** with the ternary (U, Zr, O) liquefied fuel and on debris formation in reflood quenching of molten fuel. These experiments would be performed in FY 1984 and 1985. The steel-steam interaction is also a significant source of hydrogen but is associated with higher temperatures than the (U, Zr, O) liquid phase. It will be studied separately.

4.1.5.3 Analytical Modeling

The major part of the analytical component of the integrated program is developing the integral SCDAP code for the detailed analysis of fuel behavior

*Three of these experiments will have been conducted prior to FY 1984.

**The observations to date are that the liquid mixture flows down the gap between fuel and cladding like the wax melting on a light candle, hence the term "candling."

during severe-accident transients, including melt progression to reactor-vessel failure and the development of the individual models of the dominant processes incorporated in SCDAP. The development of SCDAP and particularly of the individual models will depend on the data base provided by the integral tests in the PBF and other reactors and on the in-pile and laboratory separate-effects phenomenological experiments. The analytical program will also include analysis of the high-probability accident sequences involving severe fuel damage to determine the governing phenomena and uncertainties. This analysis will be performed early in the program to guide the experimental program and also the model development for SCDAP.

A model for fuel/cladding melting has been developed at Stuttgart-IKE, FRG, to model the severe fuel damage incurred in experiments at KfK, FRG. Documentation of this model, called EXMEL, has been obtained, and the model 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, TRAP-MELT, MELCOR, and MATADOR. These latter codes are intended to describe the overall reactor core, vessel, and containment behavior during severe accidents.

4.1.5.4 TMI-2 Core Examination

The TMI-2 core examination constitutes a unique and invaluable resource on the characteristics of severely damaged fuel. Early recovery and adequate analysis are highly desirable to provide a benchmark for research on and an understanding of the behavior of severely damaged fuel, including development of the SCDAP code. The current program on the behavior of damaged fuel includes support for analysis of TMI-2 core debris but does not, of course, address the cost of the TMI-2 recovery operation. Early results of the SFD program will be of use in planning and performing the TMI-2 recovery operation. The actual program of SFD research needed to provide a sound technical basis for accident management and licensing activities may prove to be considerably less extensive than outlined in this plan. This program was derived from our current state of knowledge on the characteristics of severely damaged fuel and on the behavior of such fuel, for which the data base and verified models are in a primitive state. It may well be that later, with data from the PBF Phase 1 scoping experiments, the early ACRR phenomenological separate-effects experiments, and the TMI-2 core examination and with models developed from these data, some of the planned program will prove unnecessary. In any case, the SFD program requirements and program plans should be and will be reexamined periodically.

4.2 Fuel-Melt Behavior

4.2.1 Issue

The regulatory issues are the need to develop source terms for fission products, noncondensible gases, and aerosols within the containment building and to determine missile, temperature, and pressure loads that could cause structural failure and result in threats to the health and safety of the public.

The technical issues include the interactions of core material or hot severely damaged fuel with the internal containment environment; the interactions of fuel and water; the rapid generation of steam and the possible formation of missiles; the loads on the containment structure; the effect on instrumentation required to follow or control the accident; the source terms of heat, pressure, and fission products required for the design of mitigation systems; and the quantification and verification of parameters for analysis codes.

Steam explosions from in-vessel core-melt/water interactions have the potential to fail the reactor vessel and also to generate missiles that threaten the containment. Nonexplosive rapid steam generation during both in-vessel and ex-vessel melt/water interaction (the "steam-spike" problem) also have the potential to breach the reactor vessel and the containment directly. The characteristics of the particulate debris formed by water reflood of core melt form a key element in assessing the coolability of that debris. Debris-bed coolability is assessed by determining the conditions at dryout of the bed where water can no longer penetrate the hot region of the bed to cool it.

4.2.2 Research Program Objective

The research program objective is to provide a sound basis for resolving the issues noted above. Experimental data will be provided for core-melt/concrete interactions, with and without water present for steam explosions, and for long-term hot-debris/basemat interactions. The results will be provided in codes capable of addressing specific licensing applications.

4.2.3 Relationship to Other Programs

The KfK Beta Facility is expected to be generating data with large thermite melts by 1985. No other data for comparison evaluation are expected. However, the analytical tool being developed for containment analysis (CONTAIN) will have the core-melt/concrete module (CORCON) provided as an independent code that may be used with other containment analysis codes. In addition, the cooperative international program at Sandia on debris-bed cooling will have furnished much basic data; there may be a follow-on program by Euratom that will offer additional data.

4.2.4 Background and Status

Fuel-melt behavior research has been divided into three subelements: (1) fuel melt and hot solid interactions with the basemat, (2) core-melt/coolant interactions, and (3) core-melt accident analysis.

4.2.4.1 Fuel Melt

The scope of this task includes small-scale scoping and phenomenological experiments of thermal, mechanical, and chemical interactions of fuel above thesolidus temperature and of high-temperature core debris simulants with concrete; refractory and sacrificial materials; large-scale scoping or model-verification tests; development of computer models of the interactions; quantification of gaseous and aerosol source terms for the interaction; heat redistribution with gaseous or aerosol sweepout; and evaluation of the effect of coolant on the melt-concrete interaction. Experimental investigations of reaction rates and the definition of aerosol source terms should be completed in FY 1983. The initial large-scale core-melt tests will be conducted in FY 1982 and systematic investigations of core/concrete interactions will continue throughout FY 1983.

4.2.4 2 Core-Melt/Coolant Interactions

An extensive data base on the conversion of core-melt thermal energy into steam-explosion mechanical work has been developed in the Fully Instrumented Test Series (FITS) facility. These results, when combined with analysis of missile generation by in-vessel steam explosions and missile failure of containment, led to an early estimate that the probability of containment failure by steam explosion in an LWR meltdown accident is considerably less than the 0.01 estimate in WASH-1400, but still not zero. Medium-scale FITS tests with drops and thermite-generated corium melts into water have been completed. In addition, single-drop experiments on the phenomenological mechanisms involved in steam explosions have been made. These experiments have provided important information, but sufficient understanding does not currently exist to construct a mechanistic model of the thermal detonation process that would have predictive capability.

4.2.4.3 Core-Melt Accident Analysis Status

The ex-vessel interaction of core-melt materials with the concrete and the resultant loads on the containment are being programmed in the CONTAIN code. The detailed analysis of the hot core material interacting with the concrete basemat is being made in the CORCON code. This code is available both in a stand-alone version and as a module of CONTAIN. Both CORCON and CONTAIN will be validated by adjustments to large-scale core/concrete tests to be conducted in FY 1982 and 1983. An aerosol module, MAEROS, in CONTAIN is considered a state-of-the-art tool for computing fission product transport via aerosol migration. However, source terms are less well defined and the core/concrete type of testing will produce aerosol source-term information in addition to the interaction information.

4.2.5 Research Program Plan

4.2.5.1 Fuel Melt

A systematic study with large-scale fuel-melt interactions with concrete and retention materials will continue throughout FY 1984 and the effects of pre- and postintroduction of coolant to the fuel melt will be evaluated. The CORCON model will be verified with tests and analyses throughout FY 1984-1985. The KfK Beta Facility is expected to be generating data with large thermite tests by FY 1985, and these results will be assessed against the LMF results. Core coolant effects and design considerations are expected to be the primary emphasis in FY 1985, and verification experiments will be concluded in FY 1985. FY 1987 efforts will be mostly analytical, completing documentation and evaluation of previous tests. Design support will constitute a major portion of the FY 1987 effort but at a much reduced cost level from FY 1985 or 1986. All work is expected to be completed in FY 1988 and the core-melt investigation terminated.

The investigation of hot solids and their interaction with concrete will proceed concurrently with the above tests. The hot solid interaction tests are to evaluate the long-term cooling problems and, in general, will be representative of solidified melts. These tests will have nominally the same schedule as the melt tests, i.e., most work will be completed in FY 1985 and verification experiments will be concluded in FY 1988.

4.2.5.2 Core-Melt/Coolant Interactions

Starting in FY 1984, emphasis in this program will shift to in-vessel steam explosions under reflood conditions in severe-accident recovery, a potential threat to reactor vessel integrity. Also experiments will be initiated on nonexplosive rapid steam generation (steam spike) that might threaten the integrity of both the reactor vessel and the containment. Experiments in these areas will continue through FY 1985 and then terminate unless important problems requiring further work become apparent.

The majority of the experiments will be done with thermite-generated melts using the FITS facility. Some check data will be obtained with furnace-heated purely oxidic melts, with one large-scale check test (200 kg) probably performed in the LMF. Analysis of both the explosive and the nonexplosive rapid-steam-generation processes, in both the dropping and reflood contact modes, will be continued in an attempt to develop predictive, mechanistic models.

As currently foreseen, the needed research on core-mass/coolant interactions will be completed in FY 1985.

4.2.5.3 Core-Melt Accident Analysis

The CORCON model should be complete and the interface with CONTAIN refined by the first quarter of FY 1983. The CONTAIN code is expected to be a verified code by the fourth quarter of FY 1984. CONTAIN will then be interfaced with input codes such as TRAP-MELT that supply fission product and energy source data. The output from CONTAIN will also be interfaced with external fission product dispersion codes that compute offsite doses. These interfacing tasks will continue throughout FY 1985 and 1986. The program will continue in FY 1987 and 1988 at a much lower funding level for necessary maintenance and minor improvements.

The schedule for this element is shown on Figure 4.1.

4.3 Fission Product Release and Transport

4.3.1 Issue

NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," identified a number of key uncertainties related to estimating fission product source terms. The most important of these are:

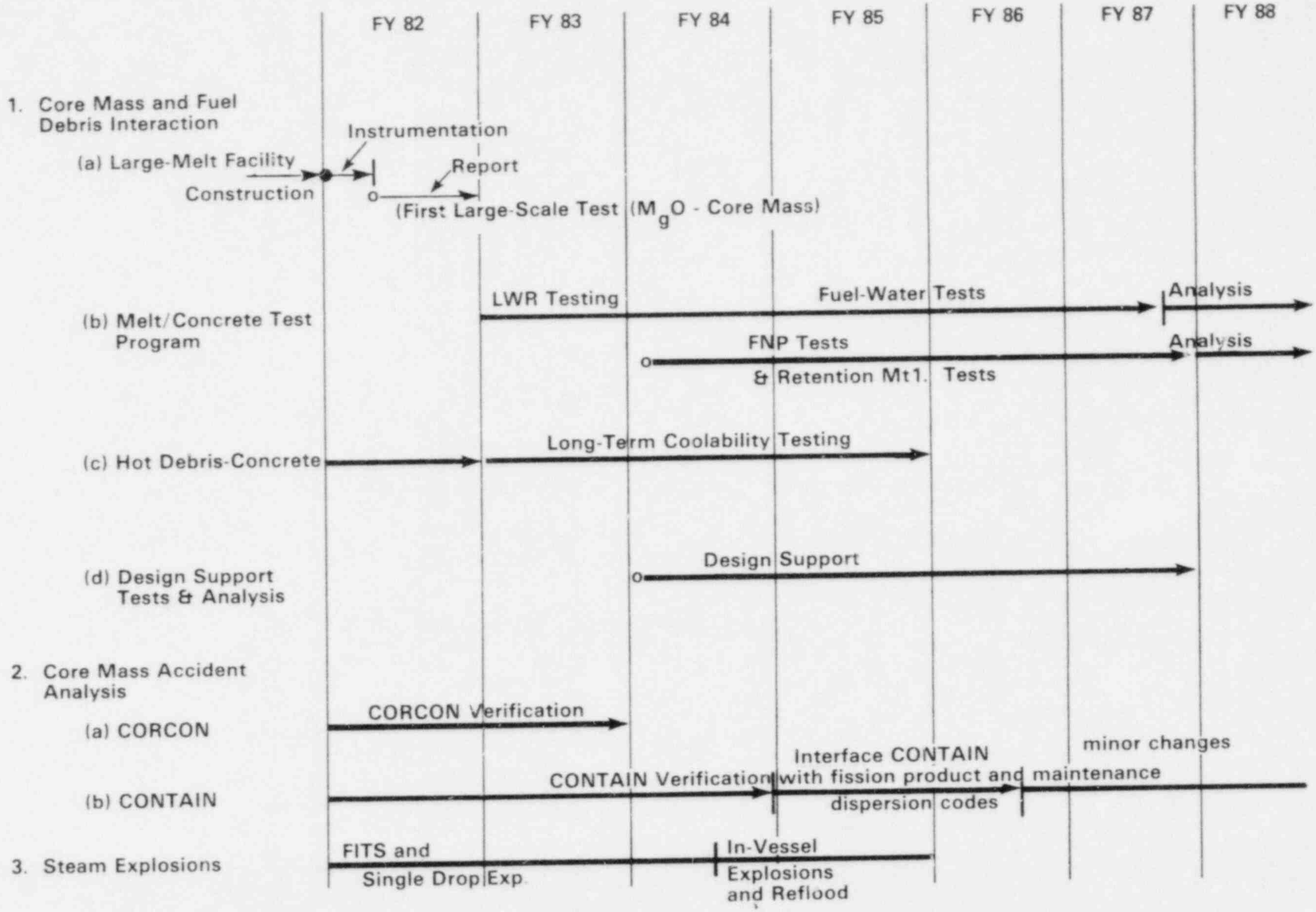

Figure 4.1
FUEL MASS BEHAVIOR

1. Reactor coolant system (RCS) aerosol and fission product behavior (experimental data for model verification).

2. RCS thermal-hydraulic models under core-melt accident conditions.

3. Containment failure time, mode, and location (experimental data and analysis).

4. Fission product vapor phase and aqueous phase chemistry (experimental data).

5. Less volatile fission product, control material, and structural material aerosol formation rates (in-vessel and during interaction with concrete) (experimental data).

6. Aerosol behavior in condensing steam containment atmospheres (experimental data).

7. Removal of particulate fission products in water pools and ice beds (experimental data and models).

8. The effect of a hydrogen combustion on fission product physical and chemical forms (experimental data).

9. Coupled models of containment fission product vapor transport, aerosol behavior, steam effects, and effects of engineered safety features (ESFs).

4.3.2 Research Program Objective

The objective is to develop a data base for assessing fission product release from the fuel and fission product transport behavior during transport from the fuel to the environment. This research will focus on severe core damage and core-melt accident conditions. The data base needs include information on the release of fission products and nonradioactive aerosols from overheated and melting fuel, the chemistry of the released fission products, aerosol formation mechanisms, the transport behavior of fission products and aerosols in the RCS and in the containment, and the effectiveness of ESFs in mitigating fission product release under severe-accident conditions.

4.3.3 Relationship to Other Programs

Related research programs on fission product release and transport are being conducted by the following organizations:

Organization	Research Description
RES Division of Risk Analysis	Development of improved CORRAL code (MATADOR and development of improved CRAC (Calculation of Reactor Accident Consequences) code for predicting health effects.
RES Division of Engineering Technology	1. Evaluation of fission product mitigation performance of ESFs under severe core damage and core-melt accidents.
	2. Development of improved inplant source terms for safety equipment qualification.
Electric Power Research Institute	1. Large-scale experiments and model development on fission product and aerosol attenuation in water pools.
	2. Fission product chemical form and release rates from high-temperature fuel.
	3. Assessment of containment aerosol code performance against experimental data.
	4. Participation in international program planning for primary system aerosol transport tests in the Marviken facility.
Federal Republic of Germany	1. Development and improvement of the NAUA aerosol transport code.
	2. Tests on fission product release from overheated fuel and during melt/concrete interactions using simulated irradiated fuel in the SASCHA facility.
U.S. Department of Energy	The DOE is developing a plan to initiate research in the fission product release and transport area. Data needs in this area have been identified in a draft report (NUS-3808), "Source Terms: An Investigation of Uncertainties, Magnitudes, and Recommendations for

	Research" (October 1981). DOE plans to have workscopes developed for specific research programs by mid-FY 1982.
Industry Degraded Core (IDCOR) Steering Group	A program of limited scope to review existing experimental fission product release and transport behavior information is planned by IDCOR in support of source-term assessments.

4.3.4 Background and Status

An intensive program to evaluate realistic source terms under severe LWR accident conditions was conducted during the Reactor Safety Study. 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 severe accident conditions and to develop models for fission product transport behavior within the reactor coolant system. These programs have provided: (1) data on fission product escape from fuel rods under LOCA conditions in the temperature range of 500°C to 1600°C and (2) a mechanistic model (TRAP-MELT) for fission product behavior within an LWR primary system under severe-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.

In the area of aerosol behavior within containment structures, significant progress that is broadly applicable to all aerosol studies has been made under the fast-reactor program. 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 MAEROS.

The following three sections describe specific research projects and results expected during FY 1982 and FY 1983.

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. First results at high temperature (2000°C) are scheduled for FY 1982 with the higher-temperature tests (to 2600°C) to begin in early FY 1984.

2. Experiments to investigate the release of fission products and structural material aerosols from larger bundles of fuel (0.5 to 10 kg) using simulated irradiated fuel (fissium) and out-of-pile heating techniques (FY 1982-1983).

3. 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).

4. Examination and analysis of samples of the TMI-2 core (schedule to depend on the TMI-2 cleanup schedule.) This program will be conducted by DOE with the technical assistance of NRC and NRC contractors.

5. Development and improvement of mechanistic models to predict the release of fission products from the fuel under accident conditions (FASTGRASS and START) and during interactions of the damaged and molten fuel with residual coolant and plant structures.

6. Measurements of fission product release during Phase 1 SFD testing in the PBF reactor (FY 1982 and FY 1983).

Research programs in the areas of fission product vapor and aerosol transport and deposition include:

1. Continued improvement of the TRAP-MELT code (models fission product behavior within the primary reactor coolant system under severe-accident conditions) and the coupling of the mechanistic, multicompartment TRAP-MELT RCS code to models that predict containment fission product (and aerosol) behavior and models for fission product release from the core (ongoing, to be completed in FY 1984). Results from this program will be factored into the CONTAIN code.

2. An experimental and analytical program 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 1983, but may be extended).

3. Continuation of experimental and analytical programs to develop models for containment aerosol fission product behavior under severe-accident conditions. The aerosol models will be incorporated into the TRAP-MELT, CORRAL, and CONTAIN codes to predict overall fission product transport behavior (to be completed in FY 1983).

4. A series of small-scale experiments to be initiated to provide data for interim verification on the TRAP-MELT code. These experiments will also be directed toward investigating the potential for resuspension of

deposited aerosols from RCS surfaces. This program was initiated in FY 1982 and will be completed in FY 1983.

5. Modification and operation of a large-scale facility to test and verify the primary system fission product and aerosol transport codes. Tests on volatile fission product (e.g., cesium, iodine, tellurium) transport will be initiated in FY 1983 and completed in FY 1984.

6. An experimental program to investigate the chemistry of various fission product species (currently focused on various forms of iodine and tellurium) in aqueous reactor solutions and their liquid/vapor phase distribution under representative accident conditions.

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. Within this decision unit is a program to investigate and quantify the radioiodine retention performance of impregnated activated charcoal absorbers under accident conditions (complete FY 1982).

The NUREG-0772 follow-on research will consist of:

1. Development of updated severe-accident release-from-plant fission product source terms to replace WASH-1400 estimates (complete FY 1983) for use in developing siting policy, risk assessment, emergency planning guidance, etc.

2. Development of estimates of the uncertainties associated with these source term predictions and major sources of the uncertainty (complete FY 1983).

3. Analysis of past reactor accidents and reactor destructive tests to gain insights into fission product release and transport behavior and to compare current assumptions and models with measured releases (complete FY 1982).

4.3.5 Research Program Plan

4.3.5.1 Fission Product Release from Overheated Fuel

Beginning in FY 1984, the Phase 2 tests will be initiated in the high-temperature fission product release program to investigate the release of fission products and aerosols from commercially irradiated fuel in the temperature range from 2000°C to approximately 2600°C. Two tests series will be conducted in FY 1984, three in FY 1985, three in FY 1986, and two in FY 1987. Final program reporting will be completed in FY 1988.

4.3.5.2 RCS Fission Product and Aerosol Transport Tests

The tests on RCS fission product and aerosol transport will continue through FY 1985 and perhaps into FY 1986. In FY 1985 this experimental program will focus on determining the transport behavior of high-density aerosols within the

RCS. Tentative plans call for tests with up to 800 kg of prototypic core-melt aerosol materials.

4.3.5.3 Fission Product Transport Code (TRAP-MELT) Development

Pretest and posttest analyses of the RCS tests discussed above will be conducted with the TRAP-MELT code. Code predictions and experimental results will be compared and model improvements initiated (if deemed necessary) to correct deficiencies in the code. These analyses and model development activities should continue through FY 1986. At the end of FY 1986, the TRAP-MELT code will have been tested and validated by comparison with these large-scale integral tests.

Similar analyses will be performed using the extended TRAP-MELT code, the CONTAIN code, or the MELCOR/MATADOR code on planned large-scale containment fission product and aerosol tests (to be conducted in the FRG). These analyses should also be completed by FY 1986.

4.3.5.4 Updated Best-Estimate Release-from-Plant Source Terms

Beginning in FY 1986 and continuing into FY 1987, systematic analyses will be performed using available core-melt system models (e.g., MELCOR) and the improved and validated fission product transport models. These analyses will determine best-estimate source terms for risk-dominant LWR accident sequences for the various PWR and BWR plant designs. Uncertainty analyses will be performed and revised estimates of uncertainties associated with these source-term estimates will be determined. It is expected that the uncertainties associated with these revised estimates will be much smaller than those determined from the NUREG-0772 follow-on studies being conducted in FY 1982 and FY 1983.

4.4 Severe Accident Mitigation

4.4.1 Issue

For the general objective of reducing residual risk, a basis is required to assess the value of features to mitigate severe accidents. The final barrier to protect the public is the containment. This unit addresses the issue of containment integrity and the ability of mitigation systems or devices to improve this capability, particularly for severe accidents.

4.4.2 Research Program Objective

The objective is to determine the design criteria for features that can prevent or mitigate containment failure and the relative costs and benefits of such features.

4.4.3 Relationship to Other Programs

The basemat melt-through segment of this element is large-scale testing with only the KfK thermite program on a comparable scale. The latter program is expected to start testing in 1985. The hydrogen control program is closely coordinated with the EPRI effort on hydrogen control. The feasibility and desirability of joint EPRI/NRC large-scale tests are being studied.

4.4.4 Background and Status

4.4.4.1 Improved Containment

This research includes the feasibility and utility of (1) filtered vented containment, (2) containment cooling, and (3) increased containment capacity. This work is closely related to the aerosol source-term development work. Substantial analysis will be completed in FY 1983 but the bulk of the program will continue into FY 1984.

4.4.4.2 Core-Retention Systems

Associated evaluations of core retention or core-melt interdiction in LWRs are in progress. Retention beds of interlocking refractory bricks are being tested, and an engineering evaluation of brick construction for core-melt impingement will be completed in FY 1982-1983. Initial feasibility studies of rubble beds for core-melt interdiction have been made and large ThO_2 bed coolability studies will be completed in FY 1983.

4.4.4.3 Hydrogen Generation and Control

The program on hydrogen generation is currently providing analytical studies of hydrogen control systems for various plants and containment types, including the effects on safety-related equipment. Also included is an effort to quantify the amount of hydrogen generated from corrosion reactions initiating from plant sprays onto galvanized, organic, and aluminum coatings. Key technical questions have arisen as to the timing of the release, the amount, the transport or mixing of gases in containment, and the potential for a transition from simple deflagration to detonation.

EPRI is currently sponsoring a large effort on hydrogen control, including both analysis and experimental research. The EPRI experimental program includes combustion limits in the presence of steam, mitigation, equipment survival, and hydrogen transport. DOE has a limited program, currently concentrating on hydrogen detectors. Foreign programs are being initiated on hydrogen transport and mixing.

Major NRC program components are listed below:

1. Accident analysis calculations will be performed with the MARCH code for the Zion, Sequoyah, and Grand Gulf plants.

2. An improved multicompartment deflagration code, HECTR, will predict pressure and temperature histories during and after a hydrogen deflagration in the presence of steam and other gases.

3. Based on the accident analysis results, calculations will be performed for local regions in the containments where detonations are considered possible; the potential for missiles will be assessed.

4. A manual will be prepared and published on the behavior of hydrogen as a guide for use in the preparation of plant-specific operator emergency manuals.

5. A two-pronged attempt aimed at modeling accelerated flames will be initiated. One attempt is to produce a model based on the underlying physics and the available experimental data. The other will employ existing computer codes for combustion analyses.

6. The first two test series will be conducted in the FITS facility to investigate deflagrations and detonations in ternary mixtures of hydrogen/air/steam. Scoping tests on the effects of aerosols on igniter performance will be performed.

7. The steam/hydrogen jet facility will be checked out, and the first two test series will be performed to study autoignition, flame characteristics, and stability (including the effects of flame holders).

8. The Variable Geometry Experimental System (VGES) 16-foot tank facility will continue to provide scoping information on combustion phenomena. Tests will address mitigation effects, flame acceleration, and direct initiation of detonation.

9. The construction of the flame-acceleration facility will be completed. Experiments will be initiated to study flame acceleration as a function of obstacle characteristics. Experiments will then begin to investigate detonations. These tests will be closely coordinated with the bench-scale tests being performed at McGill University and both will be compared to available analytical models.

10. Safety-related equipment will be procured and tested in the FITS, VGES-16 tank, and radiant heat facility to characterize the response of this equipment to hydrogen deflagration in the presence and absence of steam.

11. Experiments to determine the corrosion kinetics of galvanized, aluminum, and organic coatings in containment will be performed.

4.4.5 Research Program Plan

4.4.5.1 Improved Containment

The aerosol source-term work will continue through FY 1983. Studies of existing plants and separate effects tests of containment features will largely be concluded in early FY 1984. This program will evaluate the costs and benefits of features to prevent or mitigate containment failure and will develop criteria for evaluating engineering designs of the features. Results, recommendations, and specifically requested analyses will be the major efforts in FY 1986 and 1987. The program will essentially be completed by FY 1988.

4.4.5.2 Core-Retention Systems

Research to study core-retention concepts is tied to the need to interdict the basemat-failure pathway to public exposure and to the prevention of abovegrade containment failure from the additional loads from combustible gases generated in the core/concrete interaction. The relative priority of this work in the later years will depend on regulatory requirements for data to support the decision whether or not to require core retention devices.

The assessment effort for the core-retention concept will include investigations of castable refractory materials and engineering evaluations of cooling systems around core-retention devices in FY 1984. The modeling for the CONTAIN code of retention device performance should also be completed. FY 1985 efforts will address standardized concepts and will evaluate long-term heat transfer to steel structures representative of offshore power plant designs. Coolant/melt-retention-device interaction studies will be carried out in FY 1984-1986, and overall engineering evaluations will be completed in FY 1982. Potential applicable design concepts will be evaluated in FY 1987; design support evaluations will be conducted in FY 1988; and final reports and termination of the work are scheduled for the following fiscal year.

4.4.5.3 Hydrogen Generation and Control Program

This program will provide an assessment of hydrogen control systems for various plant and containment designs in line with the requirements in the Interim Hydrogen Rule.

Analysis of accidents involving hydrogen for specific plants or standardized plant designs will be performed as requested using the deflagration code HECTR and a validated hydrogen transport code. The work to assess the potential threat of missile generation will be completed in FY 1984. Experiments and model development will continue in the area of flame acceleration to include scale and geometry effects. The work on aerosol effects on hydrogen control systems will be completed. Experiments on the radioanalysis of water with impurities or under boiling conditions will be completed in FY 1984 and an update of the hydrogen compendium will be published. Experiments on equipment survival should be completed in FY 1984 and should result in standard methods for testing equipment under hydrogen burn conditions. A final report on the corrosion of galvanized aluminum or organic coated materials in containment will be completed in FY 1984.

In FY 1985, analyses of hydrogen behavior for specific plant designs will continue as necessary, and experimental studies will continue in the area of flame acceleration. Tests in collaboration with EPRI or complementary to the EPRI tests will continue. Remaining tests in the VGES 16-foot tank will be completed. The testing in FITS will be directed toward the quantification of hydrogen vs. steam generation rates for various scenarios of core slumping and steam-steel interactions. A third international workshop will be conducted to assess the level of understanding on hydrogen behavior and control issues.

It is anticipated that most of the work on hydrogen generation, transport, behavior, and control will be completed by the end of FY 1987 unless there are specific problems arising out of the experimental and analytical work up to that date. Any large-scale proof tests on hydrogen behavior and control systems in containment will be completed during FY 1986 and 1987. The testing in FITS to verify hydrogen source terms for various accident scenarios should also be completed during this period. It is expected that only the support function for new or modified plants will exist in FY 1988.

The schedule for this element is provided in Figure 4.2.

4.5 Severe Accident Sequence Analysis (SASA)

4.5.1 Issue

This element addresses the problem of improving the understanding of reactor accidents both within and beyond the design basis with the goal of developing better strategies to prevent, manage, and mitigate severe accidents. The issues being resolved by this element are:

1. Severe accident policy,

2. Licensing and safety concerns generated by NRR abnormal transient operator guidelines (ATOGs),

3. Operator instrumentation information needs,

4. Fission product release and transport,

5. NRC unresolved safety issues (USIs) such as

    a. Station blackout,
    b. Shutdown decay heat removal requi· ·ents,
    c. Anticipated transient without scram (ATWS),
    d. Safety implication of control systems and systems interaction in nuclear power plants,
    e. Hydrogen control measures and effects of hydrogen burn on safety equipment, and

6. Increased capability of NRC emergency response.

4.5.2 Research Program Objective

The objective of the proposed program is to improve understanding of reactor accidents and of the human/machine interface during a broadened spectrum of accident sequences, including those within a beyond-design-basis limit. Particular emphasis is to be placed on the perceptions of the operator, the operator needs for information, the alternative actions the operator might take given various combinations of component failures, the effectiveness of these actions, the influence of multiple failures on plant safety system functional capabilities, the ability of degraded safety systems to be used to bring the

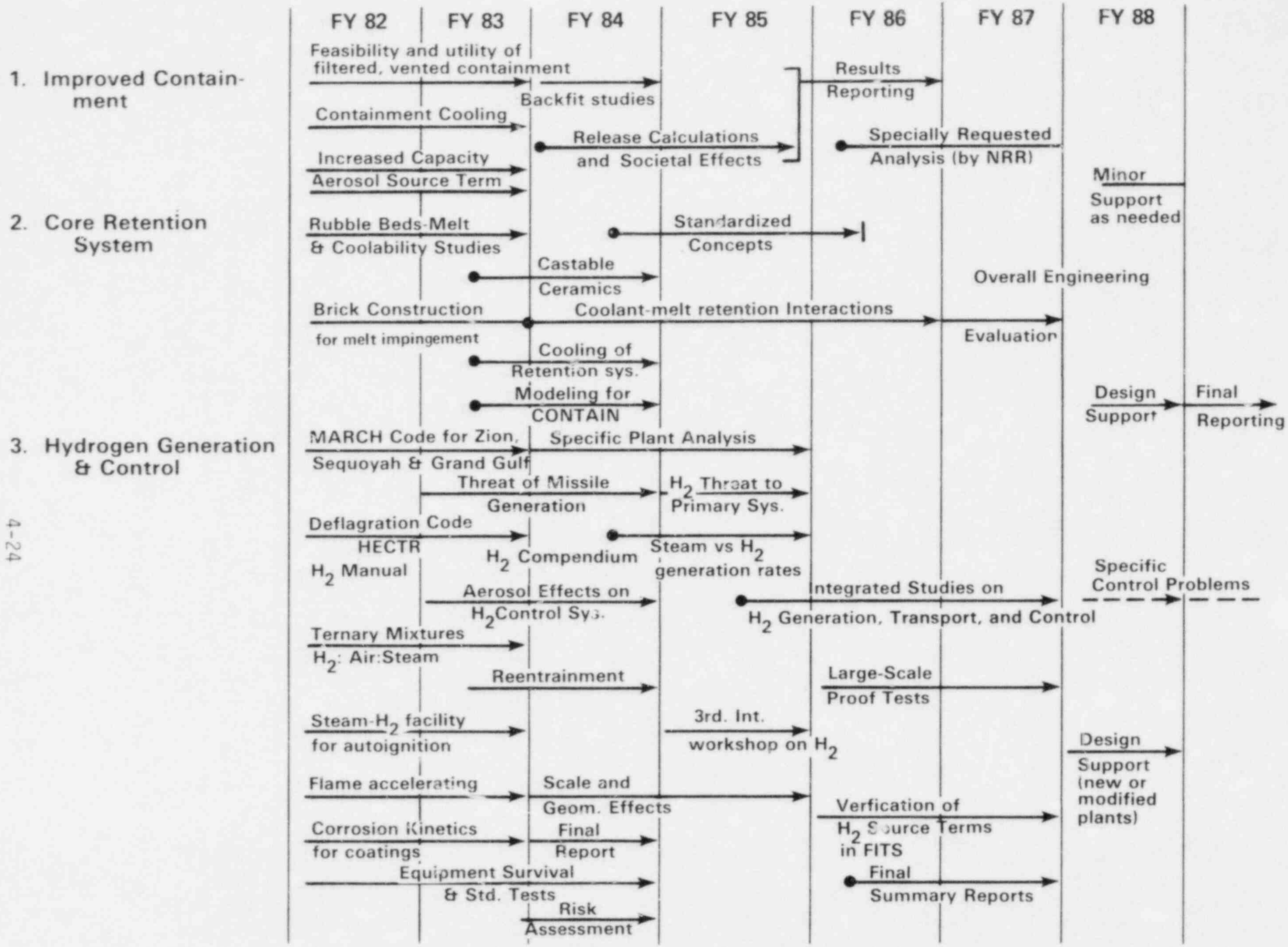

Figure 4.2 Severe Accident Mitigation

plant to a safe shutdown condition, and the environment in which safety systems will be required to survive.

4.5.3 Relationship to Other Programs

Risk assessment programs such as the Interim Reliability Evaluation Program (IREP) will define high-risk sequences for SASA consideration. The SASA program will in turn characterize these sequences and evaluate the impact of design changes and operator actions on the outcome of the sequences. These detailed analyses will be fed back into risk assessments to improve the assessment of risk under the revised conditions. The SASA program will also define environments necessary for system or component function to support mechanical and electrical equipment qualification rulemaking.

Test programs such as PBF, Semiscale, LOFT, and FIST will be used to provide data to improve the resolution of issues of concern to be addressed by SASA. Such test programs will also produce data that can be used to evaluate SASA analysis results in an effort to evaluate SASA products.

Licensing and safety concerns generated by NRR ATOG and other licensing reviews will also serve to define SASA issues resulting from identified deficiencies in the symptom-oriented review.

A cooperative exchange of information between the SASA efforts and the ongoing French-related effort has been established.

4.5.4 Background and Status

Small breaks, loss of ac power, large LOCAs, interfacing-system LOCAs, and loss-of-feedwater transients have been analyzed to perform pertinent evaluations of numerous accident strategies associated with these categories of sequences. The loss-of-ac-power analyses are assisting in the resolution of the Station Blackout USI.

An in-depth analysis of the behavior of a representative Westinghouse four-loop plant (Zion I) was completed for small-break, loss-of-ac-power, and loss-of-feedwater scenarios. The completion of this analysis provided valuable insight into the response of this plant design to scenarios in categories of concern to NRC licensing and research.

The analysis of the response of a BWR and the analysis of fission product noble gas and iodine transport under station blackout conditions were completed. These analyses considered the impact of the availability and unavailability of various cooling systems. Additional analyses are in progress addressing blackout behavior using advanced thermal-hydraulic codes.

Programs allied to SASA such as plant status monitoring have developed methodologies for identifying instrumentation necessary to monitor PWR and BWR status. Such methodologies interface appropriately with a program such as SASA, which encompasses the identification of operator information required to properly manage accidents and transients.

A SASA calculation log was established. This log will be expanded in the future to become a handbook of accident signatures that can be used to improve simulator and other operator training programs.

Symptom-oriented procedures have been used in SASA loss-of-feedwater analyses to begin to assess the adequacy of these procedures.

The completion of the sequence analysis to date has developed a continuously expanding data base of great value to other programs. This base is being used to develop operator action event trees that can be used to define appropriate operator action for a variety of scenarios. The data base can also be used to evaluate the accuracy of the PRA methodology, which will probably play a role in the future process for plant licensing. SASA possesses a unique capability and position in developing this data base.

Work is in progress in FY 1982 and will continue in FY 1983 to analyze the thermal-hydraulic, fuel, and fission product transport phenomena in eight aging PWRs with a hypothesized reactor vessel break because of a sensitized flaw, thermal shock, and cold repressurization. The analyses will assess mitigative actions whose purpose is to maintain containment integrity.

Work is in progress in FY 1982 to address the analysis needs in support of operator guidelines for responding to transients and accidents, including:

1. Depressurization capability in Combustion Engineering (CE) plants without power-operated relief valves (PORVs),
2. Tube ruptures in multiple steam generators,
3. Babcock and Wilcox (B&W) NSSS design features,
4. Emergency guideline development for ATWS events,
5. Unmitigated boron dilution event, and
6. High point vents.

4.5.5 Research Program Plan

4.5.5.1 Evaluation of Procedural and Plant Configuration Changes Supporting the Severe-Accident Rule for Nuclear Reactor Development

This research includes evaluating (1) system functional requirements to assess prevention and mitigation of a core-melt accident in severe accidents involving multiple system failures, (2) equipment and system survivability in severe-accident environments, and (3) the impact of proposed prevention and mitigation features on severe-accident sequences in which these features were not primarily intended to function.

This research will analyze risk-assessment-based dominant sequences and evaluate sources of uncertainty and the accident management effectiveness of potential plant system and procedural modification.

This effort is currently in progress with final documentation and program completion to follow in FY 1985.

4.5.5.2 Evaluation of Procedural and Plant Design Changes for ATWS

This research includes assessing the effectiveness of potential procedural and plant design changes to ensure the acceptability of the consequences of an ATWS. The principal tasks include assessment of operator guidelines and evaluation of the effectiveness of both operator and plant design remedial options. This effort will assist in the elimination of unnecessary and ineffective changes in procedures and plant configuration. This program will peak in FY 1984, with the bulk of its completion dependent upon the Final Rule and Regulatory Guideline Development Schedule.

4.5.5.3 Evaluation of Safety Implication of Control Systems

This program includes analyses to characterize plant responses to accidents or transients whose severity is potentially increased by control system failures. The SASA program will analyze such cases as the thermal-hydraulic response for the steam generator overfill transients in PWRs and the reactor overfill transients in BWRs. This effort will peak in FY 1984 with the bulk of the program's completion to be determined when NRR has completed the Task Action Plan.

4.5.5.4 Radionuclide Transport in Severe Accidents

This research includes analyses to (1) define fission product transport pathways within the reactor system, containment, and reactor building, (2) evaluate the assumptions required to estimate the rates of fission product movement throughout the plant, and (3) estimate the fission product inventories and transport rates. This effort will assist in reviewing licensing requirements with respect to fission product transport and source-term models. It will also provide a basis for modifying the models to expedite plant licensing. Some results will be available in FY 1984, with the bulk of the program's completion dependent upon associated experimental work described in Section 4.3.

4.5.5.5 Evaluation of Plant Abnormal and Emergency Operating Procedures

This research includes evaluation of various operator actions for their effectiveness in preventing and mitigating severe accidents in both PWRs and BWRs. Examples include (1) depressurization of CE plants without PORVs, (2) diagnosis and management of steam generator tube rupture sequences, and (3) accident diagnosis and management of B&W plants. This effort will assist in resolving concerns that stated actions might prove to be ineffective or counterproductive in preventing core uncovery and a subsequent fuel-damaging accident. This research will continue to study the impact of current and future abnormal and emergency guideline changes on accident prevention and mitigation for the next 5 years and beyond. Final results and documentation for each action studied will be provided upon completion.

4.5.5.6 Identification and Implementation of Operator Information Needs

This program will provide analyses to assist in the determination of the need, range, and justification for instrumentation required by plant operators to unambiguously monitor plant status during an accident or a transient. The program will also determine the need, range, and justification for these instruments. This effort will assist in providing a vital link in the process emanating from sequence identification and culminating in a safety parameter display system. This process consists of sequence analyses, sequence signature development, and accident/transient diagnostic methods. This effort will peak in FY 1984. Complete accident signatures will be developed by FY 1986. Algorithm (process) to be used by the operator to prevent, diagnose, and respond properly to reactor accidents will be developed by FY 1984. Final results and documentation are to be provided in FY 1988.

4.5.5.7 Characterization of Plant Behavior Under Complex Transient Conditions in Conjunction with Multiple Failures

This research will perform operational, thermal-hydraulic, and fission product transport analyses for sequences to be identified by (1) risk assessment programs, (2) the severe-accident research program, (3) test programs such as PBF, Semiscale, LOFT, and FIST, and (4) NRC unresolved safety issues. This effort will assist in defining plant behavior, in evaluating procedural and plant design effectiveness, in defining plant environments, and in evaluating functioning system effectiveness with respect to preventing or mitigating a severe accident. This effort is currently under way, peaking in FY 1984, with final results, documentation, and program completion in the following 4-5 years.

The schedule is shown in Figure 4.3. The completion dates and interfacing activities are based on current plans documented in draft rulemaking plans, status reports of USIs (NUREG-0606), and status reports of TMI Action Plan items.

	FY 1984	FY 1985	FY 1986	FY 1987	FY 1988
Evaluation of Procedural and Plant Configuration Changes Supporting Severe Accident Research	Define the conditions for most severe accidents. Define operator actions in preventing or mitigating the accidents	Documentation	Complete Research and Development		
Evaluation of Procedural and Plant Design Changes for ATWS	Assessment of operator guidelines	Develop effectiveness of operator & plant design remedial options		Completion dependent upon schedule for Final Rule and Regulatory Guideline Development Schedule	
Evaluation of Safety Implication of Control Systems	Analyze plant T-H response for SG overall transient for PWRs	Same analysis for reactor overfill transient in BWRs		Completion date will be determined when Task Action Plan completed within NRR	
Definition of Radionuclide Transport Characteristics	Define FP transport pathways within reactor system, containment reactor building	Evaluate rate of FP movement in the plant		Completion date dependent upon resolution of degree of conservatism of current models	
Evaluation of Plant Abnormal and Emergency Operating Procedures	Work to address analyses needs in support of operator guidelines (a) as requested by NRR and (b) as required within SASA				
Resolution of Licensing Concerns		Analyze plant grouping, e.g., ANO-1			Documentation
Identification and Implementation of Operator Information Needs	Provide analysis for by plant operators		Complete accident signatures		Complete documentation develop. algorithm
Characterization of Plant Behavior Under Complex Transient Conditions in Conjunction with Multiple Failures	Complete thermal-hydraulic, FP transport analyses identified by Risk Assessment Program		Define plant behavior and plant environments; evaluate procedure and plant design effectiveness		Documentation

Figure 4.3. Severe Accident Sequence Analysis

5. ADVANCED REACTORS

As in the past, this decision unit comprises research on the safety problems of reactors not yet in the commercial licensing stream. Most of the safety research is therefore carried out by the developers and the government agency charged with development, the Department of Energy (DOE). The NRC effort is devoted to developing the necessary expertise and regulatory tools to support the licensing audit of the Clinch River Breeder Reactor (CRBR) plant.

This decision unit contains two elements:

1. Liquid-metal-cooled fast-breeder reactors (LMFBRs), wherein the formerly generic program has been narrowed to concerns specific to the licensing of the CRBR. Any long-range program will be responsive to objectives of the Administration for long-term research within DOE.

2. High-temperature gas-cooled reactors, wherein the formerly generic program is focused on the preapplication review of a proposed prototype plant and on gathering data crucial to early licensing decisions on such an application, should it develop.

5.1 Liquid-Metal-Cooled Fast-Breeder Reactors

5.1.1 Issue

Congress and the President have given approval to the CRBR project, and licensing review of the plant by NRC for a construction permit has resumed.

An ad hoc Research Review Committee on Fast Reactor Safety Research chaired by the Chief Scientist of the Office of Nuclear Regulatory Research has made a review of the licensing issues and needs. A number of safety issues that need to be assessed by the NRC were identified. Some of the more important of these are:

1. Decay heat removal by natural convection cooling.

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

3. Consequences of failures in the heat transport system, e.g., broken pipes and frozen pumps, on thermal margins.

4. Consequences of complete loss of offsite and onsite power.

5. Interaction of liquid sodium with concrete and the effect of sodium fires and hydrogen combustion on the containment integrity.

6. Consequences of core meltdown and challenge to the containment structure by molten fuel.

7. Consequences of failure of the plant protection system.

8. Consequences of malfunction of the plant control system.

9. Definition of radiological source term.

10. Identification of safety research that DOE should perform.

In addition to the safety issues above, there are a number of areas that need consideration by NRC for licensing assessment where new programs may be needed in FY 1984 through 1988. These are:

1. Evaluation of LMFBR instrumentation for reliability and capability to meet regulatory requirements.

2. Evaluation of the adequacy and reliability of control and shutdown equipment for LMFBRs.

3. The impact of human factors on the regulatory requirements for LMFBRs.

4. Occupational radiation protection at LMFBR facilities.

5. Identification of areas where regulations need to be augmented for future advanced reactors and design and/or acceptance criteria need to be established.

5.1.2 Research Program Objective

The major objective of the program over the next few years is to provide NRC with data and analytical tools to make the licensing decisions necessary for CRBR licensing. The longer-range objective of the program in FY 1984-1988 will be to complete research needed for the CRBR operating license review and on a schedule consistent with Administration and Congressional actions and to develop a basis for licensing a large commercial LMFBR that is likely to be significantly different in design from the CRBR.

The objectives of the major experimental programs are the resolution of issues in which there is significant disagreement among those working in the field. They include the sodium/concrete interactions and CDA energetics and verification of core debris coolability models. The latter involves initiation- and transition-phase phenomenology and energetics work potential. Experiments proposed for DOE funding will continue to be defined in detail.

At the direction of the NRC Executive Director for Operations, a joint NRR-RES working group has been established to define a joint Technical Assistance and Research Plan to be carried out by the NRR CRBR Program Office and RES to support the CRBR licensing review.

Beyond work to support CRBR licensing review, research is needed to support the development of a regulatory position for post-CRBR LMFBRs. The NRC should provide regulatory guidance and safety advice at the design initiation stage for future LMFBRs as well as during the construction permit review stage. The nature of the research necessary to support this requirement will come from the review and hearings associated with CRBR as well as the CRBR probabilistic risk assessment (PRA) to be completed in 1984. The experience of the CRBR licensing review and PRA will be valuable in defining needed research programs, their costs, their priorities, and their schedules.

5.1.3 Relationship to Other Programs

DOE has had a program of LMFBR safety research that has been funded at the $30-50 million level for the past few years. The NRC maintains close liaison with DOE and has planned its research to deal with regulatory issues. Since the DOE is the applicant in the CRBR licensing case, it is necessary that NRC develop an independent capability for assessment of the CRBR design and develop the necessary regulatory tools for licensing CRBR. Also there are areas, namely sodium/concrete interactions and key phenomena in accident energetics, where independent experimentation has led to unresolved disagreements between NRC and the CRBR project on related safety issues.

The NRC has information exchange agreements with the U.K., Japan, and FRG on LMFBR safety research.

5.1.4 Background and Status

The LMFBR safety research program through FY 1981 has been directed at generic development and verification of safety analysis computer codes and of experimental data needed for evaluating LMFBR safety. As a result, a number of computer codes (COMMIX, SIMMER, SSC, CONTAIN, etc.) are in place for support of LMFBR licensing.

A report prepared by the ad hoc Research Review Committee on Fast Reactor Safety Research reviewed the safety issues and established priorities for their resolution.

The licensing review of the CRBR plant for a construction permit has been renewed. The RES LMFBR research program is being directed to provide support to CRBR licensing needs.

The SIMMER code at Los Alamos National Laboratory (LANL) will be used to help resolve the CDA energetics issue on CRBR licensing. Some additional development will be needed in FY 1982 and 1983 to provide the capability of analyzing heterogeneous cores.

The SSC code at Brookhaven National Laboratory and the COMMIX code at Argonne National Laboratory will be used for thermal-hydraulic assessment of the CRBR. Some additional modeling specific to the CRBR will be needed in FY 1982 and 1983.

Experimental programs on sodium/concrete interactions, key phenomena in accident energetics, and debris behavior will be continued in FY 1982 and 1983 to resolve major issues for CRBR licensing.

5.1.5 Research Program Plan

5.1.5.1 Analysis

The following detailed objectives of the SIMMER program should be met by the end of the plan period:

1. A consistent, comprehensive, and defensible method for the treatment of all significant accident sequences, including transition phase (transition from an intact core to a molten core), LOF/TOP (loss of flow/transient over power) progressions, and investigations of recriticality potentials.

2. A comprehensive and largely verified, but not proof-tested, treatment of work potential from CDA energetics.

3. Methods for the analysis of CDA that are sufficiently fast running to permit extensive parametric calculations to be used in probabilistic analysis of that portion of significant accident sequences within the primary system.

4. Integration of the CDA analysis method with similar containment analysis methods provided by the CONTAIN code.

5. A methodology for performing both detailed and simplified analyses of at least two-dimensional material motions within subassemblies, verified as thoroughly as possible by comparison to existing experimental data.

A version of SSC-L with a model of the Fast Flux Test Facility (FFTF) will be tested against operational data. A version with a model of the German SNR-300 plant exists, but operational data for verification may not be available during this 5-year plan. A model for the PHENIX reactor will be developed and tested against such operational data as are available. The long-term heat removal version of the code will be completed and verified against FFTF data.

The single-phase version of the COMMIX 3-D transient thermal-hydraulic code is complete and has been well tested against experimental data. The two-phase version of the code will be completed and verified against experimental data in FY 1984-1986. A version of the code for analysis of natural convection in whole cores will be developed in FY 1983.

The LMFBR version of the CONTAIN code will be completed and released in FY 1983. During FY 1984-1985, the code will be extended to the analysis of consequences of severe accidents for application to probabilistic analysis.

5.1.5.2 Accident Energetics and Core-Debris Behavior

Initiation-phase clad-relocation experiments in the Annular Core Research Reactor (ACRR) will be completed in FY 1984, as well as models developed for use in establishing transition-phase initial conditions. A series of experiments in ACRR on initiating-phase fuel failure and sweepout for both LOF and TOP conditions will be started in FY 1984 and will be completed in FY 1988. These experiments will use a new flowing sodium loop and the unique, newly developed, high-precision coded aperture imaging system (CAIS) for fuel motion diagnostics that makes these experiments meaningful.

Transition-phase TRANS experiments in the ACRR on molten-fuel streaming and freezing in slab (between can walls) geometry will be completed in FY 1985. Similar experiments in fuel-bundle geometry will be performed from FY 1985 to FY 1988. Analysis will be performed in FY 1985 on the programmatic need, feasibility, result quality, and cost of fission-heated experiments in the ACRR on the stability and dynamics of boiling pools of fuel and of fuel and steel with internal heat generation, using the results from previous laboratory experiments with water as a base. If the need is indicated, such experiments will be started in FY 1986 and completed in FY 1988.

Integral CDA work-potential experiments in the ACRR, both with and without sodium, will be resumed in FY 1984 and completed in FY 1986.

Modeling will continue throughout the 5-year period on key CDA energetics phenomena, using the results of CDA energetics experiments by NRC and others. The resultant models will be used in CDA analysis codes for LMFBR safety assessment.

If the need is indicated for follow-on ACRR experiments on core-debris coolability and post-dryout behavior after conclusion of the joint NRC-EURATOM-PNC program in FY 1983, such experiments will be started in FY 1984 and completed in FY 1986. Out-of-pile experiments on the characteristics of the debris from core-melt quenching in sodium will be performed in FY 1984 and FY 1985.

Modeling the dryout coolability limits and post-dryout behavior of core debris for both in-vessel and ex-vessel conditions will continue throughout the 5-year period. The resultant models will be used in CDA analysis codes for LMFBR safety assessment.

5.1.5.3 Aerosol Release and Transport

Aerosol source-term tests under sodium in the Fuel Aerosol Simulant Test (FAST) facility were begun in FY 1981-1982 and will be completed in FY 1984 leading to development of a final source-term model in FY 1984. During FY 1982 initial source-term models will be developed for the CRBR and improved models will be available in FY 1983 and 1984. These early tests (1981-1982) will be single-pin tests. Later experiments will be done with structure and multiple pins.

Core-melt aerosol source-term experiments with 5 kg UO_2 will be completed leading to a source-term model in FY 1984. Phenomenological models for aerosol source terms will be available in FY 1983 with improved models in FY 1984. The effects of control and structural material will be included in these models.

Final improvements to the HAARM-QUICK code series will be completed in FY 1982 and code validation completed in FY 1983-1984 based on Containment Systems Test Facility (CSTF) tests (in accordance with DOE/HEDL) and Nuclear Safety Pilot Plant (NSPP) separate effects tests on core-melt aerosols. The NSPP tests will include studies on the release fraction of fission products in aerosols from burning and boiling sodium pools, sodium-concrete aerosol behavior and effects of these aerosols in engineered safety features (ESFs).

The schedule for LMFBR research is shown in Figure 5.1.

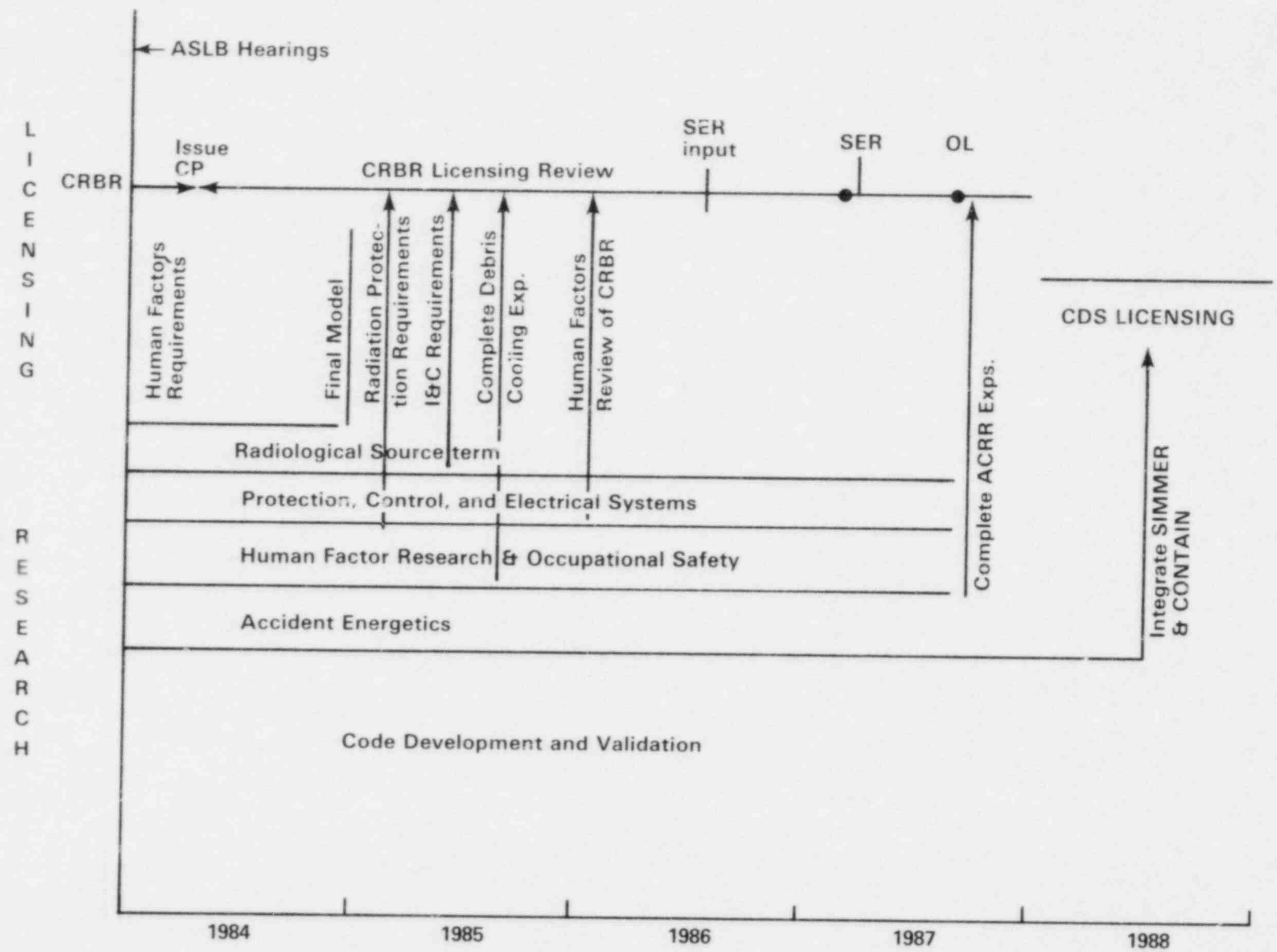

Figure 5.1 Liquid-Metal-Cooled Fast-Breeder Reactor (LMFBR) Research

5.2 Gas-Cooled Reactors

5.2.1 Issue

The U.S. Congress has authorized funding (NRC Authorizations Committee Report No. 97-22) for research to accelerate the efforts on preapplication review of high-temperature gas-cooled reactors (HTGRs). The legislation calls for development of specific topics for establishing licensing ground rules for HTGRs and for research on safety methodology and data, including efforts to eliminate uncertainties in fission product release, containment of radioactive products, and site dose calculations.

The preapplication review is to focus on developing licensing review bases specifically appropriate to HTGRs. This should include general design criteria, siting criteria, siting source terms, assessment of basic standards, recommended changes to the Standard Format and Content of Safety Analysis Reports for HTGRs, development of NRC positions on high-temperature code cases, regulatory guide modifications, and the development of a base of physical data, computer codes, and design and engineering information so that the technical basis for licensing HTGRs is clear.

Specific technical and safety issues for new generation HTGRs will be more accurately identified when an initial lead-plant design is provided to the NRC. In the intervening period, some specific issues are being identified through the results of newly initiated probabilistic risk assessment efforts and current information from the industry. Certain residual issues from the operating gas-cooled Ft. St. Vrain reactor also have general application to future HTGRs. The most pressing early technical issues are the development of NRC's site suitability criteria and the determination of appropriate safety criteria for HTGR-specific systems such as the prestressed concrete reactor vessel liner and its thermal barrier.

5.2.2 Research Program Objective

The objectives of the research are to prepare NRC for licensing the next HTGR plant without significant unresolved issues affecting the process; to assist NRR in developing safety criteria for HTGRs; to provide whatever guides and standards may be appropriate on a timely basis; and to extend the scope of NRC rules to include the HTGR. Specific objectives are to identify and develop or verify the chemical, metallurgical, structural, and system performance data and methods necessary to allow the NRC to adequately assess the licensability of the HTGRs.

5.2.3 Relationship to Other Programs

This long-range plan focuses on development of data and methods for licensing a new generation HTGR; however, implicit in the plan is the continuation of research efforts directed at issues originally raised in connection with the operating Ft. St. Vrain plant. The resolution of many of these items provides data applicable to the next generation HTGR as well.

Most of the HTGR research carried out in this country is funded and guided by DOE. DOE programs for development of the reactor design also include work on reliability and design of systems and equipment that affect the safety of plant response to accident conditions. Foreign involvement (primarily Germany and Japan) with gas-cooled reactors focuses on both development and safety. NRC research seeks to cooperate with both DOE and the foreign programs to maximize the effectiveness for NRC of safety research carried out both in the U.S and abroad.

5.2.4 Background and Status

The schedule furnished by industry for a new generation of gas-cooled reactors is:

1982 ASSUMED MILESTONES FOR LEAD HTGR PLANT

Lead Standard Plant (LSP) site identified	1982
PSAR effort initiated, design concept complete	1983
Complete early site review	1984
PSAR submitted	1985
Construction permit granted, begin LSP construction	1987
FSAR submitted	1991
LSP operating license issued	1993
Commercial operation	1994

NRC is developing in FY 1982 and 1983 an improved baseline for HTGR licensing. This encompasses the initial efforts at resolving licensing readiness issues identified above. Probabilistic risk analysis (PRA) is being employed to better identify the source terms for HTGR siting and to refocus the research priorities. General design criteria specific to HTGRs are being identified, and regulatory guides specific to HTGRs are being developed. The industry positions on safety systems and analyses as given to NRC for the lead plant siting are being evaluated.

The industry's Accident Initiation and Progression Analysis (AIPA) developed in support of their safety and licensing positions is being assessed for elements of usefulness and applicability to the NRC PRA efforts. These efforts will trace potential initiating events and evaluate consequences with respect to siting suitability, appropriate siting criteria, and potential siting rule modifications.

Experimental and analytical efforts that had been ongoing in the RES gas-cooled reactor safety research program prior to 1982 are being continued. These efforts are useful in the long-range development of licensing base data and techniques and include fission product transport and deposition behavior in the reactor materials and system, better understandings of the structural material performance limitations, and better capability of evaluating system transient performance under accident conditions. Particular attention is given to development of criteria for and behavior of the structural graphite core support system.

5.2.5 Research Program Plan

Consistent with the above-described congressional mandate for gas-cooled reactor work, the current plan for HTGR safety research is designed to focus on a number of key elements each year as available funding permits. As such, the plan is to maintain a base of readiness within the NRC, and at the involved supporting laboratories, to handle issues that may arise in connection with the operating Ft. St Vrain plant and to be capable of expanding the safety research efforts when and if required for an actual plant license application. The following FY 1984-1988 program is predicated on a continuation of the funding of about $2.5 million per year for several years.

5.2.5.1 Analysis and Licensing Preparations

1. Licensing Preparation. Plans are being made in FY 1982-1983 for review and modification as necessary to standards and guides, beginning with the 1973 draft of the HTGR edition of the Standard Format for Safety Analysis Reports, in FY 1984 and later. Attention will be given to criteria for inservice inspection of structural graphites and of components and structures within and including the primary system boundary. (Investigation, development, and potential use of improved inspection techniques and instrumentation will be considered.) Regulatory criteria will also be considered for prestressed concrete reactor vessel (PCRV) liner cooling capacity and redundancy. Plans will be implemented later for cooperation with industry in standards development such as ASTM standards and ASME Code Sections III and XI as related to HTGR technology. To the extent possible, inservice inspection requirements for thermal barrier, PCRV closure design, and containment requirements will be identified in FY 1984-1985.

   Review and coordination will be made with safety research and evaluations performed by other countries, especially Germany and Japan. In particular, information of use to U.S. programs and information gaps that may be investigated either in the U.S. or abroad will be carefully identified.

2. HTGR Safety/Licensing Handbook. Development of an HTGR Safety/Licensing Handbook will be initiated in FY 1982-1983. This handbook will draw on the large amount of fundamental information being generated by the Fort St. Vrain project that is important to the support of long-term Fort St. Vrain operations and is of generic application to advanced HTGRs. The handbook will contain guides, standards, data, and analytical techniques in an organized and concise manner. The development of the structure of the handbook will be aided by an HTGR edition of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition." The contents will be continually upgraded in the FY 1984-1988 time period.

3. Probabilistic Risk Analysis. Efforts in the mid-1970's to rank safety issues in order of priority and identify research needs used PRA methods in the industry AIPA study. That study and some German work in the PRA area are being reviewed during FY 1982-1983 and applied to safety considerations for the currently anticipated lead HTGR plant. New information from this survey of important accident sequences and safety issues will be factored into the licensing evaluations and the planning for detailed risk analysis in the FY 1984-1985 period.

4. Systems Analysis. The system transient analysis efforts make use of the CHAP code at LANL and ORTAP, CORTAP, and ORECA at Oak Ridge National Laboratory (ORNL). The CHAP code will be used in conjunction with other codes such as ORTAP and ORECA to identify and develop additional computational capabilities required for accident consequence evaluation appropriate to a probabilistic analysis of HTGR safety during the FY 1983-1984 period.

   Ongoing and planned LMFBR and LWR safety research will be surveyed for applicable work, and plans will be made for the use of this research in this program.

   Because of the potentially high temperatures of the helium coolant in accident situations, it is necessary to establish time and temperature relationships for all critical components of the primary system as well as for the fuel during emergency core cooling conditions. Continued investigations of convective flow mixing and natural convection phenomena, including hot streaks, stratification, and plume effects, are needed for design-dependent cases. Means for benchmarking analytical techniques will be established. Improved understanding of conditions relating to the transition between laminar and turbulent flows will be developed in the FY 1983-1986 time frame.

   If funds permit, human factors engineering studies that will include task and system analysis dealing with control room design, training and staffing requirements, procedures development, and establishment of simulator needs will be initiated in FY 1985.

   In about FY 1986, after accident-sequence analyses, PRA, and code evaluations and development are well under way, studies will begin on potentials for low-probability accidents with the study of consequences possibly more severe than the design basis accidents. Low-probability accidents include core drop, control rod ejection, simultaneous moisture ingress with reactor depressurization, rapid depressurization of the PCRV, and unrestricted core heatup. Research supporting the study of these accidents is expected to be largely design specific. Initial efforts will use probabilistic methodology to assess which low-probability accidents, if any, should be considered in the design of mitigation systems. Studies of protective system instrumentation will be considered in this area.

5. Component Analysis. A detailed assessment will be made of thermal barrier and liner cooling system requirements in FY 1984-1985. This will encompass the heat loads under normal accident and degraded accident conditions,

the required performance of the materials, the materials' interfaces and penetrations, and the functional requirements of the system including redundancies.

A computer code system for analyzing the behavior of the containment system during depressurization events coupled with another process that produces combustible gases will be developed. A preliminary version is scheduled to be available in FY 1986. To the extent feasible, NRC will use verification data that may be obtained from DOE experiments.

Specific attention will be given to the evaluation of existing natural circulation codes for the analysis of the adequacy of cooling and coolant mixing under such conditions within the PCRV. Verification testing for the code(s) will be considered in FY 1986-1988.

5.2.5.2 Materials Interactions

1. Graphite. Graphite material characterizations concentrating on thermal oxidation of graphite, graphite oxidation profiles, mass transport in graphite, and mechanisms for strength losses in graphite will continue over the FY 1984-1988 period. In FY 1984-1985, consideration will be given to the potential for effects on core graphite due to possible concrete decomposition products. A significant effort will be undertaken in large-sample testing where the effects of geometry, flow rate, pressure, stress, and temperature will be examined. Oxidant levels simulating the ones experienced in Ft. St. Vrain will be applied to the large samples over long-term exposures.

   Studies of high-temperature mass transport of gases in HTGR graphites will address the determination of effective diffusion coefficients of ternary mixtures of gases (e.g., H_2, CO, H_2O) in several HTGR candidate graphites during FY 1983-1984.

   Development and modification of codes and standards will be undertaken for graphites as supporting data become available. Plans will be made in FY 1982-1983 to determine these data needs.

2. Metals. Particular consideration will be given to the adequacy of codes and standards for the thermal barrier and liner materials, thermal barrier and liner attachment characteristics, and interface effects with concrete beginning in the FY 1983 period. Consideration will also be given to development of any needed criteria for nil-ductility temperature (NDT) effects, considering the appropriate epithermal neutron flux. This work is coordinated with Section 5.2.5.4, "Structural Integrity."

3. Metals/Concrete Compatibility. Concerns of particular significance to the HTGR will be evaluated in FY 1983, and needs for program work to resolve issues will be developed for FY 1984 and beyond. Of concern are the liner/concrete and anchor interfaces and other primary system components since they may indicate some safety significance over the long term.

4. Containment Atmosphere Effects. The behavior of the fuel above its melting point and the potential for imposing substantial additional loads on the containment as a result of the production of combustible gases will be addressed beginning in FY 1986-1987. Applicability or modifications will be considered for Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." Studies of the chemical compound formations and the retention of released iodine and other key fission products in the PCRV and in containment during core heating will follow.

5.2.5.3 Source-Term Development

1. Fuel Integrity and Fission Product Release. Plans for future (larger program base) in-pile testing of fuel particle retention and release of fission products will be considered in FY 1984. Specific tests will be designed for verification of the data base and the performance of the HTGR fuel, and particularly the fuel particle coating integrity at high (accident-induced) temperatures, eventually to be run in a facility such as Sandia's ACRR. A series of four instrumented capsules is expected to provide sufficient information unless some significant and unexpected observation is made. This information could also be used in verifying some of the models in the LARC computer code. Additional data could be provided for NRC's release model (NUREG-0111). Plans will be prepared for the NRC evaluation and data base preparation of low-enrichment uranium (LEU) fuel anticipated for HTGR application. This may be followed by fission product transport studies with LEU fuel in FY 1986-1988 to reassess the data bases for releases, transport, plateout, and liftoff.

2. Fission Product Plateout and Liftoff. An understanding of fission product adsorption and desorption on primary- and secondary-system surfaces as a function of temperature, surface condition, etc., is essential to a realistic prediction of accident consequences. These phenomena are incorporated into liftoff and plateout models in the fission product code SUVIUS, and good data are needed for model development and verification. Design Basis Accident No. 2, "Rapid Depressurization," will be reviewed for how fission product liftoff is enhanced and how the consequences affect building habitability and offsite doses. Isothermal studies of the adsorption and desorption of cesium, iodine, and strontium on steels, Incoloy 800, and silica in helium and in helium mixed with water vapor will also be resumed in the FY 1984-1986 period.

3. Aerosols. Experiments from the FY 1983 period will be continued to investigate the interaction of volatile fission products with airborne particulates and to study the rate and extent of fission product adsorption on aerosol particles as a function of aerosol material and concentration. It is projected that the investigation of the high-temperature materials interactions associated with the sustained loss-of-forced-circulation accident will quantify the preliminary indications of graphite aerosol formation and the effectiveness of the graphite aerosols in serving as condensation centers for condensable fission products. Under these conditions, an investigation of aerosol properties to characterize agglomeration characteristics will be performed about FY 1985. Adaptation of the aerosol

transport analysis methods developed under the LMFBR program such as HAARM and QUICK would then be appropriate.

5.2.5.4 Structural Integrity

Topics to be considered during FY 1983-1986 for structural integrity research include PCRV structure, liner and thermal barrier, penetration closure design, and the liner coding systems. Design and selection criteria must be developed using the ASME Code (Section III, Division 1) if metal closures are to be considered. Code requirements and limitations will be considered as they apply to high-temperature metals that may be used within, or which serve as a portion of, the primary system boundary.

Evaluations will be made in FY 1984 of what further design-specific research is required to confirm the applicability of the existing seismic design computer codes and techniques. Analysis of the core response to seismic excitation is needed to establish conditions required to induce failure in prototypic fuel elements. Consideration will be given to continuation of small-scale seismic tests to improve and test core seismic analysis methods during FY 1985-1987.

Material property and strength data for different grades of graphite, particularly nuclear-grade graphites under HTGR operating conditions, will be addressed in FY 1983-1984. Testing efforts will be planned to measure irradiation-induced creep and dimensional changes in isotropic graphite in FY 1985.

American Society for Testing Materials (ASTM) code cases do not cover design specifications for nuclear graphite components. The NRC is planning efforts beginning in FY 1983 to allow the timely development of the appropriate ASTM data. Particular attention is needed for graphite core supports (floor and posts) and thermal stresses.

For concrete, it is planned to design and conduct a series of tests beginning in FY 1984, or later as funds allow, to confirm our ability to evaluate the effect of local anomalies in the concrete structures such as the PCRV and the containment. Tests are planned for the verification of the NONSAP-C methodology for treatment of posttensioning of concrete structures. A program will be developed for obtaining additional data for concrete under triaxial stress as required for verifying the three-dimensional concrete constitutive models. Tests will also be designed to establish heat capacity and offgassing and spalling rates. More detailed assessment will be made of concrete constitutive laws (including effects of moisture, temperature, stress) and the results will be incorporated in the concrete behavior models and applied to the analysis of PCRVs and containments. Standards, where appropriate, will reflect the findings.

Special emphasis will be given to the development of tasks to focus on thermal barrier and liner cooling systems and PCRV/liner penetration integrity and failure modes and consequences, as funds permit, in FY 1984-1986. Data will be developed as necessary for appropriate standards.

5.2.5.5 Full-Scale HTGR Research Effort

The plans described above will be augmented if an actual licensing application is received. Consideration will be given to programs for equipment qualification and fire protection, to long-term metals programs, to larger-scale testing programs for code validation, and to in-depth evaluations of specific safety issues pertaining to the reactor design and sites identified.

6. REACTOR AND FACILITY ENGINEERING

In accordance with the Commission's Policy and Planning Guidance, the reactor and facility engineering research program emphasizes support of the safety of operating reactors and other operating nuclear facilities with its first priority on light-water-reactor (LWR) safety. Although by far the majority of the research effort is directed at operating facilities, particularly LWRs, it is recognized that a small but significant effort must continue to be directed at research on new facilities. In this regard, it must be noted that many of the results obtained from research on new facilities can also be applied to operating plants. Similarly, much of the operating facility research applies to new facilities.

The need to support the safety of operating reactors and make tough regulatory decisions will become more demanding with time. As present operating plants become older and new plants come on line, the research to support regulatory concerns in the engineering area will become more crucial if proper safety decisions are to be made without imposing unnecessary burdens.

We anticipate that, as plants become older and more come on line, research to support the safety of these plants should be directed at understanding time-related effects such as aging and degradation, methods of examination and testing to determine the conditions of components, interpretation of results of these tests for appropriate action, and the need to reevaluate operating plants in view of changing requirements and new methods of analysis such as probabilistic risk assessment (PRA).

In accordance with the above, our program will include a comprehensive aging element to ensure proper scoping, organizing, and managing. This program will be directed at identifying those components of greater concern from a safety viewpoint as they age and what conditions are of most concern in terms of safety deterioration of components important to safety. We believe that such a comprehensive program is necessary to ensure that important safety components are not overlooked and that any conditions that could adversely affect these components significantly are not omitted. This scoping represents a necessary road map, although the major portion of research associated with aging, we believe, will be executed through individual component research because it is the combination of the component and the environment it sees that must be understood.

As the systematic aging program e olves to reduce the probability that we have overlooked important components of concern, we will undertake or continue to pursue programs associated with specific components that clearly will be of greater concern as plants become older. These include programs associated with the following components and phenomena:

1. Pressure vessels - Irradiation effects, including dosimetry measurements, development and validation of analytical techniques for predicting material behavior and condition, and evaluation of pressurized thermal shock. With the unique opportunity of having vessels aged in service upon which to perform research to validate conditions that were simulated in our analytical and experimental programs, our long-range plan includes using

materials from actual pressure vessels from decommissioned plants for research associated with irradiation effects.

2. Piping - Evaluation of piping that has been degraded in service to understand its margin-to-failure and techniques for predicting its condition. Investigation of specific phenomena such as intergranular stress-corrosion cracking is included.

3. Steam generators - Time-related deterioration principally due to corrosion.

4. Electrical equipment - Aging of different types of equipment, including its insulation subjected to normal and accident environments.

To complement the aging program, we anticipate a greater need as operating plants become older to have in hand satisfactory nondestructive examination methods and the capability to interpret these examinations so that proper action can be taken in a timely manner to support safety. To this end, our program is directed at developing reliable, reproducible nondestructive examination techniques, including their associated interpretation and analysis for decisionmaking, that could be used for examining steam generator tubes, piping, and vessels. In addition, we continue our effort to evaluate on-line detection methods such as acoustic emission which, if it can be demonstrated to be practical and applicable to a full-scale plant, would give us early detection of potential safety problems. Our program also anticipates that, because it is practical to apply several of these techniques only at specific periods such as shutdown, it is important that we understand the mechanism of deterioration, particularly the rate, so that proper actions can be taken to ensure that deterioration doesn't progress to an unacceptable level between inspections. Conversely, if the period between inspections is not fixed, this program will aid in identifying frequency of inspections.

Some of the corrective actions flowing from the aging and nondestructive examination research could result in maintenance. Thus, our program anticipates that there will be a greater need to look at repair welding, sleeving of steam generator tubes, plugging of tubes, testing of pumps and valves in service, and replacing of components.

Leaving the aging and detection problems, our program anticipates, and history supports, the fact that requirements continually change as the technology progresses. To this end, our program has elements addressed at anticipating reevaluation of operating plants to new criteria, newly postulated events, or new methods of evaluation. Our structural engineering program anticipates that containment and other safety structures of operating plants may have to be evaluated against criteria for which they were not designed and decisions regarding the safety adequacy made without imposing an unnecessary burden. The containment of an operating plant may be required to consider hydrogen or seismic loads for which it was not designed and the question raised: "Is the inherent margin satisfactory?" Similar questions may be raised with other structures regarding seismic or other phenomena such as tornadoes. We anticipate a growing need to address these issues from a strong base as the technology evolves and plants age. It is further recognized in our containment program that aging may play a part, and thus we anticipate a need to investigate the effects of aging on containment performance, particularly penetrations.

Our chemical control systems program has also been developed anticipating that there will be a need for decisionmaking resulting from changes in current knowledge or criteria and their application to operating plants designed on a different basis. We anticipate, for example, that fission product and hydrogen source terms will change in the future, and systems to cope with these new conditions will be proposed, thus requiring regulatory decisions on the acceptability of either systems in place or those newly proposed. In addition to aiding decisionmaking on operating plants, this information may also be useful in upgrading safety because, as an offshoot, insights will be gained into the various methods of control that could be fed back and could upgrade these systems as appropriate. In addition, we anticipate this information will be needed for rulemaking and standards development.

We anticipate that fire protection will continue to be an area of changing current requirements and the associated decisionmaking on the need for applying them to operating plants. Specifically, we expect that control room designs, particularly regarding their susceptibility to fire, will be raised as an issue, and our program reflects this.

Aging plants increase the probability that many will go out of service and be decommissioned. We anticipate that as plants begin the decommissioning process there will be unique opportunities to validate any analyses and procedures that have evolved or to develop rules, standards, or other guidance for decommissioning plants. Our long-range plan includes direct involvement in plants undergoing decommissioning.

More operating plants and longer periods of operation mean more spent fuel will be generated. Anticipating the need to store this increased volume of spent fuel and in anticipation of applications for dry spent fuel storage, which appears to be more flexible than wet storage, our long-range plan includes research in this area.

We anticipate that PRA will become a more useful tool in reactor safety. This is reflected in our long-range plan in several areas, including load combinations, seismic research, pressurized thermal shock research, and piping research.

By FY 1984, the Seismic Safety Margins Research Program (SSMRP) will have contributed the necessary seismic element to PRAs that had not previously been included, thus increasing the credibility of this important tool. The contribution of seismic risk is not precisely known but could be high. We anticipate that this important area will need validation, and our long-range program is scoped to address this.

As discussed initially, by far the majority of our effort is directed at operating plants; however, we do have programs that are fundamentally directed at new plants although, as we discussed, they also may have application to operating plants. In the engineering area, we anticipate that our primary contribution to new plants will be through the standardization effort.

Our program on qualification of equipment, both electrical and mechanical, will be a necessary element if standardization is to move forward. This program is primarily directed at validating or developing adequate procedures for the qualification of equipment. It will address such questions as what are the various conditions for which the equipment must be qualified, what test must be performed, in what sequence should the tests be performed, what conditions must be applied simultaneously, and what constitutes acceptance. Research is particularly needed to develop simulated conditions such as radiation source terms and to develop practical methods for establishing conditions such as aging. We anticipate one of the key items that must be addressed in standardization is replacement parts. Procedures to be developed for qualification would be invaluable. It should be noted that the qualification research program is also useful in evaluating operating plants when attempting to make decisions regarding (1) unqualified or questionable equipment now installed, (2) equipment presently installed that is qualified to requirements different from those called for by present criteria, and (3) requalification of installed equipment. Similar to what was discussed above on the pressure vessel research program and recognizing there is no substitute for components operated and aged in service, our qualification program will include pumps, valves, and electrical equipment from operating plants and decommissioned plants.

The second area of research that has direct application to new plants is that of development of standard methods of analysis. Again, in a well-founded standardization program, the need for standard methods for seismic evaluation, piping analysis, and structural analysis is evident. Questions are already being raised regarding whether or not the proper balance of safety has been achieved in piping systems considering seismic and other loads. We anticipate this problem will become more critical, and our research program is directed at researching stiff vs. flexible piping to ensure maximum overall safety. Through this mechanism, we can not only help standardize processes and increase safety for new plants, but we may also apply the feedback to operating plants. Programs to develop and benchmark analyses in these areas are included in the long-range plan in anticipation of new plant applications.

The last program applicable to new plants is directed at facilitating decommissioning. Our research program is directed at developing design conditions that will facilitate decommissioning and aid in standardization.

The Policy and Planning Guidance of the Commission states that ". . . the agency must be prepared to carry out its research mission with fewer resources." Anticipating that resources in the future will become more limited, the need to coordinate our programs with other Federal agencies, industry groups, and international groups to avoid duplication and address higher-priority items becomes more demanding. The long-range plan identifies the transactions now going on to make clear that these must be strengthened.

We plan to offer something we have or will do in exchange for something that someone else can contribute. In particular, we will concentrate on validating much of our analytical work with data accumulated by others in experiments or experience.

In summary, our primary direction is toward operating plants and specifically toward concerns regarding time-related deterioration, its detection, and its correction. In addition, our program is directed at the anticipated need for reevaluation of operating plants against newly identified conditions with potentially new methods. For new plants, our program is directed toward aiding standardization.

6.1 Plant Aging

6.1.1 Issue

As plants become older, it is reasonable to expect the progressive degradation of components due to the normal course of operations. It is also reasonable to assume that this degradation, or aging, will result in operating anomalies not explicitly foreseen by the designers. (An example of an identified anomaly is the "denting" of steam generator tubes.) At issue is the level of assurance that all aging phenomena having a potentially major impact on safety will be identified sufficiently in advance to allow timely preventive or corrective measures.

6.1.2 Research Program Objective

The comprehensive plant aging research program has as its objectives:

1. The development of an overall, comprehensive aging element to ensure proper scoping, organizing, and managing of the research program,

2. An initial determination of systems, components, and structures important to safety and their respective susceptibilities to aging environments and mechanisms,

3. The establishment of priorities and schedules relating to further aging research,

4. The identification of significant structure-component/environment aging mechanisms and the development of suitable corrective or preventive methodologies (e.g., inspection techniques, maintenance and test criteria, and replacement criteria), and

5. The coordination of all structural, mechanical, materials, and electrical follow-on research into specific areas (e.g., pressurized thermal shock, denting, accelerated aging methodology for qualifying electrical and mechanical components).

6.1.3 Relationship to Other Programs

As the comprehensive research program on plant aging, this program relates to all NRC and industry research programs in this area. Specifically, this program is intended to ensure that important safety structures and components are not overlooked and that any conditions that could adversely affect (age) these structures and components are not omitted from consideration.

Programs addressing specific components and materials that will be accounted for in the comprehensive aging program and are discussed later in this chapter are:

1. Evaluation of pressure vessel embrittlement,

2. Evaluation of intergranular stress-corrosion cracking in piping,

3. Corrosion in steam generators,

4. Aging degradation of cast stainless steel, and

5. Methods to simulate the aging of electric components on an accelerated basis for purposes of qualification testing.

Other components, structures, materials, and mechanisms will be added as the research progresses.

6.1.4 Background and Status

The comprehensive aging program is being initiated in FY 1982 as a scoping study. At present, we envision a systematic effort to:

1. Identify all electrical and mechanical equipment, structures, and materials important to safety that could be adversely affected by aging processes,

2. Identify, through the construction of a comprehensive matrix, all age-related mechanisms (e.g., various degradation-inducing environments) for each of the structures, components, and materials identified in (1) above,

3. Identify those structure-component/aging mechanism sets in order of potential impact on reactor safety, and

4. Recommend, based on (3) above, new or modified research programs and their respective priorities.

6.1.5 Research Program Plan

The research tasks currently in progress and to be undertaken in FY 1984-1988 are listed below. The tasks discussed in this section will be carried out as part of other work on materials, components, systems, and structures described in later sections of this chapter.

6.1.5.1 Scoping Study

The initial phase of the comprehensive aging program will be conducted during FY 1982-1983. This phase will be a scoping study to determine, on a systematic basis, which structures and components and their respective aging environments constitute significant potential risks to public safety. These structures and components will be ranked in order of potential risk, and additional research programs relating to these structures, components, and environments will be recommended on a priority basis. These new research programs should include consideration of testing techniques to determine the onset of aging, recommended maintenance procedures and schedules, and replacement schedules.

6.1.5.2 Coordination Efforts

Upon completing the scoping study, the comprehensive plant aging research program will be principally directed toward the coordination of all NRC aging research programs. This effort will include the continuing identification of potential new research areas, designation of research priorities, recommendations of funding levels, recommendations for assignments of specific research projects to the various branches within RES, surveys of aging research being conducted by domestic and foreign organizations, and coordination of the dissemination of research results.

We expect the coordination phase to begin in FY 1984 and to continue at a relatively constant level at least through FY 1988.

6.1.5.3 Pressure Vessels

Every program being carried out by NRC research dealing with reactor pressure vessel material has dealt, is dealing, and will continue to deal with the aspect of the performance of these materials under both normal operating and upset or accident conditions as a function of time, i.e., a study of aging effects. These programs deal with the determination of as-fabricated material and fracture toughness properties to establish baseline and initial material acceptance criteria; the effects of irradiation, particularly regarding the degradation of the baseline properties; the development of accurate dosimetry measurement and calculational procedures to establish precisely the amount of irradiation to which the pressure vessel steels have been and will be exposed; the development and experimental validation of analytical fracture mechanics procedures to allow the assessment of the margins of safety of pressure vessels as they age; and the development of standardized, universally accepted laboratory test procedures for determining the material and fracture toughness of new and aged pressure vessels from small-scale specimens. With several commercial reactors being taken out of service in the relatively near future, the opportunity exists to use materials from these inservice-aged vessels to validate the artificially aged materials that have formed the basis for all prior studies. Such action is therefore planned for implementation during the period FY 1984-1988.

6.1.5.4 Piping

Time-related effects of degradation on piping are largely concerned with corrosion and stress-corrosion cracking. Typically, these effects require a long time for incubation and can proceed in the cracking phase over a long period of time also. The materials and stress levels are, of course, selected so that they should easily survive the design life, but unexpected stress raisers (e.g., weld residual stresses), vibrations, thermal stresses arising from unexpected fluid stratification, and simply time-at-temperature can act and combine to reduce mechanical properties and create the conditions for crack initiation. An important part of the planned work is to determine the margin-to-failure of cracked and degraded piping and to develop methods for predicting the safe operating lifetime.

6.1.5.5 Steam Generators

Degradation of steam generators is caused by corrosion, stress-corrosion cracking, intergranular attack, denting, wastage, and thinning, all of which are long-term, time-related effects. The plan is that tube integrity must be determined both for leak and for burst, as caused by these degradation factors. The measured tube integrity must then be correlated back to the nondestructive examination signals so that accurate predictive methods can be established and validated, and tube inspection and tube plugging schemes can be established for use.

6.1.5.6 Cast Stainless Steel

Cast austenitic stainless steel is a metastable product so that, with sufficient time at temperature, the material can slowly transform to equilibrium metallurgical states that may have lower strength. From a physical metallurgical standpoint, it would not be expected that the cast austenite would degrade in LWR service lifetimes, but such an event appears to be occurring. Pump and valve bodies are the components most affected by this phenomenon. Centrifugally cast stainless steel piping may also suffer from this same phenomenon, but it has not been in service quite long enough to begin to show the symptoms. One problem that may occur with this piping is the more rapid development of leaks as a result of cracks following the boundaries of the large columnar grains that result from the unusual cooling process involved in the centrifugal cast process. The nature and kinetics of this aging process will be established, and criteria for establishing proposed fixes will be developed.

6.1.5.7 Electrical Equipment

Research will continue in the area of accelerated aging methodologies for different types of electrical equipment subjected to qualification tests in simulated normal and accident environments. The continuing refinement of these methodologies will lead to more valid qualification test results for "aged" components, thus ensuring to an even greater degree that no unforeseen anomalies due to aging will occur under actual design basis event conditions.

6.1.5.8 Additional Programs

Research evolving from the initial scoping study will be performed on additional structures, components, and materials in accordance with designated priorities.

6.2 Primary System Integrity

6.2.1 Issue

The primary system is the prime barrier of defense against accidental activity release during severe accidents; thus, the integrity of the pressure vessels, primary piping, and steam generators is of paramount importance. These components are particularly vulnerable because plants must operate in a hostile environment of elevated temperature, high pressure, nuclear radiation, and corrosive environment for 40 years with only a limited ability to check and inspect for embrittlement, degradation, cracks, leaks, etc. Furthermore, plants must be capable of withstanding overpressures and accident and seismic loadings despite service degradation. A review of the operating experience of the past several years shows plant shutdowns because of pipe cracks, leaks, and steam generator tube failures and construction delays because of improperly welded pipes. For operating plants, highly embrittled vessel welds and the need to analyze vessels for the potential consequences of pressurized thermal shocks from accident loadings have placed significant analysis workloads on NRC and the industry. Finally, the corrosion and cracking in some steam generator tubing has been so severe that a number of entire steam generators have been removed from service and replaced with new units. The primary system integrity research program addresses all these problems and has as its objective both the development of a reference foundation of knowledge upon which materials and engineering licensing decisions are made and the establishment of reference standard methods for analysis and evaluation that can influence new designs. The program is focused on operating plants in anticipation of needs for tough licensing decisions and improvements in pertinent pressure vessel codes; the program also looks to potential problems that could arise in the future such as irradiation, corrosion, and other aging mechanisms that continue to degrade components.

The primary system integrity (PSI) research program needs to continue as a viable, reactive resource capability that can rapidly put a major effort to work on pressing problems that arise. The program cannot always identify the specific problem that will require attention some 5 or 6 years hence; it can and must, therefore, maintain technical capabilities in the appropriate areas that will likely be at the focus of new problems that arise as a result of operating complex plants for long periods of time in a hostile environment.

6.2.2 Research Program Objective

1. To develop and experimentally validate 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; to critically examine present criteria for postulated pipe rupture; to develop mechanistic and probabilistic data bases on pipe fracture; and to evaluate the leak-before-break concept for nuclear piping systems;

2. To establish correlation of fracture toughness from small laboratory specimens to the material in pressure vessels and piping in both unirradiated and irradiated material conditions;

3. To develop methods for calculating, measuring, and predicting neutron flux and fluence in vessel surveillance capsules and in the vessel wall itself;

4. To determine changes in the fracture toughness and integrity of steels, welds, and components that result from radiation, thermal aging, corrosion, and environmental degradation and to evaluate means of mitigating this degradation, including annealing;

5. To establish the integrity of cracked and degraded steam generator tubing and supports for validation of inservice inspection techniques and regulatory interpretation of results;

6. To identify and study the environmental parameters that cause cracking in PWR and BWR piping systems and to study PWR and BWR water chemistry parameters that cause corrosion of LWR components, especially the effects caused by changes in operating and offnormal conditions; and

7. To evaluate repair welding in piping.

6.2.3 Relationship to Other Programs

The PSI research program provides a great deal of fracture mechanics expertise to other groups in NRC. These groups include the RES Division of Risk Analysis for their program on thermal shock, the RES Mechanical and Structural Engineering Branch (MSEB) for their writing of a guide on shipping casks, and NRR for their resolution of items such as pressurized thermal shock and ductile fracture toughness in vessels. The PSI leak-before-break project is closely coordinated with work under way in the MSEB on structural effects in piping.

Other PSI research programs under way in the U.S. are sponsored by the Electric Power Research Institute (EPRI) as well as by the four LWR vendors. Other efforts are being undertaken by foreign governments, especially in Germany, France, the U.K., Belgium, and Japan. The NRC and EPRI work are regularly coordinated, especially in elastic-plastic fracture mechanics, crack arrest, irradiation effects and annealing, neutron dosimetry, crack growth, and environmental effects on pipe cracking. RES maintains a continuing exchange of research information with European governments and research laboratories. We will continue our efforts to develop a satisfactory exchange with Japan. A strong European exchange exists with Germany and Belgium wherein RES has developed additional separate "agreements" for closely coupled research in fracture and structural mechanics, neutron dosimetry, and nondestructive examination (NDE). In the fracture and structures area, Germany has given emphasis to experimental programs, while the U.S. has emphasized theoretical and analytical work; these two programs have been carefully integrated with exceptionally good results. Results from mutual irradiation experiments are also shared. The HDR reactor test facility in Germany is being used as a validation tool for our analytical developments. The Belgians cooperate closely in the RES neutron dosimetry program, both experimentally and theoretically, and with exchange personnel; the U.S., on the other hand, relies heavily on the Belgians for certain precise measurements and for experimental facilities at the Mol laboratory. RES has established a steam generator group project at Richland, Washington, for NDE and tube-integrity-based safety criteria evaluation, in which several foreign governments have agreed to participate; these include France, Japan, and Italy, with the excellent possibility of Germany's joining later.

We continue to stay abreast of the PSI research programs that U.S. LWR vendors are conducting, and we keep them informed of our activities. This is primarily accomplished through the availability of reports and regulatory interfaces.

6.2.4 Background and Status

The main emphasis for fracture mechanics research during FY 1982 and FY 1983 is on the development of a unified fracture mechanics analysis method to permit the evaluation of all regions of fracture behavior so that flaw extension and arrest can be correctly assessed under the broadest range of postulated transients. During FY 1982, the Pressurized Thermal Shock Experimental (PTSE) facility will be constructed, and by the end of FY 1983 it will be used to conduct the first experiment to validate the methodology for analyzing vessel fracture or integrity under pressurized thermal shock conditions. This methodology and its extension to probabilistic analyses forms the current basis for licensing decisions on the ability of operating plants to survive significant overcooling and overpressure transients.

An irradiation program using two types of weld material in large (up to 4-inch width) specimens for verification and extension of the present codified data bases for K_{Ic} and K_{Ia} for nuclear vessel material toughness curves will be initiated in FY 1982.

During FY 1982, a study will be started involving irradiation of the stainless steel cladding typical of that found in present-day reactors in order to set the licensing basis for decisions on cracking in vessels under thermal shock. A benchmark experiment will be available in FY 1983 that will effectively calibrate the methodology used to predict the neutron dosimetry and embrittlement in the reactor pressure vessel wall, based on measurements from surveillance capsules. The methodology, using irradiated steel removed from an out-of-service reactor pressure vessel, will also be validated in FY 1983.

The mechanisms and probability of pipe failure program will be completed in FY 1982. It will use probabilistic methodology to develop the most probable location for pipe breaks in both primary and secondary systems of PWRs and BWRs. The degraded piping program will begin experimental demonstration of the capacity of cracked pipes to withstand postulated accident and transient loadings and evaluation of the validity of elastic-plastic fracture mechanics methods for predicting the loading capacity and failure mode of cracked pipes. Construction of a test facility will be completed in FY 1983 and testing will begin, it is hoped, with the cooperation of foreign research establishments. The leak-before-break program, to begin in FY 1982, will set out the mechanistic and probabilistic bases for decisions on generic and plant-specific acceptability of leak-before-break design, construction, and operating philosophy.

Research in steam generator integrity is focused on establishing the integrity of cracked and degraded steam generator tubing and supports and on establishing reliable, reproducible NDE methods for detection and characterization of the corresponding flaws and modes of degradation. The correlation of NDE information with tube integrity forms the basis for translation of the results into regulatory requirements. Following emplacement of a retired steam generator into the steam generator examination facility in FY 1982, nondestructive and destructive characterization of the primary and secondary side will begin. (Details of NDE aspects are described in Section 6.3.) During FY 1983, the first tubes will be tested to determine burst strength and leak rate, for initial correlation of NDE signals to tube integrity, and to validate the integrity modes previously developed. These data will be of significant benefit to the updating of regulatory guides on inservice inspection of PWR steam generator tubes and on tube-plugging criteria.

In FY 1983, research will be completed to develop models for the predictions of the stress-corrosion-cracking service life of steam generator tubing. These models will be subsequently validated by testing tubes removed from the degraded steam generator of known prior operating history.

Research will start in FY 1983 on the mechanisms and associated rates of corrosion in steam generators by secondary coolant and will include parametric studies using test sections exposed to simulated service conditions. The test sections will incorporate both sound and degraded tubes with actual support plates, tube sheets, crevices, geometrical discontinuities, cracks, and sludge piles. 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. Concurrently, similar measurements will be made in crevices, crack tips, and sludge piles. The research information developed will be used to establish time-related safety effects to aid in determining inservice-inspection frequency and to make licensing evaluations of water chemistry control criteria for operating reactor steam generators to minimize corrosion degradation.

Research in FY 1982 and FY 1983 in the area of environmentally assisted cracking in LWR systems will concentrate on intergranular stress-corrosion cracking (IGSCC) of BWR piping and validation of remedies proposed by industry. Full-scale pipe tests under transient conditions of loading and composition of the coolant simulating startup, operation, and shutdown will be initiated in FY 1983. By FY 1983, studies will be completed as a basis for licensing to assess the possibility that proposed corrective actions such as induction heating stress improvements (IHSIs) and corrosion-resistant cladding (CRC) may induce or accelerate IGSCC susceptibility through the aging phenomenon of low-temperature sensitization (LTS) in 304 SS (stainless steel). In addition, kinetic studies of LTS conducted through FY 1983 will result in a better understanding of the importance of LTS under long-term operating plant conditions. In FY 1982, analytical models to predict redistribution of stresses in weldments as a function of pipe size and cyclic loading history due to long-term aging will be developed and will be benchmarked against pipe test measurements in FY 1983. The models will then be used to assess the long-term stability of improved residual stress states, resulting from fixes such as IHSI, under thermal and stress loading histories of operating BWRs.

In FY 1982, a program will be started on developing criteria for acceptable welding and repair welding of austenitic stainless steel piping to preclude or mitigate sensitization as an initiator of stress-corrosion cracking.

In FY 1982, tests will be started to study the reduction in toughness of cast austenitic stainless steel as a result of long-term aging at reactor temperatures.

6.2.5 Research Program Plan

6.2.5.1 Pressure Vessel Fracture Mechanics

FY 1984 — Establish the existence and magnitude of threshold levels for crack growth under LWR conditions for use in reactor vessel integrity analysis. Complete the following: (1) the correlation between Charpy energy and upper-shelf fracture toughness to be used in evaluating reactor vessels with low upper-shelf toughness; (2) the formulation of unified fracture mechanics methodology, presenting an integrated approach to analyses of situations involving both elastic and elastic-plastic toughness domains as would be encountered during a pressurized overcooling accident scenario; (3) the Thermal Shock Experiment TSE-8, defining the effects of cladding on crack initiation, propagation, and arrest under thermal shock conditions, including cladding effects in the analysis of reactor vessels subject to pressurized thermal shock transients; and (4) the Pressurized Thermal Shock Experiment PTSE-2, showing effects of warm prestress and defining acceptable levels of repressurization following an overcooling accident scenario.

FY 1985 — Complete: (1) PTSE-3, demonstrating unified fracture mechanics approach to vessel head closure and bolt loading situation, and (2) unirradiated plate and present weld practice toughness data base for confirmation and amplification of K_{IR} curve.

FY 1986 — Complete revision of fatigue curves for ferritic steels in ASME Section III to reflect environmental effects; and revise Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels," to give guidance regarding criteria and procedures to be applied in evaluating various classes of overcooling scenarios and in maintaining adequate safety margins during postulated overcooling scenarios.

FY 1987 — Complete cyclic crack growth studies of various finite flaw shapes in simulated pressure vessel configurations and environments; and recommend final revisions to the cyclic crack growth rate curve in aqueous environments for inclusion in ASME Code Section XI.

FY 1988 — Complete: (1) unirradiated fracture toughness data base for new high-strength steels to be used in establishing criteria for maintaining adequate margins of toughness in critical reactor components, and (2) the development of data base on weld fabrication defect density for probabilistic analysis and reactor pressure vessel reliability analysis.

6.2.5.2 Irradiation Effects and Dosimetry

FY 1984 Develop toughness data and annealing response information on specific plants being reviewed under the Systematic Evaluation Program for licensing decisions; and complete documentation of surveillance dosimetry benchmarks for evaluating surveillance dosimetry results and calculating and predicting fluence levels in reactor pressure vessels.

FY 1985 Complete irradiation of and start testing of 4T-K_{Ic} specimens, evaluating appropriateness of RT_{NDT} and other parameters as normalization or transfer functions for irradiated fracture toughness; complete irradiation of crack arrest specimens; and complete preparation and gain consensus approval of standards and guides on the calculation, prediction, measurement, and correlation of surveillance dosimetry and embrittlement.

FY 1986 Complete testing of irradiated 4T-K_{Ic} specimens and crack arrest specimens, evaluating appropriateness of ASME Code toughness curves and irradiated toughness normalization factors; and continue developing plant-specific dosimetry data from reevaluating surveillance program results.

FY 1987 Provide recommendations for revision of K_{IR} curve for current and higher-strength materials, especially for shape of the curve based on both initiation and arrest data from irradiation studies.

FY 1988 Provide irradiation effects and annealing recovery data on a plant-specific basis for evaluation of toughness levels and margins of reactor pressure vessel integrity following annealing.

6.2.5.3 Piping Fracture Mechanics

FY 1984 Complete evaluation of Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and provide revised criteria for postulated pipe-break locations. Provide realistic criteria for postulating pipe-break locations to avoid overstiffening and reducing reliability of piping under normal operating conditions; initiate experimentation and/or analysis to assist in establishing leak-before-break criteria; complete extended fracture toughness data base for piping materials to be used in evaluating piping integrity and demonstrating that leak-before-break criteria are satisfied for specific piping materials; and begin cooperative international pipe tests.

FY 1985 — Complete degraded piping program pipe tests, providing experimental confirmation of loading capacity of degraded pipes and validation of elastic-plastic fracture mechanics techniques in large piping specimens; begin construction of test facility for scale testing of piping systems having flaws (entire piping systems will be modeled in scale to demonstrate effects of system compliances and loadings); and begin development of computerized piping material information system to assist in NRR licensing evaluations, in standards development, and in piping reliability analyses.

FY 1986 — Perform scale-model testing of piping systems with flaws; complete cooperative international pipe test (cooperative testing is being pursued in order to use existing large-scale test facilities in Europe and to save time and expense); complete leak-before-break program, establishing final leak-before-break criteria, and initiate rulemaking to incorporate leak-before-break concepts in 10 CFR; and complete development of information system on computerized piping materials.

FY 1987 — Complete scale-model testing of piping systems, and confirm analysis techniques in scale systems; and begin development of interactive piping code for piping design compliance.

FY 1988 — Use results of scale-model piping system tests to evaluate design and leak-before-break criteria.

6.2.5.4 Steam Generator and Environmentally Assisted Degradation

FY 1984 — Provide initial validation or recommended changes in proposed fixes for BWR and PWR pipe-cracking incidents in existing plants; perform validation of IGSCC prediction and repair weld criteria by test and analysis of full-scale piping systems; develop initial findings of toughness loss in cast stainless steels from components removed from service; determine, using a retired steam generator, patterns of degradation, wastage, and remaining integrity of steam generator tubing for correlation to NDE results and corresponding development of inservice inspection and tube-plugging criteria; and, for the corrosion by secondary coolant program, assemble test section and start study of material degradation caused by changes in secondary coolant composition ranging from normal operation to upset conditions, with emphasis on the effect of crevice-concentrated impurities.

FY 1985 — Develop recommendations for (1) avoiding pipe-crack incidents in newer plants under construction or in design and (2) preparing regulatory guides on repair welding and validating final criteria for acceptance of repairs in stainless steels;

assess significance of toughness loss in cast austenitic stainless steels for reactor service exposure; and, for the corrosion by secondary coolant program, establish effects of bulk coolant composition on crevice chemistries and potential for corrosion degradation caused by changes in bulk coolant from normal operation to upset conditions in test section experiments.

FY 1986 Propose practical limits for environmental variables to control pipe cracking in LWR systems, and complete criteria for evaluating proposed fixes to pipe-cracking incidents; establish the nature and kinetics of the aging process for cast austenitic stainless steel to develop criteria for evaluating proposed fixes; and, based on coolant composition conditions revealed from test section experiments, start corrosion testing of steam generator materials to establish time-related safety effects to aid in determining inservice-inspection frequency.

FY 1987 Provide initial validation results from long-term thermally aged cast austenitic stainless steel control test specimens, evaluating and validating, as practical, recommendations from vendors on prevention or control of such toughness degradation; and continue corrosion testing of steam generator materials under chemistry conditions established from test section experiments.

6.3 Nondestructive Examination

6.3.1 Issue

Nondestructive examination (NDE), which is a most important tool for regulatory decisionmaking, has one of the highest potentials for enhancing or ensuring reactor safety - if the inspection finds the flaws, which are the typical initiators of failures, in a timely manner. Detecting and accurately evaluating flaws is critical to safety because a flaw-free component can survive most accident loadings whereas a flawed component has reduced strength and thus is much more likely to leak or even break under sustained operation or accident loads. The research program is aimed at developing and improving various NDE procedures (both for continuous and periodic inspections) for fast, accurate, and reliable detection, characterization, and evaluation of flaws in nuclear plant components. The research must also establish the data base of inspection reliability using different techniques and instruments for upgrading the code and rules for inspections.

6.3.2 Research Program Objective

The overall objective of this program is to develop reliable, periodic, and continuous NDE techniques and associated analyses that can detect, characterize, and evaluate flaws so that proper regulatory decisions can be made. Specific objectives are:

1. To quantify and upgrade the reliability of current inservice inspection (ISI) techniques for primary system components;

2. To develop, evaluate, and validate advanced techniques for flaw detection and evaluation during ISI of primary system components and steam generator tubes; and

3. To develop and validate new techniques for the continuous on-line monitoring of crack initiation and growth in reactors during operation and for leak detection.

6.3.3 Relationship to Other Programs

The NDE programs in NRC are tied to other research efforts, including the heavy section steel technology (HSST) program for acoustic-emission (AE) crack monitoring and test vessels and the piping reliability program to provide data on distribution of flaws in components. The RES NDE programs are coordinated with NRR so that, as circumstances permit, experimental developments (such as improved multifrequency eddy current instruments) are taken into the field at NRR request to enhance ISI results.

The NRC NDE research program is rather modestly funded in comparison to those sponsored by EPRI and foreign governments. The program is making significant contributions to updating the ASME Code Section XI inspection requirements. The NRC and EPRI work are regularly coordinated. Consequently, although both programs emphasize ultrasonics, there is little overlap in the technical approach. NRC is working more closely at present with EPRI to develop and share new large-sized, complex test specimens for pressure vessel inspection. The NRC inspection reliability program is closely integrated with foreign programs (PISC-II and Japanese Pressure Vessel Research Committee (JPVRC) programs) as well as with the U.S. Pressure Vessel Research Committee (PVRC) program, on round-robin inspections of special flawed test blocks. The NRC reliability program provides data to the British on piping inspection reliability in exchange for British data on pressure vessel inspection reliability. The NRC acoustic-emission weld-flaw-detection monitor has been specified for the critical welding of a very thick vessel head-flange joint. The NRC RES program on continuous AE monitoring is a key element in a German test vessel fracture experiment. While we expect to help validate our method under realistic conditions, the Germans hope to use our results to referee their own results.

We continue to stay abreast of the NDE programs that U.S. LWR vendors are conducting and to keep them informed of our activities. This is accomplished primarily through reports and regulatory interfaces. DOE programs in ultrasonic inspection of austenitic stainless steel piping are closely monitored because of the similarity in materials and type of inspection problem.

6.3.4 Background and Status

ISI research in FY 1982 will use round-robin test results to (1) establish the reliability of current ISI for piping systems and to recommend the Code changes to bring about required improvements, (2) provide experimental evidence for development of technical positions for a regulatory guide on ISI of stainless steel piping, and (3) validate statistical models for prediction of inspection reliability (probability of detection) as a function of inspection and component parameters. The available inspection techniques for detection and sizing for near-surface (underclad) cracks in pressure vessels will be evaluated during FY 1982 and FY 1983. The synthetic aperture focusing technique for ultrasonic testing (SAFT-UT), which provides highly accurate flaw characterization, will be validated in FY 1982 by actual imaging of flaws in reactor components and will be proposed for Code acceptance for flaw characterization in FY 1983.

There are shortcomings in the methods of inspection and in the inspection plan currently used for steam generator ISI. Defects in certain regions cannot be reliably detected or characterized and the ISI sampling plan does not ensure that primary system integrity can be maintained under subsequent normal operating or accident conditions. To improve detectability, interpretation, and characterization of defects in steam generator tubes and components, a number of new techniques have recently been developed or are currently under development. Research activities directed at validating these new NDE techniques for detecting and characterizing degradation in steam generator tubes and components will get under way in FY 1982, using a removed-from-service steam generator to perform in situ inspections of actual defected tubing. The inspection results will later be validated and correlated, starting in FY 1983, by selective destructive examination of defected tubes removed from the generator. The detection reliability, measurement error, and confidence limits will be established for the various techniques. A primary thrust of this program is to develop and validate reliable NDE techniques and procedures whose capability for evaluation of flaws is well characterized. Thus, the flaw information data derived from an ISI can be used with confidence to calculate the remaining strength of defected tubes using predictive models developed earlier in this program. This information is then used to formulate tube-plugging criteria to ensure safety. The detection reliability, measurement error, and confidence limits will also be established for the various techniques beginning in FY 1983. Further, beginning in late FY 1982, the generator will be thoroughly characterized on the primary and secondary side, using nondestructive (including visual) and destructive means, with respect to size, type, location, and distribution of defects. Some tubes may be sleeved so that the resulting NDE signals can be identified and catalogued for future reference. By combining the information on the NDE inspection error and reliability plus a statistically valid description of flaw type and distribution from this program with results from other research programs on stress corrosion and degradation rates, statistically valid ISI sampling plans will be developed to ensure continued safe operation of steam generators. Finally, a mockup will be assembled using tubing and components removed from the generator. This will provide a future realistic reference test bed for demonstrating the capabilities of new instruments and techniques, as well as for qualification of personnel and techniques to ensure that they can reliably inspect steam generator tubes and components.

In the continuous on-line monitoring program, a fatigue and burst test of a vessel (approximately 5-in thick by 5-ft diameter by 10-ft long) containing several cracks and defects will be conducted under simulated reactor operating conditions during FY 1982. The vessel will be monitored by AE to evaluate the crack identification and severity models developed earlier. Following the AE monitoring of the hot functional testing in FY 1982 of a new PWR, the AE method and systems developed will be applied to continuously monitor the operating reactor in FY 1983. Parametric studies will start in FY 1982 and continue through FY 1983 to develop a methodology for on-line leak monitoring using AE.

6.3.5 Research Program Plan

FY 1984 — Develop and evaluate flaw-detection probability and fracture mechanics models for reactor component integrity assessment; evaluate improved SAFT-UT near-real-time flaw detection, characterization, and display using a field-implementable system on specimens containing real flaws; determine whether AE can be used for inservice hydrotests (if feasible, establish criteria and Code/regulatory acceptance); complete new transducer probes for detecting stress-corrosion cracks and discriminating from geometrical reflectors, and establish Code acceptance; make final determination on electromagnetic acoustic methods for through-weld and cast stainless steel inspection, and establish Code acceptance; determine inspection reliability of steam generator tubing using current and advanced eddy current and other techniques through use of the removed-from-service steam generator; gain Code acceptance of multifrequency eddy current testing as developed by the Oak Ridge National Laboratory (ORNL) for steam generator tubing; and recommend final criteria and methodology for on-line monitoring and flaw evaluation using AE.

FY 1985 — Recommend a unified set of inspection requirements using NDE flaw-detection reliability and sensitivity, component material properties, and service conditions to ensure a suitably low failure probability; perform ISI reactor testing using the near-real-time SAFT-UT system to provide validation and final optimization of the methods and techniques; develop steam generator ISI plan and frequency based on eddy current inspection error and reliability and actual flaw distribution using the results from the removed-from-service steam generator; gain Code acceptance for on-line AE monitoring and flaw evaluation; and validate AE leak-monitoring and evaluation methodology by testing on reactor.

FY 1986 — Gain Code acceptance of the (1) unified set of inspection requirements for ISI of primary system components and (2) automated near-real-time SAFT-UT flaw detection, characterization, and display techniques for ISI; validate steam generator ISI plan; evaluate NDE techniques for preservice and fabrication inspection; and gain Code acceptance of on-line AE leak-monitoring and evaluation.

FY 1987 — Improve or adapt new techniques for preservice and fabrication inspection; and evaluate new techniques needed for ISI.

FY 1988 Gain Code acceptance of new techniques for preservice and fabrication inspection; and validate new techniques needed for ISI by actual testing on reactor.

6.4 Mechanical Systems and Components

6.4.1 Issue

Design criteria and input loads are changed over the years to reflect the latest technology and new information. Operating plants must be evaluated to determine if their inherent design margins are sufficient to accommodate the improved criteria or increased loads while still meeting the minimum safety requirements. Additionally, criteria must be developed or improved to provide a balance of safety requirements for new standardized plants.

The current regulatory design criteria that impose the assumption of double-ended pipe breaks and the simultaneous application of seismic loads associated with the safe shutdown earthquake (SSE) have led to massive pipe and vessel restraints to resist the very large resulting loads. This apparent conservatism may, in fact, actually reduce safety.

Similarly, design practice to resist seismic loadings leads to numerous and closely spaced static supports and dynamic restraints (snubbers). These stiff designs may, in some cases, increase nozzle loads on components attached to the piping. Moreover, snubbers may fail during normal operation and may impose additional, unaccounted-for loads on the piping. Thus, the apparent conservatisms associated with stiff seismic design may have an adverse effect on the overall safety of the plant. In order to provide improved criteria for future standardized plants as well as operating plants, there is a need to evaluate the current licensing criteria dealing with pipe break, load combinations, pipe-to-pipe impact, damping values for piping, standards for fatigue evaluations of Class 2 and 3 piping, requalification of mechanical equipment after accidents and events, and validation of analytical techniques used to evaluate mechanical components. These evaluations must also include the effects of aging (including wear), maintenance, inspection, and design and construction errors. We will then be in a position to determine and evaluate the behavior of components and systems under operating, extreme environmental, and accident conditions in terms of risk reduction and to provide a basis for balanced safety requirements both for new plants and when retrofitting older plants.

The NRC must be assured that certain aged or degraded equipment remain functional during and after an accident or environmental event to safely shut down a plant and provide adequate decay heat removal. Thus it is important that adequate equipment qualification requirements and procedures be specified for operating plants and for new standardized plants. In view of changes over the years in equipment qualification requirements and review procedures, the reliability of installed equipment, especially in older operating plants, will be subject to question. There is a need to determine adequate qualification procedures and requirements to reduce risks to the public and to determine how these requirements should be applied to operating plants, new plants, and plants in the licensing process.

Another issue focuses on the potential for additional failure modes in recently proposed designs of shipping containers. The proposed use of heavy section steel forgings and castings for shipping container applications requires consideration of the potential for nonductile failure of such containers. This was not considered a significant failure mode in previous cask designs that used separate material for shielding.

6.4.2 Research Program Objective

The research program aims at generating data that will permit an evaluation of the integrity of mechanical components and systems in LWRs under anticipated operational, environmental, and postulated accident conditions. The main regulatory objective is to provide the NRC licensing staff with the ability to assess functionality and structural integrity of degraded or aged components in terms of margins of safety and failure probabilities. Associated with this regulatory objective is the need to generate a basis for decisions regarding placement and reliability of piping supports, snubbers, and pipe whip restraints, particularly as they affect the stiffness or flexibility of piping and nozzle loadings related to such stiffness and flexibility. In addition, the techniques used to estimate the dynamic response of pipes, vessels, and components will be subjected to validation and confirmation as service and test data become available.

Moreover, the program is designed to develop information that will, in conjunction with other research programs, permit a determination to be made concerning acceptable methodology for qualification of mechanical equipment, estimates of the potential for reduction of risks for the public, and the value/impact of a mechanical equipment qualification program. The main regulatory objective of this portion of the program is to provide a basis for regulations, guidance, and licensing decisions in this area. An additional objective is to develop an acceptable qualification methodology for electrical equipment when such equipment is subjected to a seismic event.

The shipping container structural integrity program aims to provide the licensing staff with tools for assessing shipping containers constructed of heavy sections such as nodular cast iron and ferritic steel forgings.

6.4.3 Relationship to Other Programs

In the area of piping, KfK is continuing their investigations of piping behavior under simulated seismic and thermal-hydraulic transients as part of a collaborative program, and future cooperation is planned. EPRI has an extensive piping program involving both in situ and laboratory testing with which the NRC has established cooperative agreements. The nature of this cooperation consists at this time of EPRI transfer of test data to NRC contractors for use in computer validations. The Japanese Seismic Damping Ratio Evaluation Program and Large-Scale Seismic Test Facility are test efforts involving sums in excess of $250 million. Half-scale testing of BWR main steamlines and of PWR primary system piping and reactor vessels is scheduled to be undertaken there. However, exchange agreements with the Japanese have not been negotiated. Efforts to achieve an agreement are continuing.

The effort on determining the vibratory characteristics of reactor internals is part of the NRC surveillance and diagnostics program. The results of this effort will be used for evaluating noise spectra for determining abnormal behavior of reactor internals.

In the area of equipment qualification, the Seismic Safety Margin Research Program (SSMRP) is developing "fragility" curves for specific components that relate the probability of component failure in a specific manner when these components are subjected to varying levels of seismic challenge. The methods used in the SSMRP for development of seismic fragility curves may be used as prototypes for developing fragility curves related to the other conditions that may require qualification of mechanical equipment, e.g., chemical conditions existing outside a mechanical component during a loss-of-coolant accident. Conversely, data from the mechanical component integrity and equipment qualification program will be used to help validate the methodology and improve the data base for the SSMRP. Cooperation, in the form of planning and information exchange, is taking place with EPRI in the area of equipment qualification.

Research pertaining to the structural integrity of shipping containers will encompass areas of concern in using these containers as dry spent fuel storage containers. In addition, this research is related to the modal studies program sponsored by the Division of Risk Analysis. The results of these studies will be used for determining whether the current hypothetical accident conditions are appropriate for container design.

These relationships are summarized in Table 6-1.

6.4.4 Background and Status

RES efforts on load combinations have investigated the probability of pipe rupture in a PWR primary system and have concluded that guillotine pipe rupture, with or without the presence of earthquakes, is extremely unlikely. This research has depended on probabilistic fracture mechanics and fault tree/event tree analysis. Studies of safety relief valve (SRV) discharge loads and their impact on piping and equipment have been undertaken at the Kuosheng Power Plant in Taiwan. We plan to validate our computer simulations using data obtained from these tests. Studies are under way to make recommendations to Section III and Section XI of the ASME Boiler and Pressure Vessel Code, especially in the area of dynamic allowable stresses, piping support reactions, fatigue evaluations for Class 2 and piping components, valve leakage and testing requirements, examination of supports, use of fatigue curves in determining acceptance standards for inservice inspection, and determining the effect that inservice inspection has on the overall safety and reliability of the nuclear power plant. Our pipe-to-pipe impact test and analysis program has started. The test matrix includes variations in support conditions, impact velocities, diameters of impacting and target pipes and wall thicknesses so that sufficient data will be available for confirming our mathematical models of the impact event. A new computer code, WIPS, developed under NRC funding, supports this effort. Work completed at the HDR in Kahl, West Germany, has dealt with the simulated seismic response of recirculation-loop piping using explosives buried in the soil and with response of piping to thermal-hydraulic transients caused by

TABLE 6-1

RELATIONSHIP OF MECHANICAL SYSTEMS AND COMPONENTS PROGRAM TO OTHER PROGRAMS

Program Area	NRC-Sponsored Research (Performing Organization)	Other U.S. Research (Sponsoring Organization)	Foreign Research (Country)
Mechanical Component Integrity	Load Comb. Program (LLNL)	Piping & Fitting Reliability (EPRI)	GRS/TUM Load Comb. (FRG)
	Piping Computer Code Verif. (INEL, BNL, ANCO)	Seismic Piping Test & Anal. (EPRI)	Seismic Damping Ratio (Japan)
		DOE Piping Studies (HNL, WARD)	HDR Facility Tests (FRG)
			SRV Discharge Tests (Taiwan)
	SSMRP, Fragility and Methodology (LLNL)		
Mechanical Equipment Qualification	Seismic Qual. Program (SWRI)	Aging/Seismic Qual. Program (EPRI)	
	Mechanical Qual. Program (LLNL et al.)	Nonnuclear Equip. Program (EPRI)	
	Equip. Qual. Program Plan (NRR)	Safety/Relief Valve Test (EPRI)	
	TAP A-46 (NRR)		
	SSMRP-Fragility (LLNL)		
Shipping Container Integrity	Dry Spent Fuel Storage Program (HEDL)	Materials research on ductile cast iron (SNL)	Container Test and Evaluation Program (FRG)
	Modal Studies Program (Ridihalgh, Eggers and Assoc.)		

rupture discs and feedwater check valve closure. Testing funded by the NRC at the HDR also involves steel containment, flood water storage tanks, and piping. The aim of these tests is to learn how eigenparameters, especially damping, vary with level and type of excitation and to verify analytical modeling procedures used in safety assessments. Our efforts on benchmarking computer codes have led to a nonproprietary piping code called PSAFE2. This code is being modified to handle multiple independent support motions. Studies of stiff vs. flexible piping will start in FY 1982 and continue through FY 1984. These studies examine risk-reduction potential by removing some supports and snubbers used in piping systems.

Work is continuing, in cooperation with the Division of Facility Operations, to characterize the dynamic response of reactor internals (see Section 7.2). This work will support the use of neutron noise data for the detection of structural problems through instrument tube vibration and core barrel motion.

Research is currently under way to evaluate the dynamic (including seismic) qualification criteria used for mechanical and electrical equipment. This includes examination of input forms and comparisons of older and current criteria. We are also monitoring and evaluating the industry program to demonstrate the ability of safety and relief valves to operate under all postulated accident conditions. There is also a research program to evaluate existing design application and test criteria for the use of snubbers. This program will end in FY 1983.

Research efforts in the area of shipping container integrity will result in the development of a regulatory guide dealing with fracture toughness criteria for steel and nodular cast iron containers. In addition, research has been completed in the area of those normal transport loads such as rail car humping operations. Although a regulatory guide has not been developed in this latter area, these results will subsequently be used as part of the basis for a regulatory guide on normal transport loads.

6.4.5 Research Program Plan

6.4.5.1 Mechanical Component Integrity

With regard to structural integrity of mechanical components, the load combination research program will be expanded beyond our present considerations of a Westinghouse PWR to include both Combustion Engineering (CE) and Babcock and Wilcox (B&W) PWRs as well as a General Electric (GE) BWR. The GE BWR will require the introduction of stress-corrosion cracking into the probabilistic assessment of rupture of the reactor coolant pressure boundary piping. Additional work on indirectly induced double-guillotine rupture of primary system piping by earthquakes will be undertaken. In addition, studies of the influence of gross design, construction, maintenance, and installation errors will be conducted to learn how such errors impact risk assessments for pipe breaks. These studies will provide information for revising Regulatory Guides 1.46, "Protection Against Pipe Whip Inside Containment," and 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." The pipe-to-pipe impact tests, in conjunction with use of the WIPS code, will permit an evaluation and possible revision of Standard Review Plan (SRP) Section 3.6.2, which deals with licensing acceptance criteria for pipe-to-pipe impact. Studies of stiff vs. flexible piping will continue. The effects of current design practice and the potential risk reduction by making the piping more flexible through removal of snubbers and higher damping and allowable stress values will be examined.

ASME-Code-related activities will lead to new recommendations for allowable stress criteria under dynamic environments, identify sources of uncertainties associated with piping support restraint loads, and describe the basis for Code Class 2 and 3 fatigue evaluations and correlate the methods with Class 1 methods. Other studies will assess Code rules for the control of thermal ratchetting of Class 1 piping and will recommend new valve leak rate criteria and testing procedures and new sampling criteria for the inspection of supports. Additionally, efforts will be undertaken to assess Code inservice acceptance standards and to assess the impact that increasing inservice inspection has on nuclear power plant safety and reliability.

Research on piping computer code benchmarks, together with HDR piping research, will provide regulatory standards to ensure adequate models of piping under various dynamic transients and will indicate the accuracy that may be expected from such models. This effort will help fulfill the requirements of Section III of Appendix B to 10 CFR Part 50 for confirmation of the adequacy of computer codes. We will also develop recommendations for confirmatory impedance tests as well as flow transient and structural response monitoring for essential piping.

Moreover, steel containment testing at the HDR will allow an assessment of how well such structures may be modeled in practice. In cooperation with the PVRC and its steering committee on "Design of Mechanical Systems for Dynamic Loadings; Procedures and Allowable Limits," we will conduct a reliability-oriented assessment on how conservatisms in seismic piping design and analysis affect overall piping safety. The assessment will address the question of how NRC conservatism on allowable damping, linear dynamic analysis, and service level assignments for loads and load combinations increase seismic reliability and will compare it with the decreased reliability resulting from increased thermal loads, increased installation loads due to misalignment, and decreased effectiveness of inservice inspection. As a part of this effort, we will include aging effects on the reliability of piping, pump seals, and bearings. We expect to initiate a study on postevent reevaluation and requalification with particular attention to requalifying mechanical components subsequent to earthquakes that exceed the operating basis earthquake (OBE). The objective of this investigation would be to delineate the nature and extent of postevent inspections that are required to certify that a plant is safe to operate after the event has occurred.

A program will be started to investigate and evaluate the behavior of reactor vessel internals in response to dynamic loads from earthquakes and accident conditions, including thermal shock. Results from blowdown tests at the HDR in Kahl, West Germany, will provide data on the dynamic behavior of a core barrel that will permit the verification of analytic techniques used to predict the structural behavior of core barrels used in the U.S. In addition, a program will be completed that identifies the vibration characteristics of reactor internals for use in the surveillance and diagnostics program.

A year-by-year description of planned accomplishments with regard to structural integrity of mechanical components follows. The relationship between this work and licensing is shown in Figure 6.1.

FY 1984 Complete probabilistic assessment of CE and B&W primary system piping; make recommendations for allowable stress criteria under dynamic environments to Section III of ASME Code; validate WIPS code by testing; complete survey of existing vibratory data typical of LWR internals, and supply information to the noise surveillance program; and complete research to evaluate stiff vs. flexible piping.

FY 1985 Complete probabilistic assessment of GE main steamlines and recirculation-loop piping; propose revision to SRP Section 3.6.2 on pipe-to-pipe impact criteria; establish criteria for computer modeling of piping when subjected to seismic and thermal-hydraulic transients, and develop a regulatory guide addressing this issue; complete evaluation of tests and analyses of reactor vessel internals; and provide information on pipe damping to revise Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

FY 1986 Estimate impact of design and construction errors on piping safety assessments for Westinghouse, GE, B&W, and CE NSSS, and begin revisions of Regulatory Guides 1.46 and 1.48; propose schedule for inspection subsequent to OBE for requalification of mechanical equipment; and complete benchmarks for multiple support inputs.

FY 1987 Recommend changes to NRC seismic criteria for piping based on overall safety assessment of stiff vs. flexible piping; complete piping benchmarks for nonlinear analysis; develop standards for modeling steel containments subjected to dynamic environments based on HDR tests; and recommend impedance test procedures and flow transient and structural response monitoring requirements for essential piping.

FY 1988 Evaluate small-break LOCA probabilities. Establish standards for evaluating, replacing, and maintaining aged components on piping.

6.4.5.2 Mechanical Equipment Qualification

The mechanical equipment qualification program consists of several projects focused on validating or developing adequate procedures for the dynamic (including seismic) and environmental qualification of mechanical equipment and the dynamic (including seismic) qualification of electrical equipment.* The

* Environmental qualification of electrical equipment is being carried out under a separate coordinated effort. (See Section 6.7.)

THE RELATIONSHIP BETWEEN LICENSING AND RESEARCH FOR MECHANICAL SYSTEMS AND COMPONENTS

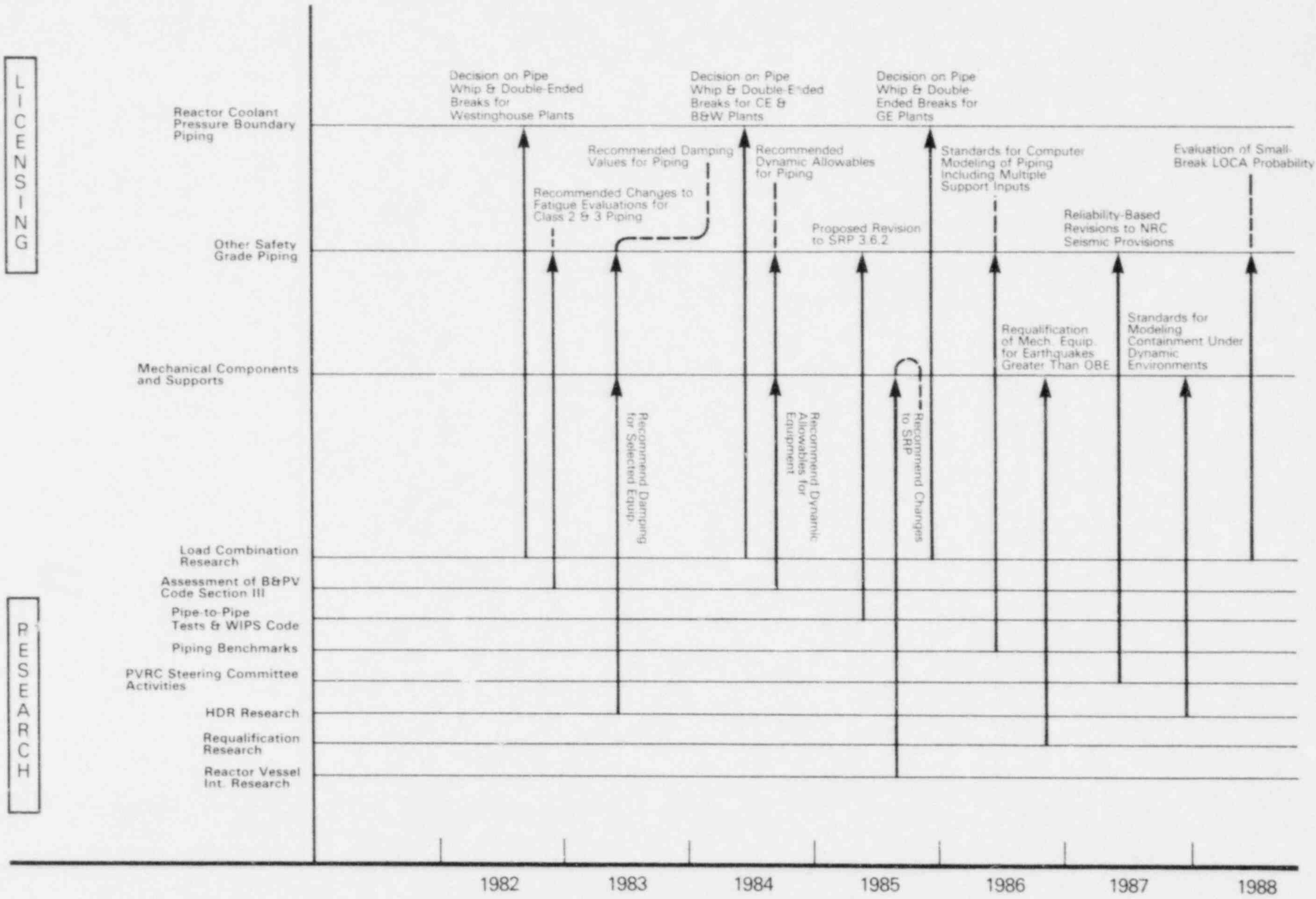

Figure 6.1

following research efforts will be carried out within this program. The relationship between these efforts and licensing is shown in Figure 6.2.

1. Perform sensitivity studies to identify specific equipment that should be subjected to qualification requirements, and define the range of conditions for which the equipment must be qualified. The identification of equipment and conditions will include consideration of potential contribution to risk.

2. Study the effects of various inputs to determine which wave forms are acceptable for simulation of earthquake excitation.

3. Evaluate the influence and importance of component aging (including wear) and environmental degradation effects on the dynamic qualification of equipment.

4. Evaluate the importance of sequential testing (including preaging) and combinations of loadings and environments.

5. Perform fragility tests to identify failure modes and failure levels on selected critical equipment identified by the SSMRP.

6. Determine scale-modeling guidelines for the dynamic testing of equipment.

7. Identify all the vibrations and accident-induced dynamic loads that may have an effect on the functional capability of equipment important to safety.

8. Assess the reliability and uncertainty of dynamic qualification methods.

9. Evaluate pump operability assurance programs currently being conducted by industry.

10. Evaluate valve operability assurance programs currently being conducted by industry.

11. Formulate methods for requalification of existing operating plant equipment.

A year-by-year description of planned activities in mechanical equipment qualification follows:

FY 1984	Identify the range of conditions to be evaluated.
FY 1985	Perform component sensitivity studies.
FY 1986	Evaluate the influence of component aging on qualification.
FY 1987	Evaluate the effect of environmental conditions on qualification; and determine parameters that relate the testing results from a component of one size to a component of a different size.

THE RELATIONSHIP BETWEEN LICENSING AND RESEARCH FOR MECHANICAL EQUIPMENT QUALIFICATION

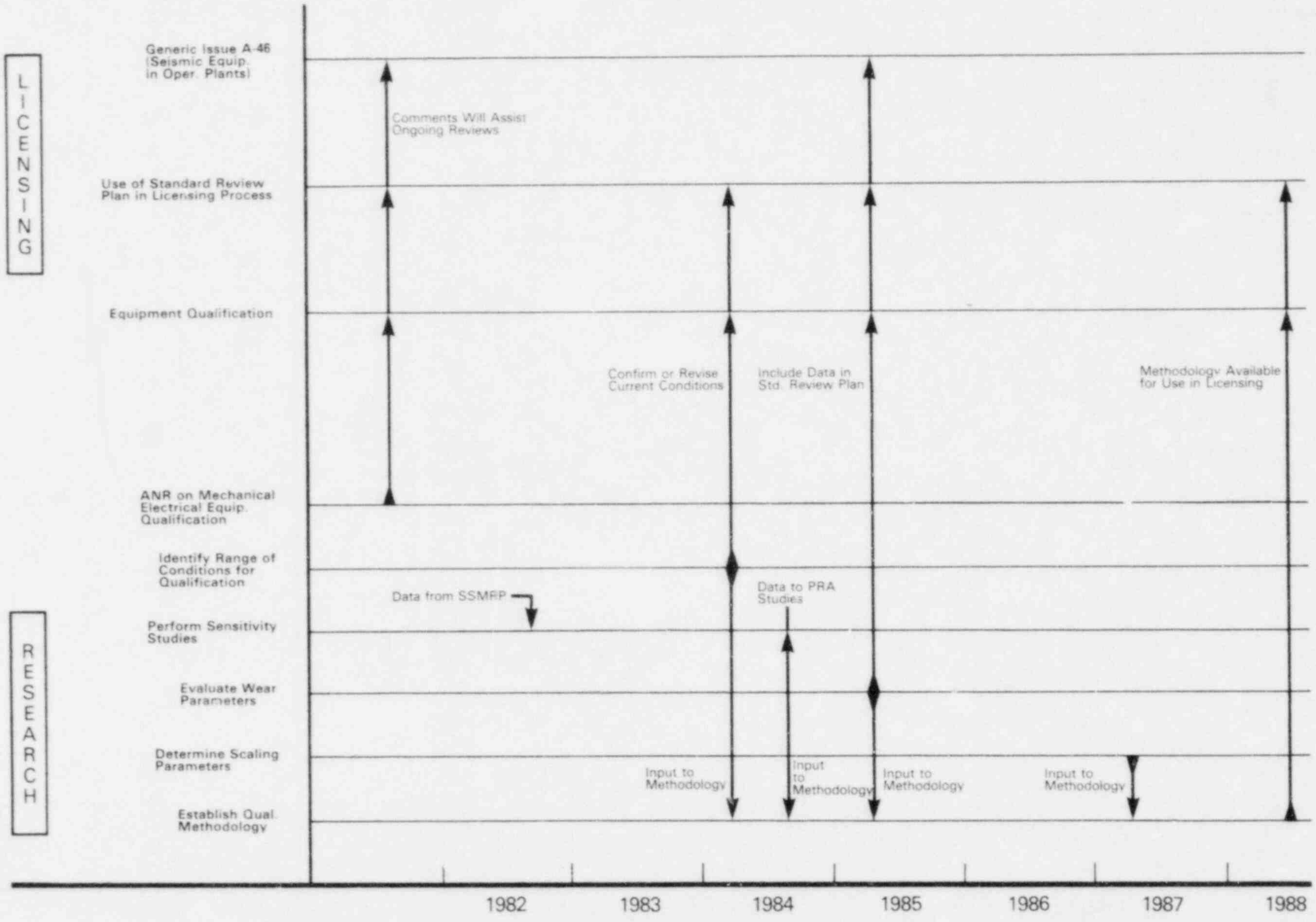

Figure 6.2

FY 1988	Complete development of methodology for determining how qualification affects safety to provide basis for backfitting requirements for operating plants.

In FY 1984, the rulemaking related to mechanical equipment qualification will become effective after the publication of an advanced notice of rulemaking in FY 1982 to secure comments from the public and the affected industry. As the program progresses, detailed guidance for qualification of mechanical equipment will be developed and issued.

6.4.5.3 Shipping Container Integrity

The shipping container integrity research program will provide the bases for regulatory guides in the areas of design, construction, and testing. The resulting guides will provide criteria for assessing fabrication processes typical of container construction, criteria and methods for qualifying materials for fracture critical components, and methods acceptable to the NRC staff for designing against hypothetical accident conditions. Research programs related to the structural integrity of dry spent fuel storage containers will be coordinated with the research areas for shipping containers especially where these containers may be used for both storage and shipping. Programs related specifically to dry spent fuel storage containers will be established if needs are identified during the research activity associated with shipping containers.

A year-by-year description of planned accomplishments for the shipping container integrity research program follows. The relationship between this work and licensing is shown in Figure 6.3.

FY 1984	Complete assessments of existing techniques for scaling flaws, machining flaws, and evaluating the results following a drop test; make recommendations for criteria to be used for evaluating shipping containers; and issue a regulatory guide on this topic.
FY 1985	Complete assessments of puncture analyses, and make recommendations on acceptable methods for three categories of shipping containers; and issue a regulatory guide on this topic.
FY 1986	Complete assessments of drop analyses, and make recommendations on acceptable methods for three categories of shipping containers; and issue a regulatory guide on this topic.
FY 1987	Complete assessments of thermal analyses, and make recommendations of acceptable methods for three categories of shipping containers; and issue a regulatory guide on this topic.
FY 1988	Complete evaluation of existing literature regarding tie-down and lifting device loads on containers, and make recommendations for acceptable loads; and issue a regulatory guide on this topic.

THE RELATIONSHIP BETWEEN LICENSING AND STRUCTURAL INTEGRITY RESEARCH FOR SHIPPING CONTAINERS

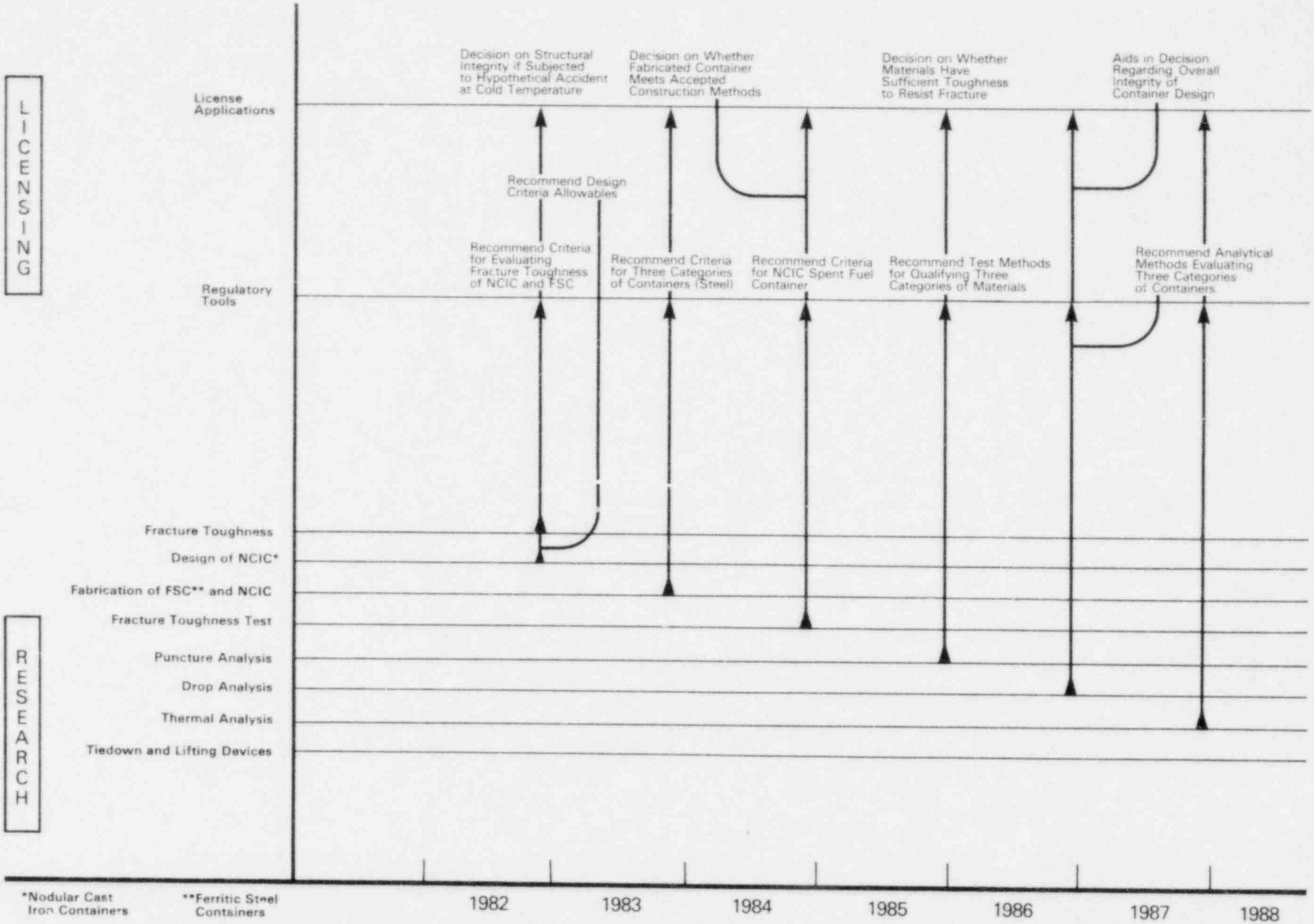

Figure 6.3

6.5 Seismic Design

6.5.1 Issue

The NRC must be prepared to provide the basis for licensing decisions involving operating plants that are required to consider changing seismic loads and design criteria. By knowing and understanding the inherent conservatisms in the seismic design (i.e., being able to more accurately characterize the realistic behavior of structures and components under earthquake conditions), the NRC will be better able to judge the necessity and extent of modifying and requalifying structures and components in older operating plants to be reviewed for increased seismic loads or of improving design criteria for new standardized plants.

Recent PRA studies have indicated that the seismic risk may be a large contributor to the total risk for nuclear power plants. Most PRAs prepared to date do not include an assessment of risk from earthquakes. Thus it is important that the NRC have methods to quantify and assess seismic risk to evaluate and enhance the credibility of PRAs. There is concern that the current licensing criteria may produce seismic designs that are apparently conservative for some features but can have adverse effects on the overall plant safety. For example, piping made stiff to resist seismic loads may cause higher thermal expansion stresses in nozzles during normal operation. What are believed to be conservatisms are added at each stage of seismic design to account for uncertainties in the design input data and the modeling techniques. In general, these conservatisms and uncertainties are not quantified. Nor is the way they compound through the design process known. Needs therefore exist to investigate safety margins and to evaluate the contribution of these margins to the overall safety of nuclear power plants. The concept of balance of safety between seismic and operational loads must be addressed through a better understanding of the behavior of structures, systems, and components subjected to postulated earthquakes.

In order to deal with these issues, the following questions must be addressed:

1. Can the complex, detailed methodology developed for seismic risk be sufficiently simplified to be of practical use as part of PRAs?

2. Can the methodology be adequately validated?

3. What is the total seismic risk for a PWR and BWR?

4. What are the major contributors to seismic risk?

5. Which uncertainties may be reduced, through research or improved design requirements, that will result in a reduction of the overall risk?

6. How should operating plants be treated in light of changing seismic loads and design criteria?

6.5.2 Research Program Objective

The objective of the engineering characterization of seismic input project is to develop recommendations for choosing design input-motion at the foundation level of nuclear power plant structures consistent with a free-field motion specified at the surface.

The Seismic Safety Margins Research Program (SSMRP) is a multiphase, multidisciplinary program whose overall objectives are to develop improved and simplified methods for the seismic safety analysis of nuclear power plants, to assess this methodology by applying it to the analysis of selected cases, and to provide data and insights serving as bases for revisions to seismic design standards and future licensing decisions.

6.5.3 Relationship to Other Programs

The broad technical nature of the SSMRP provides relationships with all mechanical systems and components and structural research programs involving seismic response loadings. The loads combination program has provided input to the SSMRP on the probability of failure of the primary coolant loops. SSMRP methodology will give results needed to make value judgments on requirements for the seismic qualification of electrical and mechanical equipment and for research in this area. Seismological research conducted by the Earth Sciences Branch of RES will improve the seismic hazard data used as input to the SSMRP calculations. Results from research programs in the areas of mechanical component integrity, equipment qualification, and containment integrity and other structures will be used to assist in validating the methodologies and data base of the SSMRP.

The SSMRP also has ties to systems research involving PRA, such as the IREP/NREP (Interim Reliability Evaluation Program/National Reliability Evaluation Program) effort. The SSMRP will provide the seismic risk methodology that is currently lacking in these programs.

Seismic risk studies are also being conducted in Germany and Japan. Seismic ground-motion studies have been conducted in the U.K. EPRI is conducting shaker tests for piping systems that will produce fragility data useful to the SSMRP. Agreements with these organizations have been made, or are being pursued, to obtain the data or study results.

6.5.4 Background and Status

By the end of FY 1983, the engineering characterization of seismic input project will have provided a quantitative basis for choosing response spectra based on translational motions. The effect of duration of motion, distance from the epicenter, short pulses of high acceleration, observed earthquake damage, and earthquake magnitude will be considered. This project will also have provided results and recommendations for methods to be used as motion at the foundation levels. The results will form the basis for developing site-dependent or regionally based input.

The SSMRP sensitivity studies to be completed in FY 1982 will determine how the various input parameters, modeling techniques, and the uncertainties associated with these contribute to the overall seismic risk. These studies will give insights into the final analysis results of the Zion risk study and will show areas where the methodology can be simplified (or where more detail is needed).

The final SSMRP Zion analysis will be completed in early FY 1983. This seismic risk study uses a full probabilistic approach in calculating responses through the seismic methodology chain (i.e., seismic input, soil-structure interaction, major structure response, and subsystem response), calculating component failures using fragility curves, and then performing a systems (event tree/fault tree) analysis to determine the probability of release. The results will provide probabilities and uncertainty bands for the different release categories (similar to the WASH-1400 categories) and for component and subsystem failures. The major contributors to seismic risk at the Zion plant will also be identified.

In FY 1982, the Senior Research Review Group (SRRG) will identify and set priorities for NRC user needs for the SSMRP. Its decisions will be used as a basis for future development and application work for this program. Anticipated projects to begin in FY 1982 include the development of simplified models/methods for both PWR and BWR seismic risk analyses and assessment of this methodology through application to Zion and also to a BWR. The simplified models/methods will reduce the analytic effort and cost. Justification for simplification will come from the results of the sensitivity studies. The BWR analysis will expand the seismic risk data base and will provide comparisons between PWR and BWR seismic risk probabilities.

6.5.5 Research Program Plan

In FY 1984, methods developed and recommended by the engineering characteristics of seismic input project will be calibrated against data available from seismic arrays and field tests. The seismic array data documenting the variation of earthquake motion with depth will be taken from available recordings (Japanese). The field data sources will mainly be those resulting from explosive tests.

The SSMRP will end at the close of FY 1984. Documentation of the BWR seismic risk analysis will be completed in that year. This and the previous documentation of the Zion seismic risk analysis and the sensitivity studies will contribute to the data base from which generic licensing decisions can be made on seismic design.

Using this data base, conclusions and recommendations will be made as part of the final documentation of the SSMRP. Estimates of conservatism (or lack of conservatism) of the present SRP seismic safety requirements will be made. In some cases, changes will be recommended to obtain improved deterministic requirements.

The SSMRP methodology, including simplified models/methods, will be completed and documented in (or before) FY 1984. The HAZARD, SMACS, and SEISIM computer codes will be available for NRC staff and public use. The HAZARD code assesses the seismic hazard at a given site, SMACS computes in-structure and subsystem seismic responses, and SEISIM calculates structural, component, and systems failure probabilities and radioactive release probabilities. Simplified models/methods developed as a result of the sensitivity studies will be available in FY 1984 for use in evaluating other seismic risk analyses. The treatment of seismic events in the IREP/NREP program will either use SSMRP simplified methodology directly or have the techniques employed validated by the SSMRP.

The SSMRP methodology uses fragility curves to describe the probability of structure and component failure as a function of the local seismic response. Fragility curves are needed along with seismic responses (calculated by the SMACS code) and the local seismic hazard curve in determining probabilities of seismic failure in the system (event tree/fault tree) analysis performed by the SEISIM code. Experimental adjustment of the fragility curves provides a means to investigate how increasing or decreasing the strength of components, and the uncertainties associated with these strengths, will contribute to overall system failure and release probabilities. This technique can be used in evaluating how the degradation of equipment, such as caused by aging, affects overall safety. Research of this type will begin in FY 1985 after completion of the SSMRP.

Fragility curve adjustment will also be a useful tool in future studies to determine the effectiveness of proposed new or increased seismic equipment qualification for operating plants. The SSMRP's methodology identifies dominant contributors to risk on a systems basis. This will show where upgrading equipment will be helpful, and where it will have no safety benefit. Thus, research for seismic equipment qualification using SSMRP methodology will begin and continue through the late 1980's.

The SSMRP results will place uncertainty bounds on the final release risk probabilities and on the probabilities of component and system failures. Uncertainties will be separated into random and modeling ("systematic") components. Since modeling uncertainties can be reduced by augmented testing or analysis, the results of the SSMRP will indicate areas where new research will be beneficial in removing seismic risk uncertainties. If random uncertainties dominate in certain areas, increased testing or analysis would not be fruitful.

Research to determine the contribution of design, construction, and operator errors to seismic risk may use some SSMRP-type methodology. The treatment of gross errors would require the development of new methodology.

Regulatory Guide 1.29, "Seismic Design Classification," will be revised and issued as Revision 4 in FY 1985. Results from the SSMRP and mechanical equipment qualification program will serve as bases for this revision.

A year-by-year description of planned accomplishments follows. The relationship between this work and licensing is shown in Figure 6.4. To some extent, the year-by-year description is based on the revised SSMRP program plan to be issued in FY 1982. Anticipated changes in the development of this plan will affect future products.

THE RELATIONSHIP BETWEEN LICENSING AND RESEARCH FOR SEISMIC DESIGN RESEARCH

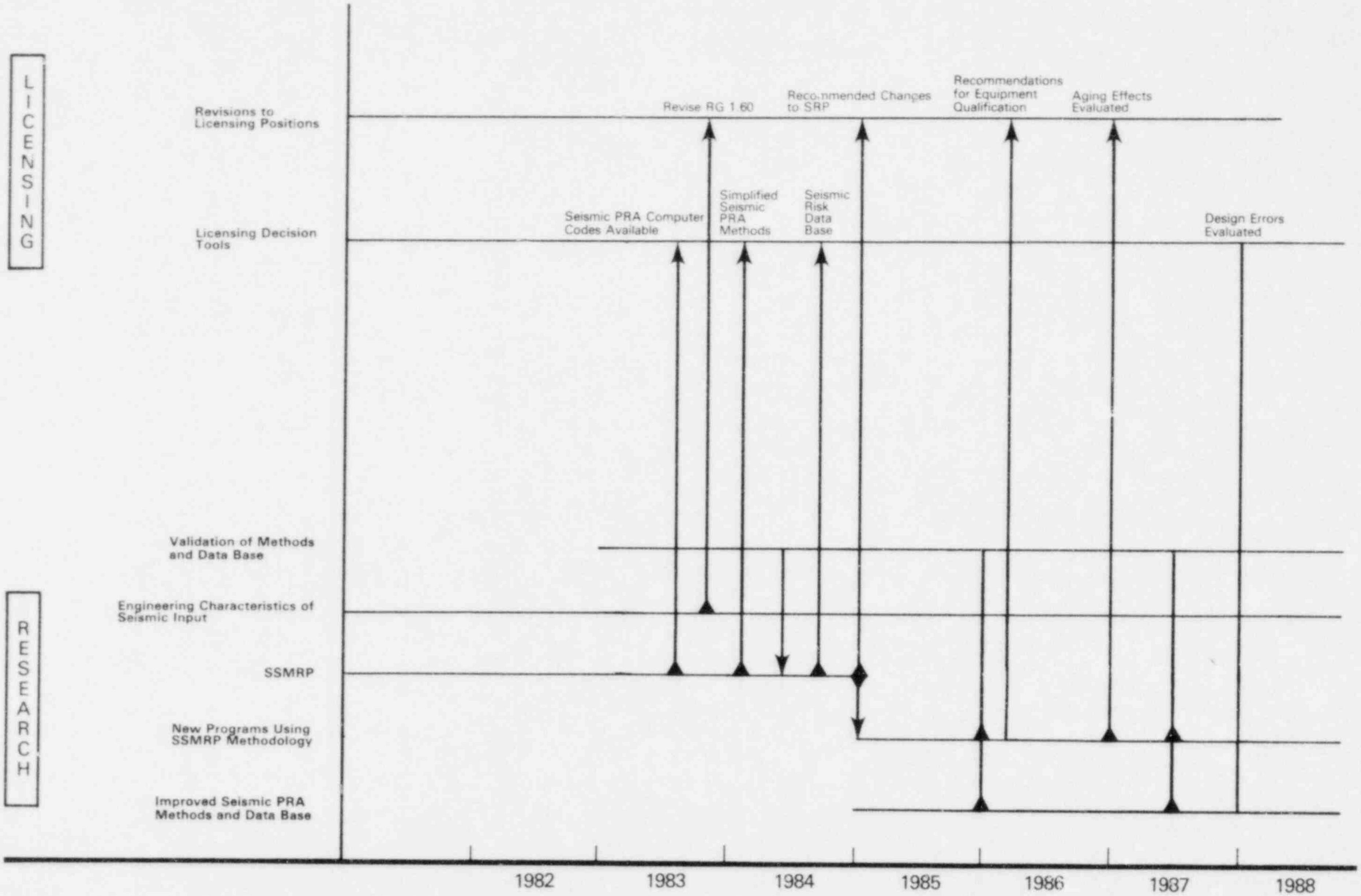

Figure 6.4

FY 1984	Conclude SSMRP; document SSMRP BWR seismic risk analysis,and make available to licensing staff simplified models/methods; validate SSMRP to the extent possible with data available from experiments and experience from other domestic and foreign facilities and research programs; make recommendations on SRP seismic design criteria using results from SSMRP data base; and calibrate methods of engineering characteristics of seismic input project.
FY 1985	Determine priorities in other research efforts to improve seismic design using methods developed in the SSMRP (these research efforts include the equipment qualification and containment integrity programs and research to evaluate the contribution of aging); merge seismic risk methodology with PRA methodology to be used in safety goal assessments; and use feedback from equipment qualification and containment integrity programs to improve seismic risk data.
FY 1986	Complete research applying the developed methodology to studies on equipment qualification and aging.
FY 1987	Begin research on contribution of design, construction, and operator errors to seismic risk using seismic risk methodology.
FY 1988	Complete design, construction, and operator error studies.

6.6 Structures

The planned research on power plant structures is concerned with assessing and reducing, where necessary, the risk to the public from operation of both new and older plants. The research is related to containments and other Category I structures. A program based on the need for structural safety of waste management and storage facilities will be developed during FY 1982. A program will also be established for estimating seismically induced lateral movements on safety-related dams and embankments.

6.6.1 Issue

Based on risk assessments for those accident scenarios in which fission products are released to the containment, subsequent containment failure can result in the greatest risk to the public. The regulatory concern in this element of the structures program is that the failure modes and associated load levels for containment structures cannot be predicted with any real confidence by state-of-the-art methods. Assessments of the risk posed by loads outside the design basis, such as hydrogen burns, and estimates of the effectiveness of proposed mitigative steps both require an ability to predict both the time and the way in which a containment will rupture or leak excessively, especially at penetrations. The associated technical and safety issues involve the ability to predict deformations for the wide array of containment types, relate these deformations to containment leakage, investigate the effects of aging on penetrations, and determine the sensitivity of predictions to uncertainties in the loads associated with accident and extreme environmental scenarios.

As a result of knowledge gained over the past decade, the operating and loading criteria for nuclear plants have become more demanding. This raises as an issue the safety of those operating plant structures designed to less demanding criteria. The corresponding technical and safety issues involve the adequacy of the data base used to support the analytical methods, codes, and standards that have been and are being used in the design and construction of the plants. Another issue is the possible degradation of structural performance due to aging or modifications of plant structures.

An issue that will receive later consideration in the structures program is the structural safety of waste management and storage facilities. Operating environments and long-term aging effects may result in unique structural problems.

A final issue is that of safety-related facilities mounted on dams and embankments being subjected to seismically induced lateral movements.

6.6.2 Research Program Objective

The objective of the containment integrity research project is the development and verification of methodologies that are capable of reliably predicting the capacity of containment structures under accident and extreme environmental loadings. The reliability of any predictive method must be verified through experiments. This project contains a combined analytical and experimental effort toward the establishment of reliable methods of predicting the performance of containment structures under accident and extreme environmental conditions.

The emphasis is, however, on the experimental effort that will provide data to help validate analytical methodology proposed by licensees and applicants. Test results will provide benchmarks for checking the ability of computer codes to predict the deformations of containment structures under increasing internal pressure, up to the point of failure due to excessive leakage. Steel containments and prestressed and reinforced concrete containments will be studied for pressure and seismic loadings. Loadings from accidents leading to hydrogen explosions are of particular interest.

The objective of the plant structures element of the structures program is to provide the NRC staff with methods, techniques, and criteria for evaluating the adequacy of new and operating nuclear power plant structures during and after anticipated operational, upset, and accident conditions, including environmental phenomena such as earthquakes. This information will provide the licensing staff with (1) an improved basis for review of the methods of analysis, design, and construction presented by applicants and licensees, (2) a better understanding of the behavior of Category I structures subjected to loads and combinations beyond those used in design, and (3) an ability to assess the risks to the public as part of structural adequacy determinations. It is intended to provide the staff with appropriate computer programs that will be of direct use to review the acceptability of plant structures and the methods for analyzing these structures. This effort is in part intended to improve on the present practice, which combines loads based largely on judgment and may well overemphasize the importance of some loadings and underemphasize the importance of others.

The first objective of the structural effort in waste management will be to determine what structural research is needed for nuclear fuel waste-processing and long-term storage facilities. To satisfy those needs, appropriate structural research programs will be developed and carried out. In turn, appropriate criteria and standards will be prepared.

The objective of a structural study on dams and enbankments is to review existing information on their seismically induced lateral movement and to determine the research necessary to predict the response on mounted safety-related equipment. A program will then be developed to perform the necessary research.

6.6.3 Relationship to Other Programs

To avoid potential duplications of effort and to promote, as far as possible, the shared use of costly facilities, the structures program includes an ongoing effort to identify potential joint research programs with interested U.S. Government agencies and industry groups and with foreign countries. We have already identified and will use the results of some ongoing and planned research programs both in the U.S. and abroad. There is, and will continue to be, significant interaction between the containment integrity program and other NRC-sponsored programs related to accident evaluation and mitigation.

Particularly close coordination will be maintained with the programs on core melt/coolant interaction, hydrogen generation and control, and improved containment. In addition, there will be interactions with the reactor safety study methodology applications program, the Seismic Safety Margins Research Program (SSMRP), and subsequent seismic probabilistic risk assessment programs. There will also be interaction with other U.S. programs. Contributions to the containment integrity program are anticipated from EPRI by their provision of analytical predictions of capacity to be compared against test results. There will be coordination with the containment capability program being contemplated by DOE and with the containment overpressure response effort sponsored by the Industry Degraded Core (IDCOR) Steering Group.

Two foreign programs in addition to those already in the technical literature have been identified as potential sources of information. One is the proposed test-to-failure of a model prestressed concrete containment to be conducted in the U. K. The other is the planned testing, on a shake table in Japan, of containment models to simulate seismic response.

The plant structures and the containment elements also relate to other ongoing programs, including Massachusetts Institute of Technology, Cornell University, and Portland Cement Association programs of testing and analysis of small sections of reinforced concrete. These programs are also yielding results related to the modeling and scaling requirements for reinforced concrete.

The breadth of the plant structures program results in its being related to many other programs under way. Table 6-2 shows the relationship between the programs in the plant structures element and other U.S. and foreign programs.

The structural program on waste management will interact with other NRC, DOE, industry, and foreign safety and licensing programs associated with high-level and low-level waste management and storage.

6.6.4 Background and Status

Accident conditions such as hydrogen burn can produce very high pressures inside containment structures. The effects of these pressures on containment integrity must be determined.

The FY 1982-1983 containment integrity program is limited to understanding the behavior of steel containment structures under static overpressures. The program plan is to conduct overpressure tests on six 1/32-size prototypical containment models. Similarly, a 1/10-size model will be designed and tested. These planned tests are shown in Table 6-3. Also, analytical studies will be performed on the behavior of concrete containments loaded beyond the elastic limit of the reinforcement steel and prestressed tendons. Results of these studies will be used in the design of reinforced concrete containment test specimens. Careful attention will be given to the design of test specimens to adequately model plant structures. Many important concrete and reinforcement material and geometric parameters must be properly modeled if test specimens are to simulate the expected response of various full-sized structures. The need and feasibility for separate testing of penetrations will also be evaluated. Related data from Canadian tests on CANDU containment models will be obtained, and interfacing with the FY 1983 British SNUPPS containment model tests will be accomplished.

The FY 1982-1983 structures program has several projects addressing the adequacy of analytical, design, and construction methods. Programs on benchmarking of computer codes, adequacy of codes and standards, shell buckling, and dynamic testing will continue. An assessment (to determine limitations and applicability range) of the theoretical basis, numerical algorithms, and underlying assumptions of the commonly used computer codes will also be performed. The adequacy of industry codes and standards will be examined through an analytically supported experimental program addressing load-carrying ability beyond design loads. The results of static and dynamic buckling tests on model steel containment shells with penetrations will be compared to analytical predictions to generate recommendations for shell buckling criteria. The evaluation of analytical models used in conjunction with the explosive tests, simulating seismic events, at the HDR in West Germany will be completed. This will be used to identify methods that can reliably predict responses to low-level seismic motions.

During FY 1982-1983, the testing of small- and large-scale Category I type shear-wall elements will be completed. These structural elements will be subjected to static and dynamic loadings of sufficient magnitude to cause inelastic behavior. The test results will provide a basis for a better understanding of the damping and nonlinear response behavior of structures subjected to large dynamic loads. A test plan for typical Category I floor slab-shear wall structures will be completed in FY 1982. Test structures will

TABLE 6-2

RELATIONSHIP OF SAFETY OF PLANT STRUCTURES PROGRAM TO OTHER PROGRAMS

Program Area	NRC-Sponsored Research (Performing Organization)	Other U.S. Research (Sponsoring Organization)	Foreign Research (Country)
Bases for Review of Analyses, Design, and Construction Methods		SIMQUAKE data (EPRI) Earthquake Hazard Reduction Program (NSF)	HDR data (W. Germany)
Behavior of Category I Structures Beyond Design	Test of reinforced concrete wall elements (Portland Cement Assoc.)	Static and dynamic shear wall tests (NSF)	Static Testing of Shear Walls (New Zealand) Reactor wall tests (Japan with NSF participation)
Risk/Benefit/Cost Perspective for Structural Engineering	Seismic Safety Margins Research Program (LLNL)	Building code research (NSF)	Building code re search (W. Germany) (Great Britain)

TABLE 6-3

PLANNED EXPERIMENTS FOR STEEL CONTAINMENTS

Experiment	Size	Description
SC-1 & SC-2	1/32	Clean shell experiment to serve as the control and to provide data for basic 2-D postyield method evaluation
SC-3 & SC-4	1/32	Ring stiffened shell experiment to provide additional postyield method evaluation data and structural effects data
SC-5 & SC-6	1/32	Ring stiffened shell with primary penetrations to provide data for 3-D postyield method evaluation
SC-7	1/10	Ring stiffened shell with penetrations using conventional construction methods

be fabricated in FY 1983. The release of a regulatory guide on anchoring component and structural supports in concrete is also anticipated.

Research efforts have produced many analytical methods for the modeling and analysis of the dynamic soil-structure interaction (SSI) during seismic events. Because of the uncertainties associated with this interaction, the acceptability of these analytical methods is in question. To provide a tool for the NRC licensing staff to judge the adequacy of SSI methods being used, a program will be introduced during FY 1983 to develop an appropriate set of SSI benchmark standard problems.

During FY 1982-1983, studies concentrating on a probabilistic design basis for structural engineering will be addressed through the load combination project. During this time period, the establishment of a data base for operational, abnormal, and environmental loads will be completed. Probabilistic load combination methods for seismic Category I concrete structures will also be developed to provide the staff with a more rational basis for choosing load combinations and associated load factors.

The program emphasis will be concentrated on the effects of seismic loadings. The major uncertainties about structural performance under severe environmental loadings are related to earthquake effects; uncertainties about responses to tornado and flood loadings are felt to be small by comparison. Research on tornado occurrence and intensity, described in paragraph 9.2.5.2 of this plan, will be completed in FY 1984. Flood occurrence will be studied in the hydrology program described in paragraph 9.2.5.3. Parallel probabilistic studies of the risk associated with severe floods will be performed in the FY 1984-88 time frame. This work is described in paragraph 10.2.5.4.

The above programs are summarized in Table 6-4.

TABLE 6-4

FY 1982-1983 STRUCTURES PROGRAM

Program Area	FY 1982	FY 1983
Bases for Review of Analysis, Design, and Construction Method	Validate analytical methods and modeling techniques	Assess theoretical bases, numerical algorithms, and assumptions used in computer codes
	Experiments investigating buckling behavior of steel containment shells	Experiments investigating buckling behavior of steel containment shells
		Evaluation of HDR predictions
Behavior of Category I Structures Beyond Design	Static and dynamic experiments on small-scale shear-wall elements	Static and dynamic experiments on large-scale shear-wall elements
Risk/Benefit/Cost Perspective for Structural Engineering	Establish data base for operational, abnormal, and environmental loads	Develop probabilistic load combination methods for Category I structures

6.6.5 Research Program Plan

6.6.5.1 Containment Integrity Program

Effort in FY 1984 will concentrate on three items. One is an evaluation of analytical predictions of steel containment capacity in light of the experimental results. The second is tests-to-failure, under static overpressure, of models of reinforced concrete containments. Currently, six tests are anticipated. It is anticipated that all models will be approximately 1/10 the size of a typical containment. However, difficulties in ensuring representative performance of the models, particularly in regard to liner thickness, bond and cracking behavior of the concrete, and properties of the reinforcement, may dictate a larger size. The first two models will be without penetrations or seismic reinforcement. They will serve as controls and will provide data for the evaluation of two-dimensional analytical predictions of postyield behavior. The next two models will include seismic reinforcement, but no penetrations. These models will provide additional data for the calibration of two-dimensional analyses. The final two models will include both seismic reinforcement and penetrations and will provide data against which three-dimensional predictions can be compared. Finally, the comparison of predictive methods for prestressed concrete containment behavior against the British data will be completed.

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

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

The following results are anticipated during FY 1984-1988. The relationship between this work and licensing is shown in Figure 6.5.

FY 1984	Compare estimated steel containment capacities with experimental static pressure results and reinforced concrete acceptance criteria for combined shear and biaxial tension.
FY 1985	Compare predicted capacities for prestressed and reinforced concrete containments with experiments under static pressure.
FY 1986	Compare predictions of steel containment capacity under dynamic pressure loads with experimental results.
FY 1987	Compare predictions of capacity for reinforced and prestressed concrete containments under dynamic pressure loads with experimental results.
FY 1988	Conduct initial tests of containment models under simulated seismic loading.

During the course of the program, regulations and regulatory guides associated with various facets of containment loading, inspection, and leakage testing will be issued.

6.6.5.2 Safety of Plant Structures

One facet of the structural engineering research program will concentrate on questions related to the analysis, design, and construction practices used for nuclear power plants and implications of those practices over the design life of plant structures. Three specific areas will be studied: (1) the ability of computer models used for structural design to predict responses of plant structures to normal and extreme loadings, (2) the long-term behavior of materials

THE RELATIONSHIP BETWEEN LICENSING AND RESEARCH FOR CONTAINMENT INTEGRITY

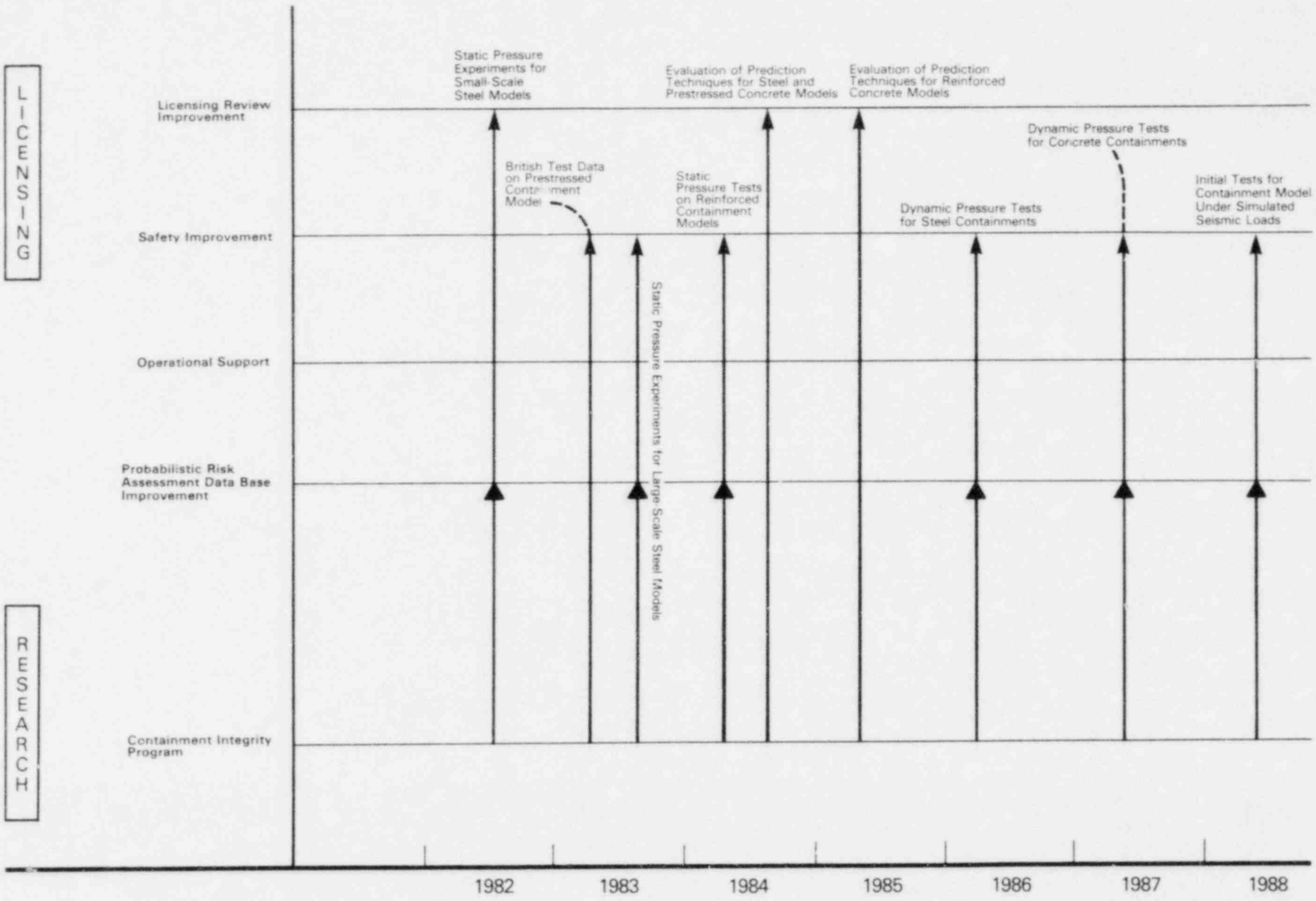

Figure 6.5

used in plant structures, and (3) the effects on plant safety of design and construction methods.

Another major facet of the structural engineering research program is the development of an ability to predict the performance of plant structures at loads beyond those used for design. The buildings, other than containments, that house safety-related equipment at nuclear power plants are often heavy, concrete shear-wall structures. These buildings use 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 effort is aimed toward assessing the margin of failure and failure modes 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 (1) damping behavior, (2) characterization of damping values due to cracks and loose joints, (3) nonlinear and cyclic response from large dynamic loads, and (4) structural damage induced by dynamic environments. This work 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 loading outside the design basis.

In FY 1984-1986, effort will be concentrated on the completion of ongoing programs related to the verification of structural design by dynamic testing methods, benchmarking of computer codes used in structural design, and experimental investigations of analytical methods used to predict shell buckling.

The role of probabilistic risk analyses in determining the adequacy of plant designs will continue to grow during FY 1984-1988. The activities described previously, centering on confidence in methods used for plant design and construction and on the ability of plant structures to sustain loads beyond their design level, are deterministic in their formulation. However, they will provide the basis for developing the more accurate models necessary to improve probabilistic risk analysis methods. Studies of performance beyond design load will provide the data base necessary to improve the predictions of consequences used in risk models. Also, the studies related to the sensitivity of plant safety to discrepancies in design and construction will provide input for the probabilistic risk analysis studies that will be performed on the topic usually categorized as "design and construction errors."

The third, and final, aspect of the structural engineering research program deals with the development of probabilistically based methods to be used in decisionmaking on questions related to structural adequacy at nuclear power plants. Initial work will concentrate on the development of methods to permit a more rational selection of load factors or allowable stresses and combinations of loads to be used for structural design at nuclear power plants. When this is accomplished, efforts will be made to specify rules for combining loads on a risk-consistent basis. The next step will be to use work done to assess the behavior of plant structures beyond design levels to incorporate estimates of structural performance near failure. Depending on the success of the foregoing, the plan is to develop methods to predict the probability that a given plant structure will sustain damage that could lead to public risk. Work done under the SSMRP will be used in this development.

The following results are anticipated during FY 1984-1988. The relationship between this work and licensing is shown in Figure 6.6.

FY 1984	Recommend load combination criteria for noncontainment structures at nuclear power plants, acceptance criteria for low-level dynamic testing of nuclear plant structures, and methods for estimating buckling loads for steel shell structures.
FY 1985	Recommend load combination criteria for containment structures, methods suitable for extrapolating the results of low-level dynamic tests of structures to the levels associated with failure, and knockdown factors to be used with analytical predictions of containment buckling; and evaluate the computational tools used in calculations of soil-structure interaction effects.
FY 1986	Assess acceptance criteria for load combinations; assess computer codes used for the dynamic analysis of nuclear plant structures; and make recommendations for the dynamic testing of undamaged nuclear plant structures.
FY 1987	Develop probabilistic methods for assessing the adequacy of containment structures; provide experimental results for large-scale models of shear-wall structures; and recommend possible ways to evaluate structural damage after an earthquake or a plant accident.
FY 1988	Using probabilistic methods, investigate sensitivity of parameters used to assess containment structure adequacy; and compare predictions of structural response for large-scale shear-wall models with experimental results.

Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," will be updated in response to the results of this research.

6.7 Qualification of Electrical Equipment

6.7.1 Issue

The NRC is proposing to change its regulations pertaining to nuclear power plants to include an amendment to 10 CFR Part 50 concerning the environmental qualification of electrical equipment. There are many issues related to this rule requiring technical resolution. A major issue is the validity of the procedures, as prescribed by national standards and regulatory guides, for the

THE RELATIONSHIP BETWEEN LICENSING AND RESEARCH-SAFETY OF PLANT STRUCTURES

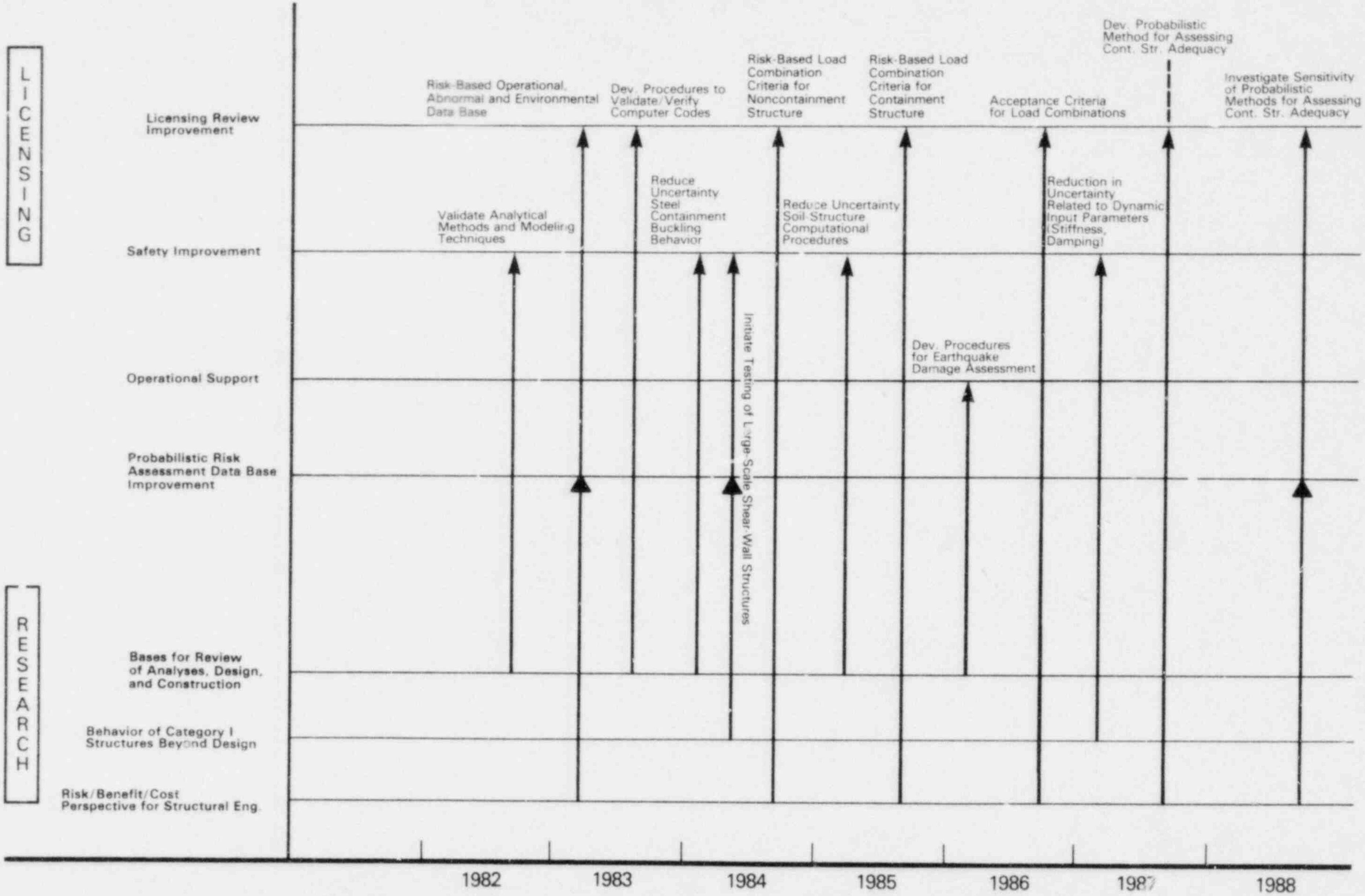

Figure 6.6

qualification of electrical equipment by testing. More specific concerns include the effects of synergism, radiation dose rate, the order or sequence in which the qualification steps are performed, preaging test conditions, and the simulation parameters and profiles used in the procedures. In selecting the procedures to be studied, the priority of the research will be considered based on the risk contribution, completeness, need and practicability, and implementation of the procedures. Other specific concerns to be addressed include determining the validity of using two identical prototypes, one for loss-of-coolant-accident (LOCA) testing and one for seismic testing, versus the more conservative test procedure wherein a single prototype is subjected to seismic and environmental testing in sequence. The issue of preaging of electrical equipment in a mild environment prior to seismic testing will also be addressed.

6.7.2 Research Program Objective

The electrical equipment qualification research program has as its objectives:

1. The evaluation and validation of equipment qualification procedures,

2. The improvement of available procedures and development of new procedures,

3. Research to provide a technical basis for safety criteria and requirements for electrical equipment qualification, and

4. Support for licensing review and decisions on the qualification of equipment in new and operating nuclear power plants.

6.7.3 Relationship to Other Programs

Extensive efforts to qualify electrical equipment for environmental and seismic conditions are being made by the nuclear vendors, independent industrial testing laboratories, and electrical utilities. These efforts generally consist of electrical equipment qualification tests of specific equipment performed in accordance with the national standards. The results of these qualification tests are submitted for review to the NRC by the nuclear power plant licensees as a basis for qualifying their equipment. In the course of this review, procedures requiring validation or requirements for research are identified. Therefore, the direction of research is strongly influenced by industrial qualification testing efforts.

A small research effort is being sponsored by the utility industry through EPRI. This effort currently includes a study of equipment survival for a hydrogen burn, seismic testing in a mild environment, a study to collect environmental exposure data from nuclear power plants, and the maintenance of a data bank of equipment qualification results. Efforts completed in 1981 included a compendium or review of aging theories and a compilation of radiation damage thresholds for organic compounds. Information and progress reports on programs concerning electrical equipment qualification are exchanged between EPRI and NRC at periodic joint technical coordination meetings and through the exchange of reports.

Programs to study the environmental qualification of electric cables are being conducted in several foreign countries. Representatives of the Japanese Atomic Energy Research Institute (JAERI) and the Japanese industry have presented papers on their research at the Water Reactor Safety Meetings and through visits by the Japanese to the NRC headquarters and laboratories where NRC research is being conducted. The data obtained to date pertains, for the most part, to the behavior of polymer materials during aging and radiation. A multiyear program of cooperative research between the NRC and the French Commissariat à l'Energie Atomique (CEA) on the thermal and radiation aging and LOCA simulation testing of a wide range of polymers and plastic materials used in nuclear power plants is being developed for initiation in FY 1982. Information exchanges and cooperative arrangements will be pursued with other foreign research laboratories where practical.

The efforts of DOE to evaluate the failure modes and assess the qualification of electrical equipment removed from TMI-2 are being closely followed. NRC is participating in this program by providing technical support and participation on the committees reviewing these efforts. Close coordination is maintained between the electrical equipment qualification research effort and the efforts to develop a technical basis for environmental qualification and seismic testing of mechanical equipment (Section 6.4).

6.7.4 Background and Status

The following activities are included in the FY 1982-1983 equipment qualification research program:

1. An independent verification testing program directed at research on equipment qualification testing methodology was initiated in FY 1981. Electric cable splices and repairs for cross-linked polyolefin and polyethylene insulated cables containing fire retardants were completed in FY 1982. Other qualification tests completed in FY 1981-1982 included an electrical penetration and several designs of electrical connectors. Some of the electrical penetration connectors exhibited low resistance in LOCA testing. This was found to be in part associated with the technique of accelerated thermal aging. Qualification tests using a different aging method are to be performed in FY 1982. Nine (single prototype) safety-related solenoid valves will be tested in FY 1982 with each prototype going through the qualification sequence: thermal aging, radiation aging, wear simulation, exposure to dynamic and seismic testing, and exposure to accident and postaccident steam environmental conditions. Other qualification tests to be conducted in FY 1982-1983 include PWR resistance temperature detectors, pressure switches, coaxial cable, and a second type of electrical penetration.

2. The general scoping study on aging will be initiated in FY 1982 and completed in FY 1983. Details of this effort are discussed in Section 6.1.4. The effects of thermal aging, radiation dose rate, synergism, and the order or sequence of steps in qualification testing will be studied in FY 1982-1983.

   Studies using insulation and sheath samples from EPR (ethylene-propylene-rubber) industrial electric cables will be completed.

An agreement is being negotiated between the French CEA and the NRC for a joint program to assess the influence of performing thermal and irradiation aging in sequence or simultaneously prior to accident simulation testing with steam. The LOCA testing would be conducted in the CESAR facility in France and the preaging would be provided in U.S. facilities. A wide range of polymers found in nuclear plant cables, gaskets, and seals, including elastomers, thermoplastics, and thermosetting compounds, will be tested. The program is expected to start in late FY 1982 and run to FY 1986. The cooperative NRC/CEA program may be extended to include other LOCA methodology testing issues in later years.

Procedures and recommendations for the accelerated aging of safety-related materials and components are to be developed and guidelines are to be written in the form of an accelerated aging methodology handbook for use in licensing reviews.

3. Equipment qualification tests for the case of exposure to a LOCA or steam-line break are to be continued using the high intensity adjustable cobalt array (HIACA) facility that was completed in FY 1981. This facility provides the capability for simultaneously exposing equipment and materials to gamma radiation and a steam atmosphere. It is being used in the independent verification test program and to support other equipment qualification research tests. The capability of the facility is to be expanded in FY 1982-1983 to include provisions for providing superheated steam transients and a partial oxygen or nitrogen gas overpressure in the test vessel plus an improved data-logging capability. A single pressurized test chamber for accident and postaccident testing of materials and components is available for use with this facility. An additional test chamber is to be built in FY 1982-1983 to accommodate the increased demands for test space.

   A joint NRC-French EDF (Electricité de France) test to assess the effect of varying oxygen overpressure during a LOCA qualification test was initiated in FY 1981 for tensile specimens of polyethylene, polyvinyl-chloride, neoprene, hypalon, EPR, silicone, and Tefzel. The test is to be completed and reported in FY 1982.

   Other questions to be addressed in FY 1982-1983 include assessment of the effects of steam impingement on specimens in the test chamber, heat transfer, chemical-steam ingress into equipment, and dust and other contaminants on equipment behavior in a LOCA.

4. Studies were completed in FY 1980-1981 to evaluate radiation simulator adequacy for qualification testing of Class 1E safety-related electrical equipment. It was found that the source "hypothesized" in Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," more nearly represents a conservative, unterminated LOCA condition. More realistic LOCA radiation signatures will be developed in FY 1982-1983 using the new source terms from NUREG-0771 and NUREG-0772 and results from follow-on studies being made in FY 1982 under the degraded core research program. The results of this research plus that discussed below will be provided for revising Regulatory Guide 1.89.

Techniques and codes for calculating the radiation dose to equipment in containment will be evaluated in FY 1982. A code for use in auditing licensee assumptions and equipment dose rate estimates is to be made operational in FY 1982. Research studies to determine generic dose rates will be made in FY 1983. Actual equipment dose rates and environments during normal power plant operation will be obtained in FY 1982-1983 by in-plant measurements and the review of data logs at selected nuclear plants.

Conservative assumptions are currently made in the qualification testing by treating beta radiation as equivalent to gamma. In order to assess the conservatisms in the above assumptions and develop a more realistic beta/gamma damage equivalence, elastomers and polymers are to be irradiation-tested with beta alone, gamma alone, and beta plus gamma radiation sources over radiation energy levels representative of the postulated accident radiation spectrum. The in-depth damage will be measured in FY 1982, and testing will continue in FY 1983.

An interim guideline manual for radiation simulation methodology in equipment qualification testing is to be issued in FY 1982. This manual will be published in FY 1983-1984 on completion of the above research.

5. Equipment is to be removed from TMI-2 for examination and testing under the DOE research program. The NRC is participating in this effort by having representatives on various committees responsible for equipment selection and examination recommendations. A radiation detector, HP-RT-0211, was removed from TMI-2 in FY 1981. It was found that transistor punch-through caused a degraded output. Other problems associated with water ingress due to the mounting configuration and cable seals were found. A solenoid valve, other radiation monitors, transmitters, switches, and electric cable are to be removed in FY 1982. Damage and failure modes are being studied by in situ measurements and laboratory testing following removal. NRC expects to support the laboratory examinations in FY 1983 by measuring electrical and physical properties of cable samples removed from containment and virgin samples.

6. Seismic Qualification of Electrical Components: The effects of aging on electrical components followed by dynamic and seismic vibration tests will be studied in FY 1982 for solenoid valves located in a harsh environment. Other components such as pressure switches and resistance temperature sensors will be tested in FY 1983.

   Seismic tests will be conducted in FY 1983 on electrical components with thermal aging but without radiation in order to evaluate the requirements in IEEE-344 for mild environments.

6.7.5 Research Program Plan

The research on equipment qualification testing procedures, including aging and synergism, is expected to peak around FY 1984 and to be largely completed by FY 1986. The effort on independent verification testing of electrical equipment and research to resolve issues identified in the audit of equipment qualification data submitted by operating nuclear power plant licensees and plants to be licensed under the existing order and proposed rule for equipment qualification is expected to increase through FY 1987 and lead to added requests for research support through FY 1988. Other new issues such as possible proposed requirements for survival of electrical equipment after a hydrogen burn and possible proposed new seismic testing requirements are expected to be added to the research program plan in FY 1984-1986.

6.7.5.1 Equipment Qualification Research Tests

Environmental qualification research tests are to be conducted for selected electrical equipment. The list of equipment and priority of testing will be determined based on needs for increased assurance of adequate qualification, evaluation of synergism and aging requirements, clarification of uncertainties in suppliers' qualification testing methods, high failure rate, and contribution to risk. Approximately three or four tests per year are to be made.

Testing of limit switches, level switches, connectors, cable, and transmitters will be completed in FY 1984. In FY 1985, testing of electric motor prototypes and valve actuators will be completed. The categories of electrical equipment to be tested during FY 1986-1988 will depend on the results of reviewing the data submitted by licensees for qualifying their electrical equipment.

6.7.5.2 Procedures for Equipment Qualification Testing

The following will be completed in FY 1984: testing to assess the importance of the superheated steam transient vs. saturated steam in simulating the LOCA or steamline break profile; testing to evaluate the effect of dust and other contaminants on the qualification of electrical connectors and switches; and studies to determine whether oxygen sweepout is leading to conservative degradation results with polymers.

In FY 1985, the assessment of the failure modes and testing of equipment removed from TMI-2 will be completed and will be factored into rules and standards as new design and qualification requirements. The NRC-CEA study concerned with LOCA testing of a wide range of polymers under both sequential and simultaneous radiation and LOCA steam exposure will be completed and reported in FY 1985. These data will provide for assessing the adequacy of vendor qualification of components and materials for operating nuclear power plants. Requirements will be revised as necessary for to-be-licensed plants.

An assessment of electrical motor and large-scale electrical component equipment qualification will be completed in FY 1986, and prototype-scale testing will be evaluated as a means of qualification. The radiation damage threshold of solid state electronics (transistors, etc.) and in-containment behavior of electronics will be evaluated in FY 1986. Revisions to regulatory guides and standards will be made if necessary.

During FY 1987-1988, research efforts on equipment qualification procedures will be continued where necessary to resolve issues identified in the ongoing licensing review of qualification data submitted by vendors.

6.7.5.3 Aging Research

The comprehensive plant aging research program (see Section 6.1.5.2) will be directed toward coordination of all NRC aging research programs and will include the identification of potential new research areas, designation of priorities, surveys of aging research being conducted by domestic and foreign organizations, and the coordination of the dissemination of research results. Limited aging methodology and generic aging studies will be carried out as part of this program, which will continue throughout the FY 1984-1988 period.

Following are specific aging efforts that are directly related to the electrical equipment qualification program:

In order to ensure that the environmental and seismic qualification tests of electrical equipment account for possible multiyear age degradation, it is necessary to use accelerated aging techniques prior to LOCA simulation exposures. Polymers, plastics, and other organic materials can usually be aged by employing a short exposure at an elevated temperature determined by the Arrhenius rule. The extent of degradation and influence of different sequences of environmental exposure has been found, in many materials, to be unique. Thus, the work on synergism, dose rate and aging with EPR, polyolefin and polyethylene cable insulation, and jacketing materials, which was performed in FY 1981, will be continued in FY 1982-1984 to obtain a better understanding of the behavior of organics in aging. The aging behavior of organics used in gaskets and seals will be studied beginning in FY 1982 and will be completed in FY 1984.

It is anticipated that components such as transmitters and cables will have been removed from operating nuclear power plants such as Beznau and decommissioned plants such as Indian Point 1 and will be provided to NRC for examination and testing in FY 1983-1984. These components will have aged in use over an 8-to-10-year period. Current aging theories such as the Arrhenius Theory will be evaluated against the results from this study, and aging criteria will be either confirmed or developed.

A systematic overall scoping study of the effects of plant age in degrading core performance of nuclear plant materials, equipment, systems, and structures important to safety are to be initiated in FY 1982. The study will identify those items that are subject to age degradation and the cause. Follow-on studies of specific questions related to materials, components, systems, and structures will be pursued and coordinated under the comprehensive aging program. The detailed follow-on studies of aging degradation to electrical equipment and its impact on safety will be continued at least through FY 1988. The detailed studies will consider questions concerning the influence of equipment maintenance on minimizing the adverse effects on aging, inservice measurement of aging degradation, and replacement part schedules.

6.7.5.4 Sequence Testing (Single Prototype)

Results from the qualification testing of solenoid valves, switches, and other components studied in the equipment qualification research tests will be evaluated in FY 1984 relative to the requirement for including seismic and vibration testing in the equipment qualification test sequence on a single prototype in a LOCA qualification test. Recommended revisions to standards and regulatory guides will be provided if needed. The need for preaging prior to seismic testing for equipment in a mild environment will also be assessed in FY 1984 and necessary changes in requirements identified.

6.7.5.5 Accident Source Term and Equipment Dose Rates

The effect of source-term changes recommended from the degraded core studies will be evaluated in FY 1984 as to their effect on electrical equipment radiation dose rates during a LOCA, and revisions to Regulatory Guide 1.89 will be made. The results of environmental measurements in operating nuclear plants and generic dose rate calculations for electrical equipment will also be factored into the guide revision.

The results of radiation exposure of materials and equipment to beta, gamma, or beta plus gamma radiation will be evaluated in FY 1985 and used to develop improved requirements for radiation simulation in equipment qualification.

6.7.5.6 Equipment Qualification for Survival in a Hydrogen Burn

Testing methodology research will be initiated in FY 1984 to evaluate qualification techniques for electrical equipment qualification for survival in a hydrogen burn. In FY 1985, a regulatory guide will be developed for electrical equipment qualification for a hydrogen burn, and changes or implementation in national standards will be developed. The requirements for qualification testing of electrical equipment for survival from a hydrogen burn will be tested and will be verified in FY 1986. Issues identified in the implementation of qualification requirements for equipment survival from a hydrogen burn will be assessed during FY 1987-1988.

6.8 Fire Protection

6.8.1 Issue

The existing fire protection rule, § 50.48 and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50, was published in the Federal Register on November 19, 1980, and became effective on February 17, 1981. Questions were immediately raised by several licensees on some of the technical issues addressed in the rule.

One of the specific technical issues in Appendix R that was questioned deals with protection to be provided for at least one redundant train of safe shutdown equipment. One means of such protection included in Appendix R is a 20-foot horizontal distance separation of the redundant trains from each other with no intervening combustibles. This separation distance is being tested under a variety of conditions typically found in operating nuclear power plants.

6.8.2 Research Program Objective

The fire protection research program has as its objectives:

1. The development of data and a technical basis for specific licensing decisions,

2. The evaluation of testing procedures, and

3. The development of criteria for protection of at least one train of redundant safe shutdown equipment.

6.8.3 Relationship to Other Programs

EPRI is supporting fire protection research at Factory Mutual Research Corporation (FM). NRC contractor and FM personnel are being encouraged to maintain direct contact with each other and have agreed to an open exchange of technical information. NRC is also maintaining frequent and direct contact with EPRI and members of the Edison Electric Institute's (EEI) Fire Protection Committee.

6.8.4 Background and Status

The following activities are included in the FY 1982-1983 fire protection program:

1. Tests are being be conducted to (a) determine the adequacy of a 20-foot separation of redundant safety trains as provided for in Appendix R to 10 CFR Part 50 to protect at least one train from damage by a single fire and (b) develop procedures and methodology for assessing potential for fire propagation across spatial distances within a fire area.

2. Questions were raised at the conclusion of the Browns Ferry replication test as to the results of changes of fire size and location using the same test assembly. A series of separate effects tests will be conducted to obtain these results, which can then be compared with the results for the replication test.

3. Tests will be initiated to evaluate the susceptibility of components within electrical cabinets to fire damage and to test the effectiveness of detectors and halon suppression systems installed within these cabinets.

4. A program will be started to evaluate the susceptibility of nuclear power plant control rooms to fire damage by comparing layout, hazards, and protection provided with data on actual fires in similar types of control rooms such as those in fossil fuel or hydroelectric generating stations, petrochemical plants, petroleum refineries, and steel mills.

5. Tests using water spray, water sprinklers, and CO_2 suppressants on deep-seated cable tray fires will be completed and the results compared with those obtained from similar halon tests already completed.

6. The test program to evaluate the effects of material and pressure on performance of different penetration fire stop designs will be completed.

6.8.5 Research Program Plan

Efforts expended in fire protection research may decline in the future. Those items described above will be completed in about FY 1984-1985. Additional effort will be required after that to resolve issues that grow out of the present research program or out of the licensing process.

Expected accomplishments for the fire protection research program are:

In FY 1984, evaluation of the susceptibility of control rooms to fire damage and of the need for additional, or special, fire protection measures for control rooms will continue. Also continuing will be the evaluation of the susceptibility of components within electrical cabinets to fire damage and the effectiveness of detectors and halon fire suppression systems installed within the cabinets to limit fire damage. In FY 1985, the above evaluation will be completed.

In FY 1986-1988, the results of the entire research program will be used in future revisions of fire protection regulations. The results will also be used to support other aspects of the licensing process. Although reduced in size, a continuing fire protection research program will be required to investigate issues, as yet unidentified, that are expected to arise out of presently identified research efforts or from future licensing or operating experience.

6.9 Fuel Cycle Facility Safety

6.9.1 Issue

The NRC must have the ability to evaluate licensee proposals to store spent LWR fuel in air for long periods of time since it is expected that NRC will be receiving such proposals. In particular, NRC must be assured that sufficient information is available for evaluating these facilities with respect to the effect of temperature on fuel cladding integrity and on fuel oxidation. Cladding rupture has a high potential for gross contamination of the storage facility with a corresponding increase in occupational exposure and difficulty in decommissioning.

To determine the capabilities of safety systems in fuel cycle and materials facilities to prevent or mitigate the consequences of accidents, data and models are needed to support realistic consequence assessment methods. The NRC staff must be able to (1) define the nature of major credible accidents that could lead to the release of significant quantities of radioactive material to the environment, (2) evaluate the effect of credible accidents on facility operations and related physical conditions, (3) assess the response of facility safety systems during major credible accidents, and (4) specify the resulting source terms for the release of radioactive material into the environment. Also, the current approach used in licensing fuel cycle and materials facilities is to evaluate the performance requirements of facility structures and safety systems based on conservative assumptions regarding accident and accident effluent characteristics so as to minimize potential releases of radioactive materials from postulated accidents. Development of experimentally validated realistic methods for analyzing facility response and radionuclide source terms resulting from major accidents at fuel cycle and materials facilities is needed to improve the NRC's ability to justify regulatory design and operating requirements and standardize the nature and scope of the licensing staff safety reviews.

The NRC must be assured that licensees using special nuclear material employ adequate procedures to avoid accidental conditions of criticality. These procedures, of necessity, must include calculational methods for criticality safety assessment. A system of codes has been developed for performing standardized computer analyses for licensing evaluation. However, various aspects of this system need updating to make it a more effective licensing to l. Further, there are still special circumstances in which it must be demonstrated that calculational procedures for criticality safety assessment are valid by comparison with experimental data as in the cases of fuel arrangements with neutron poisons and low-density moderation (water sprays or fogs).

6.9.2 Research Program Objective

The objective of the spent fuel storage research program is to collect the necessary data to confirm or establish spent fuel dry storage licensing positions relative to (1) the long-term, low-temperature (less than 250°C) performance behavior of LWR spent fuel rods in dry storage and (2) the radioactive contamination potential of crud from cladding for the dry storage cycle.

The specific objectives of the fuel cycle facility accident analysis research program are to (1) develop the necessary data on accident initiation and sequences and (2) provide methods for analyzing the facility response required to support assessment of specific fuel cycle and materials 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.

The objective of the criticality research program is threefold: (1) updating of systems of computer codes for standardized criticality safety analyses of nuclear fuel shipping containers and other equipment associated with the fuel cycle; (2) procurement of experimental criticality data for use as confirmatory benchmarks for calculations of criticality, and (3) development and updating of regulatory guides based on NRC research programs and industry standards approved by the American National Standards Institute (ANSI) for use in the regulatory process.

6.9.3 Relationship to Other Programs

Many domestic and foreign programs exist in the area of fuel cycle technology. Where appropriate programs exist that complement or support the NRC research programs, coordinated or cooperative efforts have been or will be established.

DOE, TVA, and EPRI have ongoing research programs in the area of dry storage of LWR spent fuel. The NRC program has been coordinated with these organizations to ensure that duplication of effort is minimized. ANSI/ANS is also developing standards based on the EPRI and DOE research efforts.

The accident analysis program is closely related to the fuel cycle risk assessment research program. Under the risk assessment program, probabilistic risk assessment methods will be developed to supplement existing licensing safety analysis tools for elements of the nuclear fuel cycle other than the reactor. It is anticipated that the consequences assessment techniques being developed in the safety research program will be used as part of the probabilistic risk assessment method. The risk assessment program will also provide perspectives on the relative risks and uncertainties associated with all elements of the nuclear fuel cycle. These perspectives will be used as the basis for the safety research program and could lead to additional safety research. The close relationship between the research programs will continue as both safety and risk assessment methods are developed for materials facilities and advanced fuel cycle facilities.

The NRC has been participating in a group of experts on air cleaning in accident situations. This group was formed under the CSNI (Committee for the Safety of Nuclear Installations) of the OECD/NEA (Organization for Economic Cooperation and Development/Nuclear Energy Agency). Part of the group's work is to collect and evaluate information on general capabilities for investigating the behavior of air cleaning system components under accident conditions and on source terms for various types of accidents at fuel cycle facilities. The information obtained will be compared to that developed as part of the safety research program.

6.9.4 Background and Status

Very little research has been done in this country in the area of dry storage of spent fuel. The Canadians, English, and Germans have developed dry storage concepts and have conducted related research. Experimental work will be started by NRC in FY 1982 to study the effects of storing spent fuel for a long time at low temperatures (less than 250°C) in a dry environment. This work will also include the study of behavior of crud during the storage. After the initial characterization of the fuel in FY 1982, the material will be placed into the storage environment and examined annually.

In FY 1982-1983, the development of a data base and analytical models for assessing radioactive releases from major accidents in LWR fuel cycle facilities will continue. The analysis methods discussed here and in future sections generally refer to sophisticated computer programs, but simplified models will be available where appropriate. The analysis methods and experimental data consist of two types: (1) that in close proximity to the accident site (near field) and (2) that further removed (far field), for example, the ventilation system.

Accident scenarios and parameters will be defined for selected fuel cycle facilities. These include mixed-oxide fuel fabrication facilities, away-from-reactor spent fuel storage facilities, high-level waste solidification facilities, uranium hexafluoride production facilities, and the proposed ORNL Demonstration Reprocessing Plant for the advanced fuel reprocessing experiment. The accidents being addressed include fires, explosions, and tornadoes. Source-term models will be developed for these accidents that specify the characteristics of accident-generated aerosols. A family of computer codes that describe the material transport of these aerosols throughout the facility's ventilation system and the characteristics of any radioactive materials released to the environment will be completed. Experimental and analytical work will be initiated to improve these preliminary source-term models and material transport codes. Improvements will include providing more detailed characteristics of combustion products and failed compartments, the mitigating feature of engineered safety systems, and improved fire compartment and filter plugging models.

Work will be initiated to expand the research program to include accidents at materials facilities such as radiopharmaceutical manufacturing facilities and facilities manufacturing sealed sources. The initial effort will involve the identification of facility features and operating parameters important for accident evaluation, the specification of accident scenarios that have the potential for a significant release of radioactive material to the environment, and preliminary descriptions of the near-field accident source terms.

The criticality safety program has been in existence virtually since the formation of the NRC. During FY 1982, benchmark criticality data will be obtained for single fuel assemblies of uranium-enriched fuel rods in water containing varying concentrations of soluble poisons. Because of funding limitations, no work is planned for FY 1983-1984.

6.9.5 Research Program Plan

6.9.5.1 Spent Fuel Storage Studies

In FY 1984-1988, the investigation of the effects of storing spent LWR fuel in a dry environment will continue to completion.

During FY 1984, the program will continue research started in FY 1982. Spent fuel rods will be removed from the storage environment and examined for degradation. These rods will include both PWR and BWR types and intact and defected rods. Half of the rods will be stored in unlimited air and the other half in helium. These rods will be visually and nondestructively gamma scanned. One BWR fuel rod will undergo destructive examination after 2 years of storage to determine cladding effects, microhardness, hydrogen and fission gas content, and stress-corrosion-cracking effects. The remaining rods will be placed back into the controlled storage environment. Crud and corrosion deposits on the rods will be examined and analyzed. This study will focus on visual features, spallation, and content with respect to reference crud characteristics. A crud transportation and dryout crud behavior report will be issued along with interim test and evaluation reports.

Also in FY 1984, an analysis will be conducted to determine the effects of placing a waterlogged LWR spent fuel element into a dry storage environment. This analysis will be an evaluation of existing experience to assess if any potential licensing questions exist. Proposed follow-on work, either analytical or experimental, will be identified if required. A final report will be issued.

Spent fuel rods will again be removed in FY 1985 for visual and nondestructive examinations. The same analysis of crud and corrosion products will occur. No destructive tests will be conducted. The rods will be returned to the storage environment after completion of the examinations. A rod destruction evaluation report along with other interim test and evaluation reports will be issued.

In FY 1986, spent fuel rods will be removed for visual and nondestructive examinations. The same analysis of crud and corrosion products as was conducted in FY 1984 will be performed. The rods will be returned to the storage environment after completion of the examinations. No destructive tests will be conducted. Interim test and evaluation reports will be issued.

Four rods will be removed for visual and nondestructive examinations in FY 1987. The same analysis of crud and corrosion products as was conducted in FY 1984 will be performed. Two fuel rods, one PWR and one BWR, will undergo destructive tests in the same manner as described in FY 1984.

In FY 1988, the remaining four rods will be removed from storage for visual and nondestructive examinations. The same analysis of crud and corrosion products as was conducted in FY 1984 will be performed. Additionally, segments of fuel rods will be prepared for detailed crud analysis. All crud and corrosion products will be physically removed from the segments and analyzed. The results will be compared to the initial samples obtained in FY 1982. Three spent fuel rods will undergo destructive examination in the same manner as described in FY 1984. Final project reports will be issued.

6.9.5.2 Accident Analysis for Fuel Cycle Facilities

In FY 1984-1988, the development of accident analysis methods for LWR fuel cycle facilities will continue to completion, and work will be initiated to develop similar methods for other facilities. It is anticipated that adaptation of these analysis techniques to materials facilities and alternative fuel cycle facilities will also be completed within this time period. Adaptation of these methods for evaluating the consequences of accidents occurring in LWR balance-of-plant will be continued in this time period with completion of the effort occurring beyond FY 1988.

During FY 1984, the development of computer codes to describe the gaseous flow and material transport throughout a fuel cycle facility's ventilation system resulting from criticality accidents and equipment failures will be initiated. Developing analytical models and conducting experiments to support the development of these computer codes will also be undertaken. Improvements in the existing fire, explosion, and tornado computer codes will be developed. Improvements in the near-field source-term models for explosions and tornadoes will also be developed with experiments being performed to support this development. The development of near-field source-term models for criticality accidents also will be initiated.

Investigations will be initiated in FY 1984 to determine the modifications needed in the ventilation system computer codes and near-field accident source-term models so that the analysis techniques are applicable to accident evaluation in materials facilities. Scenarios will be defined for accidents that have the potential for significant releases of radioactive material to the environment. Experimental work to study the characteristics of accident-generated aerosols that may be present in these facilities will also be initiated as needed.

In FY 1985, the ventilation system computer codes for criticality accidents and equipment failures for fuel cycle facilities will be completed. Final verification and documentation for the improved fire, explosion, and tornado codes will also be completed. The development of near-field source-term models for criticality accidents will be completed. The results of these efforts will be incorporated in a user's handbook (accident analysis handbook).

The identification and description of material facility features and operations important for accident analysis will be completed. The adaptation of near-field accident source-term models will be initiated. The experimental program to investigate the characteristics of accident-generated aerosols unique to these facilities will be continued. Modifications will be made to the family of ventilation computer codes to accommodate unique characteristics of these facilities. The descriptions of a representative materials facility will be developed for use in illustrating the analysis methods.

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. For planning purposes, it is assumed that a need will exist and that sufficient information will be available to start a study similar to that described for materials facilities. The initial effort will be to investigate the nature of the facility and processing operations to determine the extent of modifications to the ventilation system computer codes and accident source-term models.

During FY 1986, final documentation will be completed for fuel cycle facilities to be included in the accident analysis handbook and the user's manuals for the improved fire, explosion, and tornado computer codes and the criticality and equipment failure computer codes. The experimental program and the development of near-field source-term models for accident-generated aerosols at materials facilities will be completed. The procedures for specifying the near-field accident source terms will be developed and incorporated into the accident analysis handbook. Examples that illustrate the analysis procedure will be developed. Final documentation and verification of the modified ventilation system computer codes will be accomplished.

In a manner similar to that described for materials facilities, additional experimental and analytical work will be conducted in FY 1986 for alternative fuel cycle facilities to supplement the accident analysis method developed under the fuel cycle facility research program. Facility features and processes unique to alternative fuel cycle facilities may result in accidents that generate aerosols with characteristics requiring additional investigation. Scenarios will be defined for accidents that have the potential for significant releases of radioactive material to the environment. Modifications to the near-field accident source terms and the ventilation system computer codes may also be necessary. Modification of the description of the representative fuel cycle facility may also be required.

Work will be initiated to develop methods for realistic assessment of accident consequences in portions of nuclear power plants not directly affecting reactor safety (e.g., the radwaste system). The initial effort will be to describe facility features and processing operations and to identify those aspects important for accident analysis.

In FY 1987, experiments to characterize the unique features of accident-generated aerosols that may be present in alternative fuel cycle facilities will be completed. The adaptation of analytical models to describe the nature of near-field accident source terms will also be completed. Verification of computer code modifications will be performed. The development of material for inclusion in the accident analysis handbook will be initiated. Examples that illustrate the analysis procedure using the representative facility description will be developed.

The adequacy of existing experimental data and analytical models will be assessed to determine the need for and extent of additional investigations. Scenarios will be developed for accidents that have the potential for significant releases of radioactive material to the environment. Plans that specify the experimental and analytical work needed to provide the information required for modification of the accident analysis methods will be formulated.

Final documentation for the modified computer codes for alternative fuel cycle facilities will be completed in FY 1988. The description of the accident analysis procedure and the illustrative examples to be included in the accident analysis handbook will be completed. Final reports on the experimental and analytical programs will be issued.

During FY 1988, experimental programs will be initiated to provide the necessary data to characterize the accident-generated aerosols expected to be present in the LWR balance-of-plant. There may be a need to develop analytical models to describe near-field accident source terms. Experiments to investigate the performance of accident-mitigating safety features will also be initiated. Modifications to the ventilation system computer codes will be undertaken as necessary. Descriptions of representative facilities will also be developed.

6.9.5.3 Criticality

The program for conducting benchmark criticality experiments on fuel arrangements to validate codes and cross section data used in criticality calculations will continue throughout FY 1985-1986. Updating of computer programs for standardized criticality safety analyses of nuclear fuel shipping containers and other equipment associated with the fuel cycle will be accomplished during FY 1987-1988.

In FY 1985, calculational models will be verified, and any bias in calculational techniques due to the amount of fixed neutron poison present in a shipping or storage array will be validated.

During FY 1986, the calculation will be verified for the variation in k_{eff} with water density for an array of water-moderated fuel rods as the density of interspersed moderation decreases. Bulk-oxide criticality experiments, which were performed at Rocky Flats, will be analyzed to allow use of these data as analytical benchmarks in license applications and safety issues involving bulk-oxide systems having low moderation.

In FY 1987, the SCALE system for performing standardized computer analysis (criticality, shielding, or heat transfer) for licensing evaluation of nuclear systems will be updated. There will be an emphasis on criticality aspects such as (1) validation of criticality cross section libraries, (2) enhancement of KENO-V geometry capability, and (3) development and documentation of improved cross section processor.

The updating of the SCALE system will be completed in FY 1988 with emphasis on criticality aspects such as (1) development and documentation of criticality sequence to search for optimum concentration and (2) conversion of the existing SCALE system to the CDC computer system.

6.10 Decommissioning

6.10.1 Issue

Decommissioning nuclear facilities must be conducted to ensure that public and occupational exposures are as low as reasonably achievable (ALARA) and that contamination levels of the facilities and sites are reduced to acceptable levels for unrestricted use and subsequent license termination.

To establish decommissioning standards and to review licensee plans and applications, the NRC must have the ability to 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. It is also necessary to determine the degree to which siting, construction, design, and operating procedures described on initial applications will facilitate eventual decommissioning. The reliability of licensee cost estimates for decommissioning so that financial responsibility can be established and ensured must be evaluated. Very important also is the ability to evaluate the residual contamination following decommissioning to ensure that it meets existing standards for radiological safety.

6.10.2 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 studies and to improve the accuracy of the cost estimate and exposure models;

3. Determine how plants can be designed and operated to facilitate and standardize decommissioning;

4. Develop and verify analytical models to assess the costs, safety, and waste characterization set forth in future license applications;

5. Assess new approaches for improving the decommissioning process with respect to cost, safety, waste handling, and residual radioactivity levels; and

6. Assess reasonable state of the art of radioactivity measurements in establishing residual radioactivity levels at decommissioned sites and the impact of these levels on the cost and waste disposal needs of decommissioning.

6.10.3 Relationship to Other Programs

DOE has long-range programs under way (3-5 years) to provide a data base for evaluation and comparisons of decommissioning techniques that will allow the assessment of ALARA objectives in health and safety. Information developed by DOE will be incorporated in the NRC data base.

The International Atomic Energy Agency (IAEA) is developing a data base in the area of decommissioning. NRC and DOE are working with this organization for a mutual sharing of decommissioning experience. NEA has completed survey reports on decommissioning in the areas of decontamination and cutting techniques (for large equipment).

The Environmental Protection Agency (EPA) has a long-range program under way (2-5 years) for the development of acceptable residual radioactivity levels following decommissioning. NRC is working closely with EPA and is developing a data base for implementing acceptable residual radioactive levels.

The States, public utility commissions, and Federal agencies such as the Federal Energy Regulatory Commission have been involved with the financial assurance aspects of decommissioning. NRC's Office of State Programs has worked closely with these groups in developing financial guidance for licensees. Data from the TMI-2 cleanup are being used.

6.10.4 Background and Status

The NRC is currently developing detailed regulations and guides for decommissioning nuclear facilities that will establish acceptable procedures and methods for decommissioning nuclear facilities for unrestricted use within the framework of the ALARA concept. These regulations will be based in large measure on studies performed for RES that evaluate cost, safety, and effectiveness of techniques for decommissioning based on existing data. Decommissioning technology, safety, and cost reports are currently in progress for completion of a data base on various nuclear facilities and situations. In FY 1982, reports on UF_6 conversion plants, multiple reactor facilities, research and test reactors, postaccident decommissioning of reactors, and termination surveys will be completed. In FY 1983, reports on postaccident decommissioning of fuel cycle facilities and preliminary studies of facilitation of decommissioning for LWRs and fuel cycle facilities will be completed.

Actual nuclear plant decommissioning operations have provided relatively little data, although data from decontamination during operations cleanup and maintenance have proved valuable. 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. Contractors are currently developing an in-field decommissioning data base to complement and support the data base developed thus far. For FY 1982 and 1983, contractor experimental research projects include characterization of LWR contamination and radiation levels at decommissioned reactors; investigation of long-lived activation products in LWR materials; investigation of decontamination techniques as a precursor to decommissioning; investigation of available instrumentation and measurement techniques for performing radiation termination surveys; and collection of licensee decommissioning data to provide a base to check decommissioning technology, safety, and costs and to support ALARA analyses.

Work on a final environmental impact statement (FEIS), rule, and regulatory guides is currently in progress. The FEIS and proposed rule are scheduled for completion in FY 1983.

6.10.5 Research Program Plan

In FY 1984, earlier studies of the decommissioning technology, safety, and costs of LWRs will be updated. Also, the bibliography on decommissioning that was previously published will be updated. The ongoing research projects relating to the nature and location of contamination at facilities to be decommissioned will be completed. Investigations at facilities being decommissioned will be continued. Effective decommissioning rules and regulatory guidance will be issued.

During FY 1985, studies to investigate available measurement techniques and instrumentation to improve terminal radiation surveys will be completed.

Investigations at facilities being decommissioned will be continued. Preliminary termination surveys at actual sites using improved technology will be completed. Studies of the technology, safety, and costs of decommissioning on advanced reactor and advanced fuel fabrication plants will be made.

Termination surveys at actual sites and updates of studies of the technology, safety, and costs of decommissioning of fuel-cycle and non-fuel-cycle facilities will be completed in FY 1986. Research to facilitate and standardize the decommissioning of LWRs will also be completed. Investigations at facilities being decommissioned will continue.

Earlier studies on the technology, safety, and costs of decommissioning multiple reactor stations and of reactors that were involved in accidents will be updated in FY 1987. A study of the technology, safety, and costs of decommissioning a rare earth/thorium mill will be completed.

During FY 1988, research to standardize and facilitate the decommissioning of fuel cycle facilities will be completed. The previous studies of the technology, safety, and costs of decommissioning research/test reactors, of fuel cycle facilities that were involved in accidents, and of independent spent fuel storage installations will be updated. The decommissioning bibliography will again be updated. The investigations at facilities undergoing decommissioning will be concluded. The decommissioning program will be essentially completed except for any unexpected requirements for other decommissioning research.

6.11 Effluent Control and Chemical Systems

6.11.1 Issue

The NRC requires that nuclear power plants be designed to (1) ensure adequate safety under normal and postulated accident conditions, (2) suitably control radioactive materials in gaseous and liquid effluents, and (3) handle radioactive solid wastes.

The NRC policies and strategies for hydrogen control are being reevaluated as a result of current knowledge and its application to operating plants, which were designed on a different basis. At issue is the adequacy of protection against hydrogen burning and deflagrations and detonations immediately following a postulated LOCA during which large quantities of hydrogen may be generated and accumulated.

The fission product control program is developed to evaluate the effectiveness of engineered safety features (ESFs) under severe conditions. Under current regulatory assumptions, the design basis accident (DBA) LOCA source term is dominated by noble gases and iodine, primarily in elemental form. ESFs have been designed on the basis of a substantial iodine source term. Other environmental challenges imposed on the ESF have been selected from LOCA sequences. Since the iodine source term is consistent with a severely degraded core, aerosol loadings and other fission products are not included in design requirements for ESFs. The evaluation of impacts of revised realistic source terms on the design and effectiveness of ESFs for a spectrum of accident conditions is needed as a part of the information base for formulating policies and strategies to mitigate postulated severe-accident circumstances.

The Commission has adopted a policy on low-level radioactive waste to reduce the volume of waste for disposal. In addition, licensees are encouraged by increasing disposal costs to establish programs commensurate with good volume-reduction practices. There is a need to develop an up-to-date technical data base to be used in safety evaluations of onsite waste treatment systems (particularly volume reduction) and onsite waste storage facilities.

Chemical and nuclear processes in LWR coolant systems make changes in the coolant chemical compositions that have time-related effects on reactor components over the 40-year design lifetime of the facilities. Controlling cooling water compositions through chemical additions and demineralization has proved to be essential to maintaining primary system integrity. Knowledge of how cooling water composition affects both safety- and non-safety-related components and how these effects can be controlled is needed in upgrading safety-related systems.

6.11.2 Research Program Objective

The objective of this program is to provide to the NRC licensing staff information in the areas of hydrogen combustion prevention and mitigation, fission product control, onsite waste storage, and water chemistry control that can be used during the licensing process to ensure that the operation of nuclear facilities does not present undue risks to the public health and safety from releases of radioactive material. The program also provides guidance in the form of rules, regulatory guides, and NUREG reports.

Information from the hydrogen control studies will aid in formulating policies and strategies for the prevention or control of hydrogen combustion. The hydrogen combustion prevention and mitigation research program will assess current and proposed hydrogen control schemes under both accident and postaccident conditions and will aid both in developing rules and in formulating policies and strategies for the prevention or control of hydrogen combustion.

The objective of the fission product research program is (1) the development of information to support minimum ESFs required for fission product removal under severe-accident conditions; (2) quantification of the effectiveness of various engineered safety and mitigation features in reducing the potential fission product escape from containment; (3) an evaluation of the existing design features under expected aerosols and other fission product loadings; and (4) the development of simulated conditions and design features of ESFs for standardized facilities under these conditions. The research will also assist in providing guidelines for design and operating requirements and will provide a data bank on ESFs for use in future evaluations of standardized ESF systems.

The objectives of low-level onsite waste research are: (1) procurement of operating experience data evaluating performance of various volume-reduction methods and waste solidification techniques; (2) development of additional radiological safety guidance for onsite contingency storage capacity; and (3) development of operating and design criteria to be used in the licensing process.

The objective of the water chemistry program is to develop updated analytical techniques for the evaluation of licensee-proposed chemical adjustment processes for reactor coolants.

6.11.3 Relationship to Other Programs

The effluent control and chemical systems research programs are directly and indirectly related to a number of NRC, private industry, and foreign programs.

The hydrogen combustion preventive and mitigative schemes research program is closely coordinated with the hydrogen generation and control program, part of the severe-accident mitigation program, described in Section 4.4. This program is concerned with understanding the phenomena associated with hydrogen combustion, and the work includes hydrogen generation and transport calculations, deflagration and detonation model development, investigation of hydrogen deflagration and detonation limits in steam and air, and transition from deflagration to detonation. In addition, these programs are related to the following industry-sponsored programs:

1. EPRI has a program on hydrogen control that includes (a) testing of deliberate ignition devices (AECL - Whiteshell and Acurex); (b) experiments on hydrogen control methods, including water sprays and fogs (Factory Mutual Research, Acurex); and (c) demonstration of hydrogen combustion and management techniques on a large scale (Nevada Test Site).

2. Ice condenser owners group has sponsored a series of experiments on igniters for use in their plants.

3. BWR Mark III owners group is sponsoring a series of experiments at EPRI on the use of igniters in their plants.

The fission product control program is related to a broad research program on fission product release and transport (described in Section 4.3) to provide data and models to predict the radiological source terms for accident consequence assessment, site evaluation, and the radiological and thermal loads imposed by released fission products on ESFs and other safety-related plant equipment.

The fission product control program is closely coordinated with the fuel behavior program and research conducted by the FRG, EPRI, and GE: projects related to core-melt/concrete interactions (CORCON, KAVERN, and WECHSL codes development, and experiments such as Sandia BETA projects, Sandia melt/concrete experiments, and MARKIVEN experiments), steam explosions (Sandia FITS experiments), core-melt accident modeling (MARCH, TRAP-MELT, CONTAIN, CORRAL, QUICK, and other codes under development; EPRI core-slumping model), and fission product behavior (ORNL fission product release experiments, SASCHA experiments, iodine release experiments, and aerosol test in steam), EPRI large-scale experiments and model development on fission product and aerosol attenuation in water pools, fission product chemical form and release rates from high-temperature fuel, and GE suppression pool effectiveness.

EPRI has several research projects in the area of advanced onsite low-level radioactive waste treatment technology, including a program on radwaste incinerator experience. Also, DOE is funding demonstration efforts with incinerators to further encourage use of incineration for fuel cycle waste. Technical liaison between EPRI and NRC has been established and will be maintained.

EPRI in cooperation with the Steam Generator Owners Group is funding LWR water chemistry studies that relate coolant chemistry to steam generator corrosion damage. Water chemistry systems will be investigated for corrosion product transport. Chemical systems will be investigated for corrosion product transport; chemical species hideout, shutdown, and startup effects; and demineralizer operational effects. Argonne National Laboratory is investigating environmentally assisted cracking of coolant system components in LWRs. Technical liaison between EPRI and NRC has been established and will be maintained.

6.11.4 Background and Status

New research on control of hydrogen combustion began in FY 1980 under the hydrogen behavior program. In FY 1982, this program assumed those tasks dealing with hydrogen combustion mitigation and prevention. The program for FY 1982 includes the following: initiation and completion of laboratory-scale tests on foams' effect on hydrogen combustion; initiation of intermediate-scale tests on the effect of foams and of CO_2 on hydrogen combustion in the Variable Geometry Experimental System (VGES) facility; design and construction of a steam/H_2 jet facility to investigate flaring hydrogen from the primary system high point vent; and continuation of modeling on fogs, CO_2, and deliberate ignition. The planned FY 1983 program includes expansion of hydrogen control models to include foams; continuation of deliberate flaring test in steam/H_2 jet facility; performance of feasibility studies on new hydrogen control methods; and performance of intermediate-scale test on fogs in the VGES and Fully Instrumented Test Series (FITS) facilities.

The fission product control program started in FY 1982 and includes the identification and modeling of accident sequences that can affect ESF-system performance and the modeling and experimental investigation of system effectiveness under severe-accident conditions.

The FY 1982 activities focus on the assembly of a data base for (1) ESF-system definition, (2) accident sequence selection, and (3) modeling. Using available information, design philosophy and criteria for ESF systems will be compiled and catalogued, and candidate accident sequences will be selected for evaluation of system performance. Emphasis will be placed on those sequences that represent the severe-accident conditions that could conceivably result in the impairment of ESF-system effectiveness. Available computer codes and mathematical models will be catalogued to identify their predictive capabilities (including any limitations), key assumptions, input data required, and their use in the analysis of ESF-system performance. Work will include the cataloguing of models describing reactor core fission product inventory; thermal-hydraulic response, including fission product transport through and retention in primary coolant systems; fission product release from the coolant circuit and subsequent transport and deposition within the containment; the source terms that can challenge ESF systems (fission products, inert aerosols, hydrogen); and ESF-system performance under severe-accident conditions. Information concerning experimental investigations of fission product release and transport and the effectiveness of ESF systems in mitigating release will be compiled and catalogued. Facilities that can be used in experiments to verify ESF-system effectiveness will be identified.

The FY 1983 planned activities will concentrate on analytical modeling of accident sequences, identification of the range of aerosol concentrations expected to result from accident sequences along with the ability to assess effects of low and high concentrations, and identification and evaluation of methods currently being used to evaluate ESF-system performance. For the ESF-system design the assessment of the present state of the art will be completed.

6.11.5 Research Program Plan

6.11.5.1 Hydrogen Combustion Control

The planned long-range continuation of the hydrogen combustion control program includes:

FY 1984	Complete all tests on fogs, foams, CO_2, N_2, and halons in intermediate-scale facilities; continue experimental program to evaluate potential or new schemes in intermediate-scale facilities (VGES and FITS); and initiate tests in complex environment on a large-scale (6 ft x 6 ft x 100 ft FLAME facility) to look at such effects as multiple ignitions and inhomogeneous concentrations on mitigation schemes.
FY 1985	Complete theoretical and experimental investigation of all remaining hydrogen control schemes.
FY 1986	Select most promising hydrogen control mitigation scheme, and perform integral tests.
FY 1987	Initiate large-scale tests.
FY 1988	Complete large-scale tests.

6.11.5.2 Fission Product Control

While specific fission product control research efforts cannot be identified at this time, several areas such as stratification of gases in compartments, the capture of particles in suppression pools, and the capture of particles in the ice condenser containment design appear to merit consideration. Specific plans will be evaluated based on the broad research efforts to identify revised source terms for severe accidents and the experimental and modeling programs conducted by other organizations in FY 1982-1983.

Independently of the above, other activities planned in FY 1984-1988 are:

FY 1984	Continue modeling efforts to provide the tools needed for evaluating ESF-system performance in areas where present models are inadequate or nonexistent; and start experimental work to confirm the reliability of models.
FY 1985-1986	Evaluate ESF design and criteria for the standardized LWR plants; continue modeling to include experimental data; and continue experimental work to confirm the reliability of models.

By FY 1987, the main effort will be concentrated on documentation of the ESF-system designs in the form of computer code(s) to provide tools for comprehensive evaluation of existing ESF systems of new and standardized power plants.

6.11.5.3 Onsite Low-Level Waste

The onsite low-level waste program begins in FY 1984 with the collection of operating experience data for evaluating performance of various volume-reduction methods and waste solidification techniques. Consideration will be given to related research on low-level waste disposal sponsored by the Division of Waste Management of NMSS.

Prior to FY 1984, research work completed by EPRI on low-level radioactive waste incineration experience will be evaluated. If this work, together with pertinent studies sponsored by NMSS, proves to be insufficient for licensing needs, a program on survey and evaluation of onsite low-level-waste incineration systems will be initiated in FY 1984 to permit timely development of adequate acceptance criteria for the safety evaluation of these systems.

The program on survey and evaluation of onsite low-level-waste incineration systems will be completed in FY 1985.

Other volume-reduction systems as well as waste solidification techniques will be considered for reevaluation and updating in the light of current operating experience during FY 1986-1988 with emphasis on unit operations such as filtration and evaporation and on use of the urea formaldehyde process for solidification. During this same period, a program for developing additional radiological safety guidance for onsite contingency storage capacity will also be undertaken, if warranted by evaluation of prior experience in this area.

6.11.5.4 Water Chemistry

For the water chemistry program, operating reactor steam generator systems will be selected from a cross section of steam generator designs that have had a monitored operating program to maintain desired water chemistry compositions. This information on steam generator operating histories and plant water chemistry adjustment records will be used to evaluate the effect of both normal and off-normal operating conditions on water coolant chemical compositions. A literature review will be performed to identify advanced technologies used on primary and secondary coolant systems for removing particulates by new filter designs and soluble constituents using temperature-resistant inorganic resins. The operating data and literature review will be used to identify the need to develop analytical techniques for NRC to evaluate licensee chemical adjustment processes as well as corrective actions taken by licensees.

Analysis of FY 1984 program results will provide the basis for further program development in FY 1985-1988. A program will be coordinated with industry, particularly if experiments or design tests are needed.

7. FACILITY OPERATIONS AND SAFEGUARDS

Since the TMI-2 accident, greater emphasis has been placed on facility operational safety. The increasing number of operating nuclear power plants results in an increasing potential for human error and hardware-type failures. Consequently, there is an accelerated need to obtain independently generated safety information in order to provide to the NRC staff a technical basis for decisions and regulations involving operating facilities. The end objective of this program is to have fewer and less severe accidents that might affect public health and safety.

This division unit is subdivided to address the following issues:

1. Human Engineering - this effort is intended to provide the technical basis to support current and anticipated regulatory needs in the application of human engineering to nuclear facilities. Data to support the implementation of improvements in the operator-machine interface are especially needed. This program also includes work on human reliability, personnel qualifications, procedures, and design and evaluation criteria.

2. Plant Instruments and Controls - efforts in this area focus on the evaluation of individual components and systems, as well as interaction of plant control, protection, and other instrumentation and electrical systems so that these interactions can be better understood and evaluated.

3. Occupational Radiation Protection - the major objective is to ensure that occupational radiation exposures are limited to accepted levels and maintained as low as is reasonably achievable (ALARA). This includes the development of rules and guidance related to occupational radiation protection and is based in part on staff analysis of research on radiation-induced health effects and protection techniques.

4. Emergency Preparedness - data and analyses are needed to support the development and evaluation of appropriate protection actions that may be taken during an emergency at nuclear facilities.

5. Safeguards - the objectives of the safeguards program are to improve the protective systems employed at nuclear facilities to prevent radiological sabotage and to improve the systems for control, accounting, and protection of special nuclear material that could constitute a threat to the health and safety of the public or national defense.

6. Quality Assurance - the objective in the quality assurance area is to improve the regulatory criteria available for the establishment and implementation of quality assurance program activities at nuclear facilities.

The individual programs are described in more detail in the following sections.

7.1 Human Engineering

7.1.1 Issue

Human performance is now generally recognized to be a significant and perhaps dominant determinant of the risk resulting from operating nuclear facilities. Considerable progress has been made in identifying and addressing regulatory issues associated with human factors. Among those requiring long-term resolution are the following:

1. What are the roles of the operating crew and support personnel, and what should they be to ensure safe operation of nuclear systems?

2. What are the corresponding educational, training, and certification requirements?

3. How does management affect the safety of plant operations? What can be done if improvements are deemed necessary?

4. What should be the nature and extent of regulatory involvement in the development, review, approval, and implementation of plant procedures?

5. To what extent and with what urgency must recognized human engineering deficiencies in equipment design be corrected?

6. To what extent and at what rate should computer technology and automation be introduced into reactor operations?

7. Do proposed modifications really improve the performance and safety of the total system? How much improvement and at what cost?

8. What review criteria and practices will help minimize the effects on public risk of human error in system design and construction?

7.1.2 Research Program Objective

This element's principal objectives are to provide the technical basis and the resulting regulatory standards necessary to clarify and resolve these issues. The research provides experimental data and analysis that improves NRC's basic understanding of the impacts that humans have on nuclear safety and of the factors that affect human performance.

The products of this program include:

1. Analytical and empirical models of human performance,

2. Experimental data to help develop and validate these models,

3. Recommendations on implementing improved systems and procedures,

4. Data to validate regulatory requirements and criteria,

5. Methods, criteria, and standards for evaluating proposed designs,

6. Assessments of the technical feasibility of new concepts,

7. Quantitative estimates of risk-reduction potential, and

8. Literature and technology surveys.

This information supports the development of regulatory positions, the ultimate goal of which is to reduce the human contribution to risk to an acceptably low level.

7.1.3 Relationship to Other Programs

NRC's human engineering research is coordinated with the activities of other organizations (see Table 7-1) to ensure complete coverage of important topics without unnecessary duplication. Coordination occurs most often through exchanges of program plans and research results and through participation in working groups by NRC personnel. This approach will continue through the planning period. Beyond the routine coordination, particular efforts are likely to be needed in gathering data to support revised regulatory approaches, to validate regulatory criteria, to develop and use facilities for conducting human performance experiments, and to develop a repository of human performance data.

7.1.4 Background and Status

The relative level of effort for human engineering research over time including significant items influencing this level is shown in Figure 7.1. Prior to TMI-2, NRC research was limited to gathering data on operator response times for safety-related actions and to estimating human error rates for probabilistic risk assessments.

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

1. Establish a program to evaluate the safety effectiveness of disturbance-analysis systems, and

2. 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.

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

1. Improved control room instrumentation (operator-process communication and disturbance-analysis systems), and

2. Improvements in simulator capabilities.

Subsequent research requests and endorsements (see Tables 7-2 and 7-3) helped shape the current program. In addition, the reorganization of the Offices of

Table 7-1. NRC's Human Engineering Research and Standards Related to and Coordinated with Efforts of Other Organizations

Organization	Coordinated activity
NRC/RES	Long-Range Research Plan, Sections 7.1 through 7.4, and Sections 4.5 and 10.1.
NRC/NRR	Technical assistance to develop guidelines, criteria, and priorities.
Department of Energy	LWR safety research and development in response to Public Law 96-567 and LMFBR human factors research.
Electric Power Research Institute	Information exchange on computerized operator support systems; maintainability studies; control room design guides.
Institute of Nuclear Power Operations	Reactor personnel task analysis; utility and plant evaluation techniques; defining standards of excellence.
NSSS vendors and utilities	Research and development on computerized operator support systems, emergency procedures, operator training.
Halden Project	Research and development on computerized operator support systems (NRC is a signatory).
National Aeronautics and Space Administration	Research on effects of fatigue.
Institute of Electrical and Electronics Engineers	Human factors standards on control room design, human performance evaluation, and systems engineering.
American Nuclear Society	Standards on personnel education and training, simulators, and manual vs. automatic safety functions.
Instrument Society of America	Standards on qualifications of instrumentation and control technicians.
Committee on the Safety of Nuclear Installations	Human reliability reporting and data analysis.

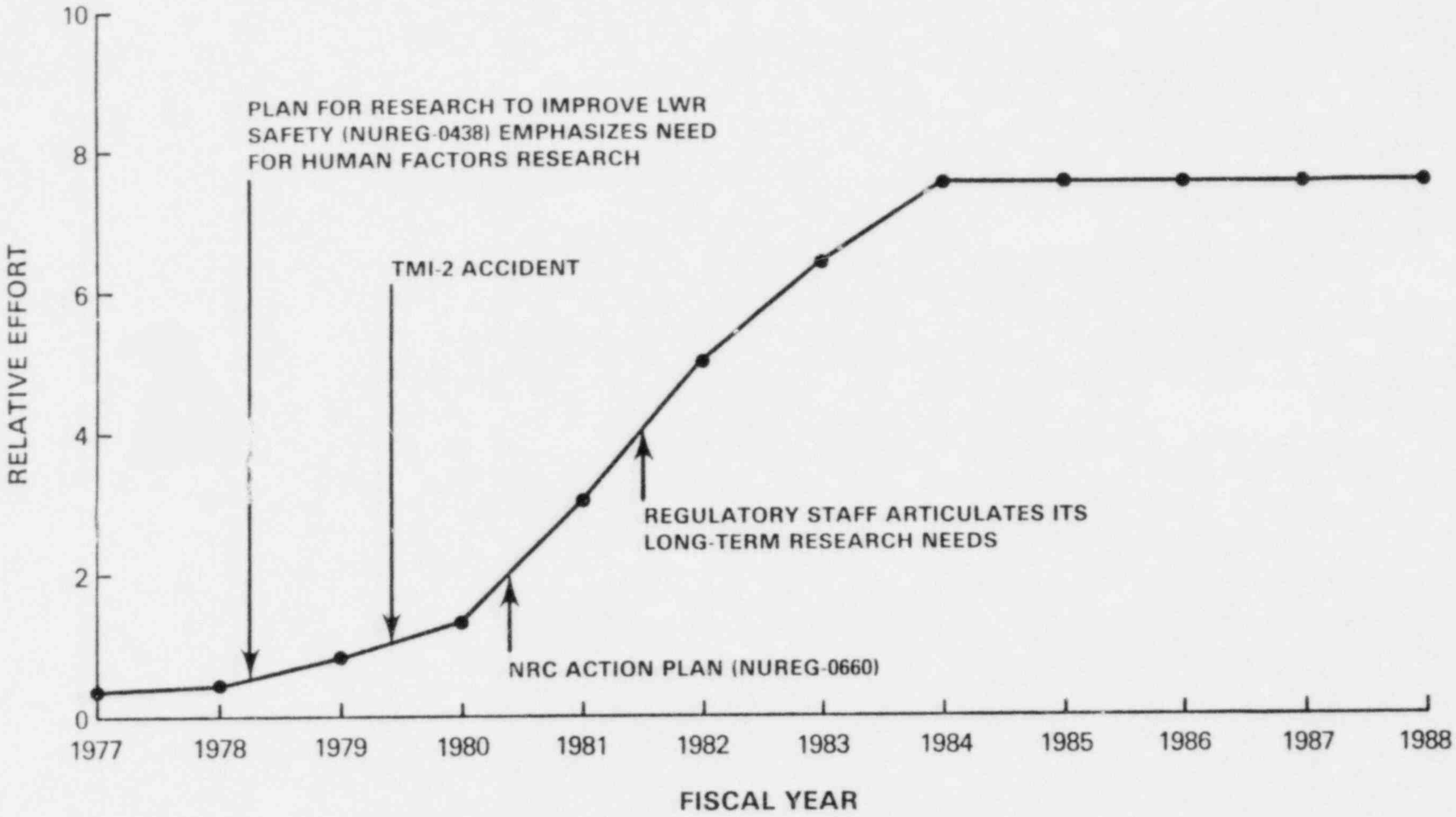

Figure 7.1 The historical perspective and projection on human factors research by NRC. Planning assumes that after the current growth period, in which near-term issues will reach at least interim resolution, research will maintain a constant level of effort to reflect a continuing commitment to human factors research. Effort includes human engineering and human reliability.

Table 7-2. Research Requests and Endorsements for Human Factors Research

Request or Endorsement Letter No.	User Office	Information Need
SD-78-2	SD(RES)	Operator response time for safety-related actions
RES-80-13	NRR	Operator's role in severe accidents
RES-80-15	NRR	Requirements for plant status monitoring; computerized display and diagnostic system requirements
NRR-80-7	NRR	Reactor operating crew task analysis
SD-80-2	SD(RES)	Assessment of simulator practices
RES-80-25	NRR	Human reliability models and data
RES-81-2	NRR	Data and design criteria for computerized aids
RES-81-6	NRR, IE	Long-term human factors program plan
NRR-81-2	NRR	Capabilities and uses of training simulators
NRR-81-5	NRR	Multiple requests (see Table 7-3)
RES-81-7	NMSS	Task analysis for ISFSI operators

Table 7-3. Information needs expressed by NRR's Division of Human Factors Safety in Research Requests NRR-81-2 and NRR-81-5 provide the foundation for long-term research planning. The information needs are listed in NRR's order of decreasing priority, as of December 1, 1981

1. Reactor operator task analysis
2. Plant procedures and their implementation
3. Validation of operator examinations
4. Validation of education and training requirements
5. Organization and management
6. Capabilities of training simulators
7. Effects of shiftwork and overtime
8. Evaluation of human factors engineering data
9. Validation of control room modifications
10. Plant maintenance
11. Effects of post-TMI requirements on operators
12. Research dependent on advanced simulators
13. Automatic plant operations
14. Task analyses for support personnel
15. Code applications to startup tests

Nuclear Regulatory Research and Standards Development consolidates human factors research into the Human Factors Branch and assists the timely incorporation of research results into regulatory guides, criteria, and regulations.

In its fiscal and planning guidance for FY 1983-1987, the Commission has designated the human factors program as a highest priority and indicated:

> "NRC will continue to study and improve its human factors acceptance criteria using data from appropriate disciplines and conducting its own research where necessary. A long-range integrated human factors program will be developed. NRC human factors efforts will focus on developing human factors guidelines for the following areas: current and future control room design (including integration of information on plant status, communications, and information retrieval as required by operators); plant procedures (normal, emergency, test, surveillance); the nuclear plant outside the control room; and waste management and other nonreactor fuel cycle activities. As part of the effort to improve safety, the NRC will develop methods for judging the 'competency' of a utility, its management, and personnel."

The current program has four related subelements:

1. Human factors engineering,
2. Licensee qualifications,
3. Plant procedures, and
4. Human reliability.

Together these subelements address the total system the NRC must regulate from a human factors perspective: the machine with its intended function and potential hazard to the public; the machine's operators with their inherent abilities and limitations; the procedures and equipment interfaces that link them together; the infrastructure of support personnel and organization; and the integral performance of the system that determines its impact on public risk. Prior to the FY 1984-1988 planning period, this program will have generated sufficient information to help reach at least interim resolution of current issues.

7.1.5 Research Program Plan

The plan for human engineering is directed toward the resolution of long-term issues related to human factors safety. Research and standards efforts to date have focused on the operators and the control rooms of light-water reactors, and these efforts are expected to continue well into the planning period. However, increasingly greater emphasis will be placed on personnel and systems outside the control room and on fuel cycle facilities other than light-water reactors. Figure 7.2 relates the planned availability of research products to our current assessment of anticipated regulatory needs.* Integral to our planning is continual assessment of the safety significance of human performance as derived from reviews of documented operating experience and risk analysis. These reviews help set priorities for research and standards activities.

*NRR must provide assistance in developing this timeline.

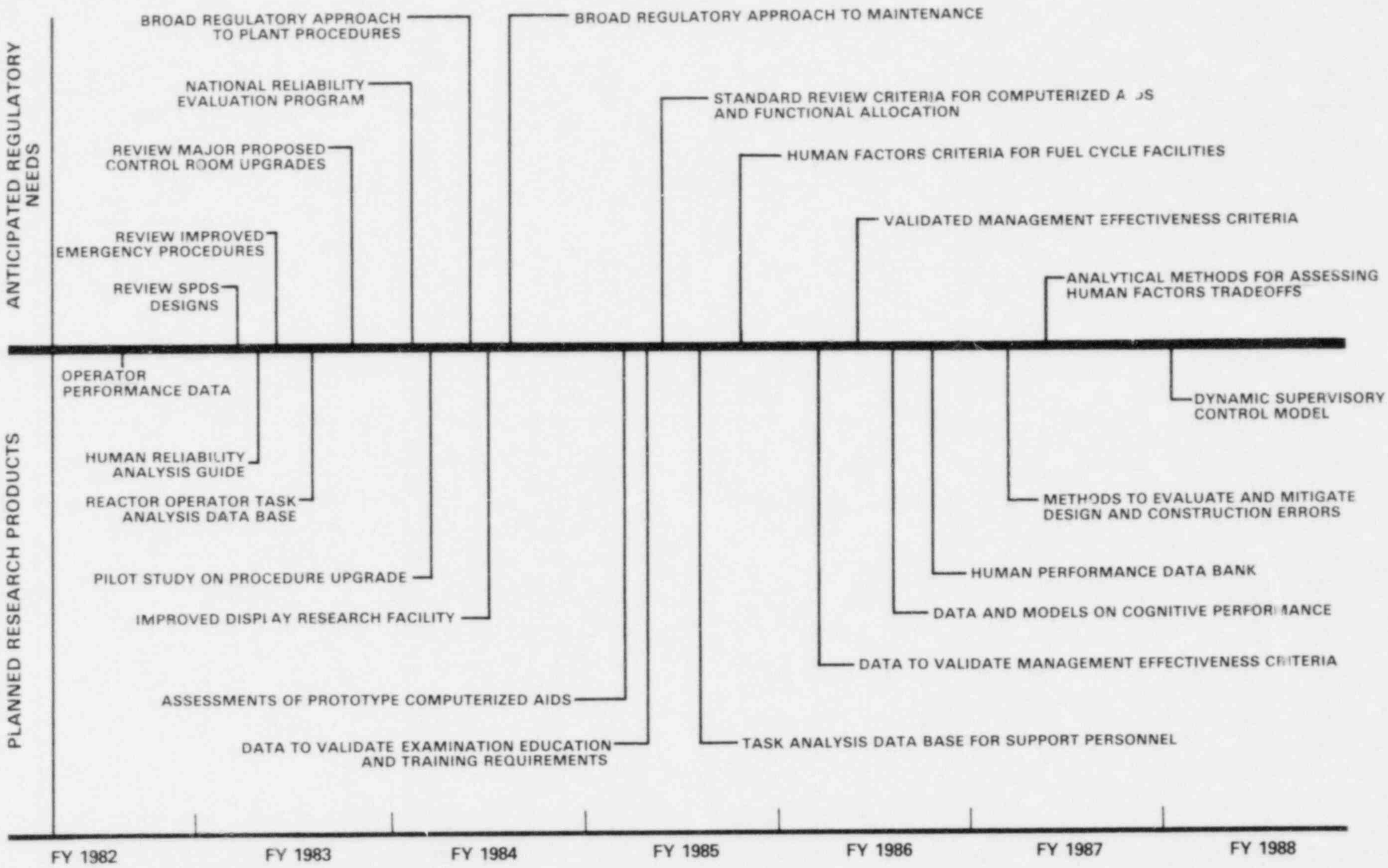

Figure 7.2 A comparison of anticipated regulatory needs and planned research products for human engineering

7.1.5.1 Human Factors Engineering

This subelement generates information, data, methods, and standards relevant to evaluating the operator-machine interface of nuclear facilities. Prior to the planning period, this subelement will have generated significant bodies of data on reactor operator response times during simulated accidents and on the effects of computerized display systems on operator performance. Task analyses for reactor operating crews will have been substantially completed.

During the planning period, task analysis for reactor operators will be reported. The products will include data on performance requirements of the crew in normal, transient, and accident sequences; the information required by operators to perform tasks; and probable sources of human errors. A software package will assist the NRC staff in using the task analysis results to confirm or develop requirements for staffing, equipment design, and training. Completion: FY 1984

Functional allocation studies will produce data and criteria to assist the staff in evaluating the proposed degree of automation of engineered safety features and other plant systems. The products will include the effects of automation on operator motivation, vigilance, and attitude and an assessment of the need to preserve manual operation as a backup to an automatic system. Completion: FY 1986

The methods, equipment, and effectiveness of computerized aids will be evaluated by laboratory experimentation, field trials, and analyses. The currently available graphic display research capability will be enhanced in 1983 and used thereafter to perform this task. The NRC will continue to participate in the Halden Reactor Project through 1987. The products will be data and information to help develop functional requirements and evaluation criteria for alarm filtering systems, disturbance analysis systems, computerized procedure manuals, artificial intelligence systems, and other computerized systems currently being considered for implementation in commercial reactors. Completion: FY 1988

Human engineering analysis will help develop or validate standards for equipment used or maintained by operators and support personnel of advanced nuclear power plants and fuel cycle facilities. The results of this research will assist the staff in preparing guidelines for future human engineering standards and in assessing the operation and maintenance of new designs. Completion: FY 1988

Supervisory control models will be developed, validated, and applied to facility operations. The products will assist the staff in confirming and developing regulatory requirements for, and value/impact assessments of, equipment design and operational and maintenance aids. The scope of this task includes conventional and advanced plants and fuel cycle facilities. Completion: FY 1988

7.1.5.2 Licensee Qualifications

This subelement generates information, data, methods, and standards relevant to evaluating the training and licensing of plant personnel and the management of design, construction, and operation of nuclear facilities. Prior to the planning period, this subelement will have generated sufficient information to arrive at interim regulatory positions on reactor operator education, training, and licensing. Performance measures that define long-term data-gathering efforts to confirm regulatory criteria will have been developed. Task analysis for support personnel of reactors and other nuclear facilities will have been initiated.

During the planning period, a primary goal is to validate the education, training, and licensing requirements for licensed operators. Criteria and training aids for operator actions during severe natural events, particularly earthquakes, will be developed (completion: 1984). The product of this work will be either a confirmation that current requirements are appropriate or detailed recommendations for changes that should be made to these requirements so that they become appropriate. Completion: FY 1985

Pending its successful application to reactor operators, the Instructional System Development (ISD) method, a technique proven in use by the military, will be used to establish training requirements for instrument and control technicians (completion: 1985), maintenance technicians (completion: 1986), fuel cycle facility operators (completion: 1987), and selected nuclear power plant support positions (completion: 1988). This work will serve as a portion of the technical basis for planned regulatory guides such as "Qualifications and Certification of Instrument and Control Technicians in Nuclear Power Plants," and "Qualifications of Maintenance Personnel."

Studies will be conducted and data collected to support three regulatory efforts. The first is the revision of Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training." This regulatory guide addresses the similarity that should exist between a simulator and the facility it simulates; simulator fidelity, testing, and upgrading requirements; and overall simulation capabilities (completion: 1984). The second is the planned development of a regulatory guide on "Nuclear Power Plant Simulator Training Programs," which will address the similarity that should exist between a simulator and the facility the operator is being trained to operate and the effective use of part-task, concept, and full-scope high-fidelity simulators (completion: 1985). The third is validation of the requirements for determining appropriate manual vs. automatic function allocation, which will be developed based on ANSI-N660, "Criteria for Safety-Related Operator Actions." Completion: FY 1986

Data will be collected to validate criteria established for evaluating the ability of a utility organization to effectively and safely manage a nuclear power plant. Completion: FY 1986

A systematic method of evaluating the effects of errors in design or construction on the ability of an operator to safely operate the plant will be developed. This will include criteria that could be used to evaluate whether additional training is an acceptable means of compensating for the operational problems that result from a particular design or construction error. An attempt will be made to define the point at which design or construction errors make a system too challenging to operate correctly. Completion: FY 1987

7.1.5.3 Plant Procedures

This subelement addresses research and standards relevant to developing and implementing sound procedures. Prior to the planning period, this subelement will have generated sufficient information to draft a proposed regulation and supporting regulatory guides aimed toward ensuring that technically accurate, human-engineered, step-by-step, and approved and validated operating procedures exist and are followed in each operating plant. Data based on quantifying the reductions in operator error rate from use of the upgraded procedures will be under development to confirm the benefits of these proposed regulatory requirements.

During the planning period, a pilot study will be completed to confirm the necessity and adequacy of the proposed requirements. Completion: FY 1984

Data will be provided to allow preparation of regulatory requirements that address operating procedures for all equipment important to safety, not just equipment covered by narrow historic definitions of "important to safety." As part of such research, data will be generated as necessary to acknowledge that as plants become more automated, the burden of proper equipment operation may shift from the reactor plant operator to the software programmer and maintenance personnel.

As detailed technical upgrading commences for each operating plant, data will be gathered to validate, refine, and optimize published regulatory requirements in FY 1986 through FY 1988.

In conjunction with the functional allocation studies and computer aid evaluations delineated in Section 7.1.5.1, research will be conducted leading to procedure preparation and use that adequately recognizes and addresses the balance that must exist so that new computer diagnostic equipment does not cause the operator to have an unsafe dependence on it and so that as the operator is placed more and more "out of the loop" by automation he does not become less prepared to handle unanticipated emergencies. Completion: FY 1988

After the fundamental deficiencies in operating procedure systems have been corrected, research may be desirable to explore and test alternative ways of presenting procedures to operators to optimize comprehension and response. In particular, studies related to the functional criteria and the design criteria of computer-based CRT displays may be necessary. Completion: FY 1988

7.1.5.4 Human Reliability

This subelement develops and verifies models of human performance and quantitatively assesses the human contribution to risk. Prior to the planning period, this subelement will have generated methodologies, guidelines, and a data base suitable for constructing models of human performance to assess human reliability ("Human Reliability Analysis with Emphasis on Nuclear Power Plant

Applications," NUREG/CR-1278), principally in those areas involving skill and rule-based behavior (stimulation and response) as opposed to cognitive behavior. Assessments will have been performed to determine if these methods and models, when utilized by different users, produce repeatable results.

The human performance models presented in NUREG/CR-1278 will, in part, be validated by two activities: (1) the licensee event reports will continue to be assessed to determine gross human error rates and, to the extent possible, the causes of human error and (2) a current program to record the actions taken by reactor operators, corresponding response times, and the action sequences for selected accident scenarios during simulator runs will be continued. These activities should reduce the uncertainties in human reliability assessments that will be performed to estimate the benefits of proposed regulatory requirements. Completion: FY 1984

Simulator evaluations of the ability of operators and operating teams to successfully deal with selected accident sequences will be conducted. The results of this data collection program will be the development of models and data requirements for modeling and evaluating the ability of operators or operating teams to perform cognitive tasks. Such tasks consist of the recognition of offnormal conditions, the interpretation of these data to ascertain reactor status, and the determination of the action required to bring the reactor to, and maintain it in, a safe state. The product of this effort will be the development and experimental validation of cognitive models for performing reliability assessments of the operator involved in decisionmaking tasks. Completion: FY 1986

A human performance data bank will be established (FY 1985) and maintained to serve as a repository of information for use by human reliability analysts.

Models will be developed to determine the reliability of plant personnel in correctly performing maintenance and test activities and to determine those design and operational factors that most affect the mechanic's or technician's ability to accomplish the intended tasks. This work will form a basis for determining the reliability improvement that could be expected from various changes or improvements in work environments, job aids, training, etc. Improved modeling capability will also improve the determination of risk contribution resulting from maintenance and test errors. Completion: FY 1986

An event report investigation group will be established to demonstrate the feasibility and effectiveness of applying classical methods such as the critical incident technique to determine, from appropriate event reports, the root causes of human error. Factors that influenced the operator either to take incorrect action or to respond correctly will be determined and used to validate operator and maintenance activity reliability models. Completion: FY 1986

Reportable human errors, defined as those that result in event reports, provide a currently available human error data base; however, this is believed to be a small percentage of those errors that occur but are promptly detected and corrected and therefore do not exist for a sufficient period of time to require reporting. A better understanding of human error potential could be determined i· all such human errors were reported for analysis. A reporting system similar in nature to the Aviation Safety Reporting System which the National Aeronautics and Space Administration provides to the Federal Aviation Agency will be considered. Plans for instituting such a reporting system and subsequent report analysis will be developed by FY 1984 and, if acceptable, implemented by FY 1985.

7.2 Plant Instruments and Controls

.1 Issue

. research described in this element covers instrumentation, control, protection, and electric systems having an effect on plant safety. The issues that this program element is intended to resolve center around concerns that have been identified in four major areas.

1. Can malfunctions in plant control, protection, and other instrumentation and electric systems lead to unanticipated transients or accidents either automatically or through operator actions or inactions or through malevolence?

2. How can individual instrumentation and electric system hardware components malfunction, and how can the probability of undesired failure modes be reduced?

3. What is the adequacy of instrumentation, equipment, and systems and techniques for analyzing their output data used for (a) diagnosing problems in reactor systems to help prevent accidents or (b) monitoring the course of an accident in operating reactors for abnormal conditions such as developed in the LOCA and Transient (Chapter 2) and Accident Evaluation and Mitigation (Chapter 4) decision units?

4. What is the usefulness of new instrumentation and control (I&C) equipment being brought on the market for nuclear power plant applications?

7.2.2 Research Program Objective

The research outlined in this program element consists of the following four major areas as discussed in paragraph 7.2.1.

1. Evaluate malfunctions of plant control, protection, and other instrumentation and electric systems to determine the impact of these malfunctions on plant operations and equipment important to safety, particularly where such events could lead to unanticipated transients or accidents. Evaluate automatic analog and programmable digital computer-based equipment and information systems used by operators for manual actions to determine their suitability for use in nuclear power plants.

2. Evaluate individual instrumentation and electric system hardware components to determine the mechanisms that can result in component or system malfunctions and to identify ways of reducing the likelihood of undesired failure modes, including safeguards considerations.

3. Evaluate equipment and systems used for diagnosing problems in reactor systems to help prevent accidents and equipment used for following the course of an accident to determine their adequacy for use in nuclear power plants.

4. Evaluate technological advances in the state of the art of instrumentation and control and electric equipment, initially developed for use in other fields, and determine their adequacy for use in nuclear power plants.

Based on the results of the research performed in these four areas, experiences at nuclear power plants, and needs identified by the NRC, by national or international standards bodies, or by members of the public, regulatory requirements or recommendations will be prepared and issued where justified.

7.2.3 Relationship to Other Programs

The work pertaining to the effects of control systems failures on plant safety systems will provide input to NRR for the unresolved safety issues A-47, "Safety Implications of Control Systems," and A-49, "Pressurized Thermal Shock." Efforts are under way to invite participation of utility representatives in these activities through the auspices of the Electric Power Research Institute (EPRI).

Evaluation of alarm systems will augment the human factors technical assistance work at Hanford Engineering Development Laboratories (HEDL) in support of the Human Factor Branch at NRR. EPRI is evaluating disturbance analysis systems, including consideration of alarm systems. NRC research efforts in this area are being coordinated with EPRI.

NRC/NRR, EPRI, the Atomic Industrial Forum (AIF), and the operating utilities are all involved in programs to implement Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." The research related to nuclear power plant instrumentation evaluation will both receive input from and help provide guidance to these programs.

The noise surveillance and diagnostics research program is being closely coordinated with the NRC/NRR technical assistance programs on (1) neutron noise surveillance and diagnostics implementation and (2) loose parts detection and acoustic noise diagnostics.

7.2.4 Background and Status

The summary of FY 1982 and FY 1983 programs for the plant instruments and controls element given below pertains to confirmatory research in the four major areas discussed. This program is aimed at obtaining information for use by the NRC staff in evaluating licensee submittals or for new or improved regulatory guidance and rules.

The FY 1982-1983 program summary by major area is:

7.2.4.1 Plant Control, Protection, and Electrical Systems

1. Complete control system failure modes and effects analyses for reference Babcock and Wilcox (B&W) and Westinghouse plants and develop preliminary criteria for control and electrical systems. Plant electrical system criteria will include consideration of degraded conditions for power supplies to control systems such as overvoltage, undervoltage, underfrequency, etc., in addition to gross equipment failure (go/no-go) situations for the plant electrical systems.

2. Identify alternative methods for setting priorities for alarms

3. Issue for public comment a regulatory guide endorsing IEEE 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Plants," in FY 1982. This guide will provide recommendations for complying with the Commission's regulations with respect to the design, reliability, and qualification and testability of the electric, instrumentation, and control portions of the safety systems. It is expected that the active guide will be issued in FY 1983.

4. Identify preliminary design guidelines for programmable digital computers to include consideration of the relative risk of using computer-based systems in place of analog systems; applications of computer-based safety parameter display systems; and protection for MOSFET devices, integrated circuit "AND" and "OR" logic chips, and memory chips.

5. Issue regulatory guidance on computer software design and quality assurance. This guidance is needed because of the trend toward greater use of computers in nuclear power plant systems.

6. Assess the potential impact of the use of solid-state motor controllers to determine if they are sufficiently reliable for use in systems important to safety.

7. Develop a regulatory guide on d.c. power system reliability in accordance with the recommendations of NUREG-0666 on d.c. system reliability.

7.2.4.2 Plant Instrumentation Components

1. Evaluate fault-current actuated devices and other isolation methods used for computers and instrumentation and controls important to safety. The evaluation will include a review of the existing regulatory guidance on the use of fault-current actuated devices such as isolation devices between 1E and non-1E circuits and the use of redundant (series) or diverse fault-current actuated isolation devices. In addition, an evaluation will be made of the reliability of optical isolation devices being used or planned for use as isolators between systems important to safety and systems not important to safety.

2. Identify, as a result of individual component assessments, needed quality assurance requirements for components such as terminal blocks and pressure transducers.

3. Develop criteria for the design, installation, protection, and testing of pressure-sensing lines; response-time testing; and the establishment of setpoints for instrumentation transducers.

4. Provide technical basis for guidance for the overload protection of valve actuators, including review of existing regulatory guidance (for example, Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," with recommendations regarding bypassing of thermal overload protective devices). Recommendations of the NRR staff on valve position indications and valve lockout features will also be reviewed.

5. Initiate studies of instruments to measure low flow (natural circulation) and pressure vessel level.

6. Review Regulatory Guide 1.97 and evaluate with regard to the problems anticipated in implementing its provisions. This guide provides the NRC staff's recommendations for providing monitoring instrumentation that would provide the operator with the necessary information to follow the course of an accident and would indicate the need for operator action in mitigating accident consequences. Initial evaluations of the guide will be completed in FY 1982 with final recommendations made in later years. Issues that cannot be corrected without further research will be identified and programs implemented to resolve these issues.

7.2.4.3 Diagnostic Instrumentation and Methods

1. Evaluate instrumentation needed to diagnose and monitor nuclear power plant status for both normal operating conditions and abnormal situations. This evaluation will complement the efforts to assess plant monitoring systems under the LOCA and Transient (Chapter 2) decision unit.

   Diagnostic instrumentation most important to safe operation or functioning of LWRs will be identified. The state of the art of diagnostic instrumentation and current use of such instrumentation in nuclear power plants will be evaluated. Recommended upgrading/modifications to existing NSSS instrumentation will be made. Techniques for analyzing data from diagnostic instrumentation will be developed, and possibly tests of particular diagnostic instrumentation will be made.

2. Continue the demonstration of a continuous on-line surveillance and diagnostics system at an operating PWR (Sequoyah Unit I).

3. Continue the development of "normal operating neutron signatures" of a representative group of BWRs and PWRs for eventual use as an essential ingredient of noise diagnostic tools.

4. Obtain LOFT and Semiscale test data associated with test programs within the LOFT (Chapter 3) and LOCA and Transient (Chapter 2) decision units respectively, for use in discerning the sensitivity of noise analysis methods to abnormal reactor operating conditions.

5. Obtain other diagnostic data of operating BWR stability characteristics to support confirmatory evaluation of industry stability analyses methods.

6. Initiate a study for the on-line remote isotopic monitoring of reactor coolant activity, and evaluate currently available on-line radiation monitors for detection of failed fuel. This program is complementary with the fission product release and transport studies under the Accident Evaluation and Mitigation (Chapter 4) decision unit.

In addition to the above, an effort will be undertaken to survey the industry to identify regulatory guides, regulations, or NRR branch technical positions that are either (1) impractical or impossible to meet, (2) outmoded because of new technology, or (3) interfering with implementation of new technology and to prepare definitive alternative regulatory positions where needed. Emerging new I&C technology will be reviewed, evaluating its feasibility, reliability, safety impact, cost/benefit, and regulatory impact.

7.2.5 Research Program Plan

The long-range research program will continue along the lines established in FY 1982 and FY 1983. The principal programs will be pursued in the four categories identified previously. In addition, a methodology for value/impact assessments in the I&C area will be developed in FY 1984. These principal programs are:

7.2.5.1 Plant Control, Protection, and Electric Systems

The work on the safety implications of control systems and the closely related program of plant electric systems evaluation will continue along the lines of the FY 1982 and FY 1983 programs. During FY 1984 and FY 1985, the failure mode and effects analysis for the Combustion Engineering and General Electric selected reference plants will be completed. Follow-on work in the period from FY 1986 through FY 1988 will include an upgrading of the plant models to include systems and components not completely covered in current models and also a more comprehensive study of systems interactions and failures. Specifically, it is anticipated that the control system and electric system studies will identify areas of interaction between the two systems that should be studied. The two models will probably be combined into a single model of the total plant for the purpose of studying these problems in FY 1987 or FY 1988. For the General Electric reference plant study, the results from the work performed as part of the Full Integral Simulation Test (FIST) program will be used to augment the modeling of the thermal hydraulics of the plant.

The various techniques that have been employed by architect-engineers in the design of nuclear power plant control systems to develop criteria for control system design practices will be evaluated. This evaluation will be performed, beginning in FY 1986, as the work on failures in control and electric systems is being completed.

The use of programmable digital-computer-based protection and control systems will introduce additional safety considerations to the regulatory review process. These issues will be addressed in the period of FY 1984 and FY 1985. Specifically, the program will address the questions of isolation, separation, redundancy, diversity, security, and vulnerability of both the functions to be performed and the equipment used and will evaluate the relative reliability of digital systems. New computer systems use small integrated circuits that are particularly vulnerable to damage due to overloads. Computer protection against various kinds of overloads such as radio frequency interference (RFI) and electromagnetic interference (EMI) will be evaluated. Methodology for protecting computer systems against software errors by various verification and validation schemes will also be evaluated. Design criteria will be developed, and additional regulatory guidance on both hardware and software will be prepared during FY 1985 and FY 1986. Computer software quality assurance criteria will be reviewed and regulatory guidance updated as needed during the period of FY 1986 through FY 1987.

A study will be initiated in FY 1985 to determine the feasibility of automatically detecting safety channel component failures that would render the safety channel(s) inoperative.

Results of this effort may result in changes to regulatory guides such as 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," and 1.118, "Periodic Testing of Electric Power and Protection Systems."

Criteria will also be developed in the same time frame for acceptance of automatic testing features for safety system equipment. Evaluations will be made of equipment that contains automatic testing features.

7.2.5.2 Plant Instrumentation Components

This research will be focused toward providing an assessment of nuclear power plant instrumentation systems to meet their functional performance requirements. Early efforts will consist of evaluation of specific categories of instrumentation and work on specifically identified problems.

Evaluations of instruments needed to monitor the course of an accident will continue in FY 1984 through FY 1986. This work will consist of additional tests and evaluations or analyses of specific instrumentation to better define its performance under the conditions defined in Regulatory Guide 1.97. A draft revision of 1.97 will be started in FY 1985.

Performance of individual instruments under two-phase conditions will be included in these studies for all instruments important to safety. Low flow (natural circulation) and pressure vessel level measurements are among those issues expected to be pursued in this period by using the transient two-phase flow test loop at the Idaho National Engineering Laboratory in FY 1984 and FY 1985.

Effects of pressure transducer instrument-sensing lines will continue to be studied to determine their effect on response time and accuracy. This work should be completed in FY 1984.

The initial evaluation of the current state-of-the-art equipment with some limited benchtesting may lead to a full-scale evaluation on a test loop capable of studying performance under a full range of normal and accident conditions, including two-phase flow conditions, in FY 1984 and FY 1985. The initial feasibility study or an in-place test method for determining the response time of a neutron sensor may lead to a demonstration test on an operating nuclear power plant in FY 1984.

7.2.5.3 Diagnostic Instrumentation and Methods

In FY 1984, the demonstration of a continuous on-line surveillance and diagnostics system will be started at an operating BWR. The BWR demonstration will be completed and an evaluation of the system's capability for surveillance and diagnostics made in FY 1985. The main thrust of the diagnostics research program will be to continue to develop noise surveillance and diagnostics techniques necessary to support NRR needs. Efforts in acoustic-emission monitoring of the integrity of pipes, welds, and pressure vessels and in fault prevention and detection by vibration monitoring of moving components such as pumps and valves will be performed in FY 1986. An effort will also be initiated in FY 1987 or FY 1988 to automate portions of the noise signature diagnostic process to enable the user to concentrate on the most vital decisionmaking tasks. The continuous on-line noise surveillance and diagnostics system may be groomed for use in specific NRC-required diagnostic tasks, depending on results of the earlier PWR and BWR demonstrations.

The diagnostic instrumentation evaluation that should be completed in late FY 1984 or in FY 1985 will be used as the basis for the preparation of a regulatory guide on this topic beginning in FY 1985.

Evaluation of instruments to measure coolant void fractions will begin in FY 1985 and will be completed in FY 1988.

The on-line remote isotopic monitoring system test demonstration at an operating LWR will be initiated in FY 1984 and completed in FY 1986. The system will be an adaptation of the isotope detection system designed for LOFT and will encompass determination of the normal operating envelope and effects of various plant parameters for coolant radionuclide concentrations. In FY 1985, data interpretative methods tailored to meet nuclear reactor operator needs will be developed, followed in FY 1986 by testing of these methods in a facility to be selected. Instrument response to fission product release during an accident and response of an on-line reactor coolant monitor to the extreme case of fuel breakup and transport of fuel particles in the coolant system will be evaluated. This evaluation will complement studies in the Accident Evaluation and Mitigation (Chapter 4) decision unit. The final product of this research program will be the development of a generic design description with associated system requirements for use in commercial nuclear power plants, including full documentation of the prototype system and operational procedures for hardware and software.

7.2.5.4 Advanced Concepts

Research programs will be initiated in the future as the needs become more definitive. These programs will capitalize on the results of the ongoing research to explore new concepts. The following are some examples:

1. Evaluation of the safety implications of advanced systems concepts such as (a) distributed controls involving microprocessors and enhanced communications among several control locations, (b) optimum controls that automatically select the most advantageous value among several variables instead of using one setpoint, and (c) diagnostics using artificial intelligence will be undertaken in FY 1985. Also a study of current test frequency requirements of engineered safety feature actuation systems (ESFAS) will be made in FY 1986 to determine if testing required by plant technical specifications is reducing plant safety.

2. A fluid emergency core cooling activation system will be evaluated beginning in FY 1987. This system would be totally diverse to the electric systems currently in use by making fluid measurements and having fluid actuators. The power source could be compressed inert gas, making the system totally diverse and thus offering possible protection against many common-mode failures.

3. A study to augment the "functional redundancy" within nuclear power plant instrumentation will be conducted beginning in FY 1987. This concept refers to the on-line monitoring of plant instrumentation by a computer-based system to discover discrepancies or malfunctions. The computer system would use diagnostic software, taking into account various diverse parameter values and plant characteristics to obtain a reasonable validation of various measurements.

7.3 Occupational Radiation Protection

7.3.1 Issue

The key issues of the NRC concerning the operational aspects of occupational radiation protection at present are (1) implementation of the occupational ALARA concept, (2) improvements in health physics measurements, (3) improvements in the control of the dose from internally deposited radioactive material, (4) improvements in radiation protection in the performance of personnel engaged in NRC-licensed activities, and (5) methods of reducing dose rates and working times in nuclear power plants. The occupational radiation protection research program is designed to ensure progress in each of these areas.

7.3.2 Research Program Objective

The basic objective of the occupational radiation protection research program is the development of a comprehensive body of regulations, guidance, and information that is intended to ensure an adequate degree of radiation protection for workers in NRC-licensed activities. The associated research program is intended to support this objective as necessary to ensure that the regulations, guidance, and information are technically sound, practical, and up to date. The specific objectives of the research program are (1) to integrate the occupational ALARA concept into the Commission's routine licensing and inspection program, (2) to establish and require the use of accreditation and certification programs affecting health physics measurements, (3) to modernize and standardize internal dose controls such as air sampling, bioassay, and dose calculational methods, (4) to develop and implement effective methods for improving the performance of workers with respect to occupational radiation protection, (5) to establish a technical basis for requirements that would reduce dose rates and working times for workers at nuclear power plants, and (6) to establish standardized health physics requirements and staff positions as needed to facilitate license application reviews.

7.3.3 Relationship to Other Programs

In early 1981, the EPA published for comment in the Federal Register a proposed revision of the Federal radiation protection guidance for occupational exposures. The proposed guidance reflects to a considerable extent new recommendations of the International Commission on Radiological Protection (ICRP) that are being adopted throughout the world. These recommendations include several new protection procedures that are to be incorporated into NRC regulations. Several elements of the occupational radiation protection research program have been included to facilitate implementation of the new recommendations.

The NRC occupational radiation protection research program is coordinated with the programs in other Federal agencies to ensure cooperation and to minimize duplication of effort. For example, the NRC and DOE currently are participating in a jointly funded and managed program to determine if currently available health physics instrumentation and bioassay laboratories can meet the performance specifications given in draft ANSI standards. The NRC also participates in an interagency committee established to ensure that all affected governmental agencies can use an accreditation program for personnel dosimetry processors that is being established by the NRC in cooperation with the Department of Commerce.

Projects intended to reduce dose rates involve corrosion product buildup, decontamination effectiveness, and decontamination impacts on solidification and waste disposal. These projects are related to a larger overall program on this general topic being conducted by the NRC, DOE, and industry.

In particular, the NRC and the Institute of Nuclear Power Operations (INPO) are initiating a cooperative effort with the objective of improved health and safety in the nuclear power industry, thus minimizing the necessity of increasing regulatory controls. This effort includes an INPO appraisal program to assess and improve the safety performance of individual nuclear power plants.

7.3.4 Background and Status

Anticipated progress prior to FY 1984 may be summarized as follows:

7.3.4.1 Occupational ALARA

An amendment to the Commission's regulations is planned that would require affected licensees to develop and implement occupational radiation protection programs that include the ALARA concept.

7.3.4.2 Measurements Improvement

Plans for development of regulatory accreditation or certification programs for personnel dosimetry processors, bioassay laboratories, and health physics survey instruments will continue. Activities on other projects will include neutron, beta, and low-energy photon dosimetry techniques; dose-equivalent index measurement capabilities; air-sampling performance and effectiveness; and guidance on health physics surveys.

7.3.4.3 Internal Dose Control

Additional bioassay guidance, including the conversion of bioassay results to intake (for compliance purposes), is planned. The respiratory protection program, including emergency preparation and response guidance and updating of the general respirator guidance, will continue. The feasibility of using single-atom detection technology in bioassay applications will be investigated.

7.3.4.4 Personnel Performance Improvement

Training guides and manuals will be issued along with guidance on minimum qualifications for Radiation Safety Officers. Minimum qualification requirements for nuclear power plant health physics technicians will be studied. Development of a program for industrial radiographer certification will be pursued.

7.3.4.5 Dose Rate Reduction

The corrosion product buildup project will include the development of models for the reactor coolant sampling system and for corrosion product transport and deposition. The decontamination effectiveness effort will include identification and evaluation of innovative decontamination alternatives and methods for primary system surfaces. The project on decontamination impacts on solidification and waste disposal will include the development of an overall problem statement and a laboratory program to assess possible solutions.

7.3.4.6 Radiation Protection Standards for Licensing

This effort will include updating of guides explaining how to complete license applications, codifying a licensing condition on teletherapy gamma monitors, and developing a guide on nuclear power plant radiation protection programs required in license applications.

7.3.5 Research Program Plan

The NRC occupational radiation protection research program is divided into the following areas for accomplishment of the program objectives.

7.3.5.1 Occupational ALARA

Beginning in FY 1984 and continuing through FY 1988, the NRC will fund optimization studies that will permit quantitative determinations regarding the benefit and cost of various occupational safety measures. Through optimization analysis, the point of minimum cost and the radiological detriment plus equipment and manpower dollars can be identified.

During FY 1984, regulatory guides will be developed that provide guidance in the design and operation of low-level-waste disposal facilities and manufacturing plants to maintain occupational radiation exposures at these facilities ALARA. Active guides will be completed in FY 1984 for occupational ALARA at uranium fuel fabrication plants and at LWR spent fuel storage facilities.

7.3.5.2 Health Physics Measurements Improvement Program

During FY 1984, testing of commercially available health physics survey instruments against a draft ANSI standard will be completed. The results of the tests will be used to verify the adequacy of the standard, and a regulatory certification program for survey instruments will be established. The staff will propose to the Commission that the regulations be amended in FY 1985 to require NRC licensees to use instruments that have successfully met the performance criteria of the certification program. The certification program will be accomplished through a third-party certification system. It is anticipated that the third-party certifier will recover the cost of the program through a fee system.

During FY 1984, tests of the adequacy of a draft ANSI standard on the performance of bioassay laboratories will be completed. The test results will be used to verify the adequacy of the standard, and the NRC will use the standard as a basis for amending its regulations to require NRC licensees to use the services of bioassay laboratories that have been tested, found to meet the provisions of the standard, and therefore accredited.

The dose equivalent index (DEI) is defined by the International Commission on Radiation Units and Measurements (ICRU) as the dose received at the tissue depth where it is maximized. The presently existing health physics instrumentation is not designed to measure the DEI. In FY 1984, a study will be initiated to investigate the feasibility of designing health physics instruments that would be capable of measuring the DEI. The NRC will use the results of this study to determine whether to require NRC licensees to measure the DEI.

The problem of making beta measurements has existed for some time and is becoming more acute because of the increased maintenance at nuclear power plants requiring workers to enter workplaces highly contaminated with mixed fission products and activated corrosion products emitting significant beta radiation. In FY 1985, the NRC will initiate a study to evaluate the occupational exposures to mixed localized and extended sources of beta and gamma emitters in the workplace. The results of this study will be used to publish a regulatory guide on beta measurements.

Currently no standards exist for extremity dosimetry. It is important to improve extremity dosimetry equipment and procedures because many workers at NRC-licensed facilities are required to work in radiation fields that may result in significant extremity doses but small whole-body doses. In 1980, 66 of the 92 reported overexposures were due to improper extremity monitoring. Research of extremity dosimetry techniques is necessary to aid in the development of appropriate performance standards for extremity dosimeters. The NRC staff expects to initiate a study concerning extremity dosimetry during FY 1984.

A national standard on instrument performance is being proposed. However, the extreme radiation environments that can occur during accidents are not included in the standard and associated tests. Since extremes in temperature, pressure, humidity, and shock can occur in accident situations, it is important to determine the capabilities of monitoring instruments to function under these conditions and to improve instrument reliability if necessary. The NRC plans to initiate research in this area during FY 1988.

7.3.5.3 Internal Dose Control Program

Recent ICRP guidance has modified the dosimetric models used to assess internal doses from the deposition of radionuclides and has resulted in recommended changes in permissible concentrations in air to which workers may be exposed. During FY 1984 thru FY 1988, the NRC will obtain analytical support for the evaluation of various mathematical models for determining doses from internal deposition of radionuclides and for interpreting ICRP guidance. This support is required in order for NRC to implement the ICRP recommendations.

The use of respirators for radiation protection is becoming increasingly important in view of the ICRP 26 recommendation that internal doses be added to external doses to determine compliance with dose limits. In order to evaluate the effectiveness of various respirators, the NRC will obtain laboratory support during FY 1984 through FY 1988. This support is required to maintain Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," and NUREG-0041 and to evaluate NRC licensee requests for special respirator applications. During FY 1985, the NRC will obtain support for a study on quantitative fit testing of respirators. Also during FY 1985, a study will be conducted to determine medical criteria for respirator users. The results of this study will be published in a revision of NUREG-0041.

In its effort to determine internal doses more accurately, the NRC will evaluate the use of diethylene triamine pentaacetic acid (DTPA) in determining body burdens. Injections of DTPA will cause certain heavy metals to be excreted from the body at a more rapid rate. This enables lower body burdens of the more radiotoxic nuclides such as plutonium and americium to be detected more readily. This research effort will be accomplished during FY 1984 and FY 1985. Evaluations of the use of single-atom detection technology in bioassay applications will continue.

During FY 1985 through FY 1988, an evaluation will be made of the bioassay programs at selected NRC-licensed establishments to determine the effectiveness of the programs in detecting and measuring the intake of radionuclides by workers. The information obtained from these studies will be used to prepare regulatory guides on bioassay procedures for selected radionuclides.

7.3.5.4 Personnel Performance Improvement Program

During FY 1984, the NRC will complete a skills analysis for competent performance as a health physics technician at nuclear power plants. The esults of the study will be used to develop training needs for in-plant training programs. If this program proves to be successful, in future years the NRC will conduct similar programs for other types of licensees.

In FY 1984, active guides will be completed concerning general radiation safety officer qualifications and recommended radiation safety training for workers at uranium fuel fabrication plants. In FY 1985, guides that provide recommended radiation safety training for workers at uranium mills, manufacturing plants, medical institutions, and special nuclear material plants that possess less than critical amounts of special nuclear material will be developed. In FY 1987, guides will be developed for radiation safety training of workers at plutonium fuel fabrication plants, gamma irradiator facilities, and source material plants.

7.3.5.5 Dose Rate Reduction Methods

During FY 1984 and FY 1985, a study on the effect of activated corrosion product buildup on occupational exposure will be conducted. This effort will use existing data from power reactor experiences and may include limited contacts with operating power reactor personnel through INPO to confirm and substantiate information. After the sources of the occupational dose are identified, means of reducing the exposure rates will be evaluated. Work will continue on developing a model to upgrade the capability to predict corrosion product transport and deposition. Models developed to date have not accounted for the radiological conditions presently observed and have not yielded methods to reduce exposure buildup or to expedite decontamination. The existing deposition models will be evaluated, and improvements will be made based on measurements of radionuclide depositions in reactors and on data collected from primary coolant sampling systems based on a new design for use at operating temperature and pressure.

Activities in FY 1984 on the decontamination effectiveness project will be concentrated on work to assess the expected benefits to total occupational dose reductions resulting from decontamination of reactor primary systems, as well as the dose required to perform the decontamination. The increase in occupational exposure in recent years has been caused by NRC-mandated activities and increased dose rates. One of the recognized improvement methods to reduce the current trend in increasing occupational exposure is decontamination of primary systems and components. Examination of decontamination effectiveness will provide NRR with the information needed to properly evaluate existing and innovative decontamination methods and procedures in an attempt to reduce the occupational dose and to assess applicant occupational ALARA designs and programs. This effort will include monitoring industry research, experimental work, and practical experience for effectiveness and evaluating the cost of alternative methods for decontamination of primary systems. Comparisons for selected operations will be made by calculating the dose rate for each operation with and without decontamination. With proper planning and after evaluating the most effective method for a particular decontamination, a reduction in radiation exposure should occur. Using the computer model and the best information from investigational/experimental work regarding radionuclide inventories in typical PWR and BWR systems, dose savings produced by decontamination for typical tasks will be calculated.

Work will continue during FY 1984 on assessing the capability of in-plant processes to convert chelating agents into more acceptable forms and the capability of solid waste systems to handle waste resulting from decontaminations during facility operations. Acceptable forms would include chelate quantities corresponding to those discussed in the draft technical position, "Disposal of Wastes Containing Chelating Agents." Work will continue on assessing the additional occupational exposure arising from decontamination solutions produced during the processing of solids, transportation, and waste disposal.

In FY 1985, a study will be conducted to identify the specific steps in work performed by technicians at nuclear medicine laboratories and nuclear pharmacies that result in the highest whole-body and extremity doses. A regulatory guide will be published concerning the findings of this study.

Medical personnel who regularly attend patients to whom diagnostic doses in the millicurie range of Tc-99m have been administered are generally not considered in licensee personnel monitoring programs. Therefore, the NRC has very little information concerning the occupational doses received by such personnel. During FY 1984 and FY 1985, an evaluation will be made of the doses received by workers who attend patients with diagnostic doses.

During FY 1988, in order to independently assess opportunities for exposure reduction at nuclear power plants through plant design, a list of maintenance tasks with significant possibilities for occupational exposure reduction will be compiled. This information will be used to evaluate alternative designs and procedures and the corresponding effects on worker exposure. It may be possible to establish preventive maintenance programs that incorporate the use of remote controls and component designs to reduce cross-exposure when maintenance on other components is being performed.

7.4 Emergency Preparedness

7.4.1 Issue

Considerable progress has been made in upgrading the state of emergency preparedness and identifying and addressing the regulatory issues associated with emergency preparedness. Among those issues requiring long-term resolution are the following:

1. What information is needed and how should it be used by Federal, State, and local governments and by licensees in responding to radiological emergencies?

2. Have all the interfaces between Federal, State, and local governments and licensees been adequately addressed in order to cope with a radiological emergency? If not, what must be done?

3. What are the optimum scope and frequency of emergency preparedness exercises?

4. Are evacuation time estimates accurate enough?

5. Using results of research on fission product sources and behavior, what emergency preparedness requirements are appropriate for fuel cycle and nuclear material licensees and for advanced reactors?

7.4.2 Research Program Objective

This element's principal objectives are to:

1. Ensure that proper data and procedures are developed to assess the course of an accident and its potential severity,

2. Ensure that adequate information is available to take appropriate actions for actual or anticipated conditions, and

3. Develop the technical basis and resulting regulatory standards to resolve the issues previously mentioned.

The research and standards program in emergency preparedness will help improve the capability of Federal, State, and local government and licensees to mitigate the consequences of an accident.

The products of this program include:

1. Data and a validated computer model for estimating evacuation times for sites with high population densities, including natural disasters or other factors that complicate the evacuation,

2. A calculation capability to verify the adequacy of notification system designs using licensee design plans and topographical maps,

3. Functional requirements and design specifications for an integrated information processing system for the NRC Emergency Operation Center for use in assessing accident scenarios in real time,

4. Emergency preparedness requirements for advanced reactors,

5. A pilot installation and testing of a ring of fixed environmental monitors with data telemetered to a central point for use as a detection system of radiation resulting from an accident,

6. A prototype rapid assessment system for measuring radioiodines and particulate fission products in the environment (e.g., air, surface water, cows' milk), and

7. A prototype system for characterizing the airborne plume from an accidental release.

This information will support the development of regulatory positions. The ultimate goal is to provide guidance for licensees and for Federal, State, and local governmental agencies to ensure adequate preparation to cope with radiological emergencies and thereby mitigate the consequences of these emergencies.

7.4.3 Relationship to Other Programs

NRC's emergency preparedness research is coordinated with the activities of other organizations (Table 7-4) to ensure adequate coverage of important topics without unnecessary duplication. Coordination occurs most often through management and staff personnel exchanges of plans and results. This approach will continue through the planning period. Beyond the routine coordination and agreements, particular efforts are likely to be needed in gathering data for the validation of regulatory criteria and in developing emergency preparedness standards.

7.4.4 Background and Status

In August 1980, the NRC published a final regulation that upgraded emergency preparedness requirements for nuclear power plants. Research to assess warning system capabilities and the public's response during an emergency is under way

Table 7-4. NRC's Emergency Preparedness Research and Standards Related to and Coordinated with Efforts of Other Organizations.

Organization	Coordinated Activity
NRC/RES	Long-Range Research Plan, Sections on Severe Accident Sequence Analysis, Severe Accident Mitigation, Siting and Environmental Impact, Fission Product Release and Transport
NRC/IE NRC/NRR NRC/NMSS	Definition of research needs; implementation of standards and regulations; technical assistance project
American Nuclear Society	Standards on emergency preparedness
IAEA	Information exchange and review of documents relating to emergency preparedness
Federal Emergency Management Agency	Research and standards development relating to offsite emergency preparedness

to help establish detailed criteria for its implementation. It is anticipated that the emergency preparedness regulation for nuclear power plants will be considered for revision at some time to reflect the experience gained in its use as well as future developments resulting from research on fission product source terms and behavior. By the start of the planning period, related elements of NRC's research program will have generated a substantial body of information on the time-dependent radiological source term for reactor accidents and improved analytical techniques for quantifying their potential health effects. This information will help formulate subsequent efforts.

Consideration is being given to a rule change to strengthen emergency preparedness requirements for those fuel cycle and materials licensees (other than reactors) that have the potential for accidents involving radioactive materials that could threaten the health and safety of the public. Publication of this proposed regulation is scheduled for FY 1982.

In parallel with the upgrading of the regulations on emergency preparedness, updated regulatory guidance corresponding to regulation changes is in preparation.

7.4.5 Research Program Plan

The long-range plan for emergency preparedness research is directed toward the resolution of long-term issues related to mitigating the consequences of a radiological emergency. Efforts to date have focused on developing emergency preparedness regulations for light-water reactors. Related confirmatory research is expected to continue well into the planning period. This will include greater emphasis on the ability to determine the magnitude and timing of a radiological release, better definition of organizational and individual responsibilities, understanding information flow during an incident, and improving the NRC's capability to protect the public health and safety during a radiological emergency. Figure 7.3 relates the planned availability of research products to our current assessment of anticipated regulatory needs. Integral to our planning is continual assessment of the safety significance of emergency preparedness, as derived from reviews of documented operating experience and risk analysis.

An in-depth review of a sample of existing emergency preparedness programs will be performed. These visits will help determine what generic improvements can be obtained through confirmatory research. A handbook describing the criteria for preparing and evaluating the radiological emergency response plan and procedures in support of certain fuel cycle and material licensees will be developed. This handbook will assist State and local governments and licensees in their development of adequate emergency plans in support of fuel cycle and material licensees. Appropriate regulations and regulatory guides will be developed to accompany this handbook. Completion: FY 1984.

An in-depth review of the emergency preparedness responsibilities among Federal, State, and local governments and licensees will be conducted in order to ensure that interfaces are well defined and consistent with the appropriate responsibilities and authorities. A study will be performed to determine the optimum frequency and scope of emergency preparedness exercises. This study will be

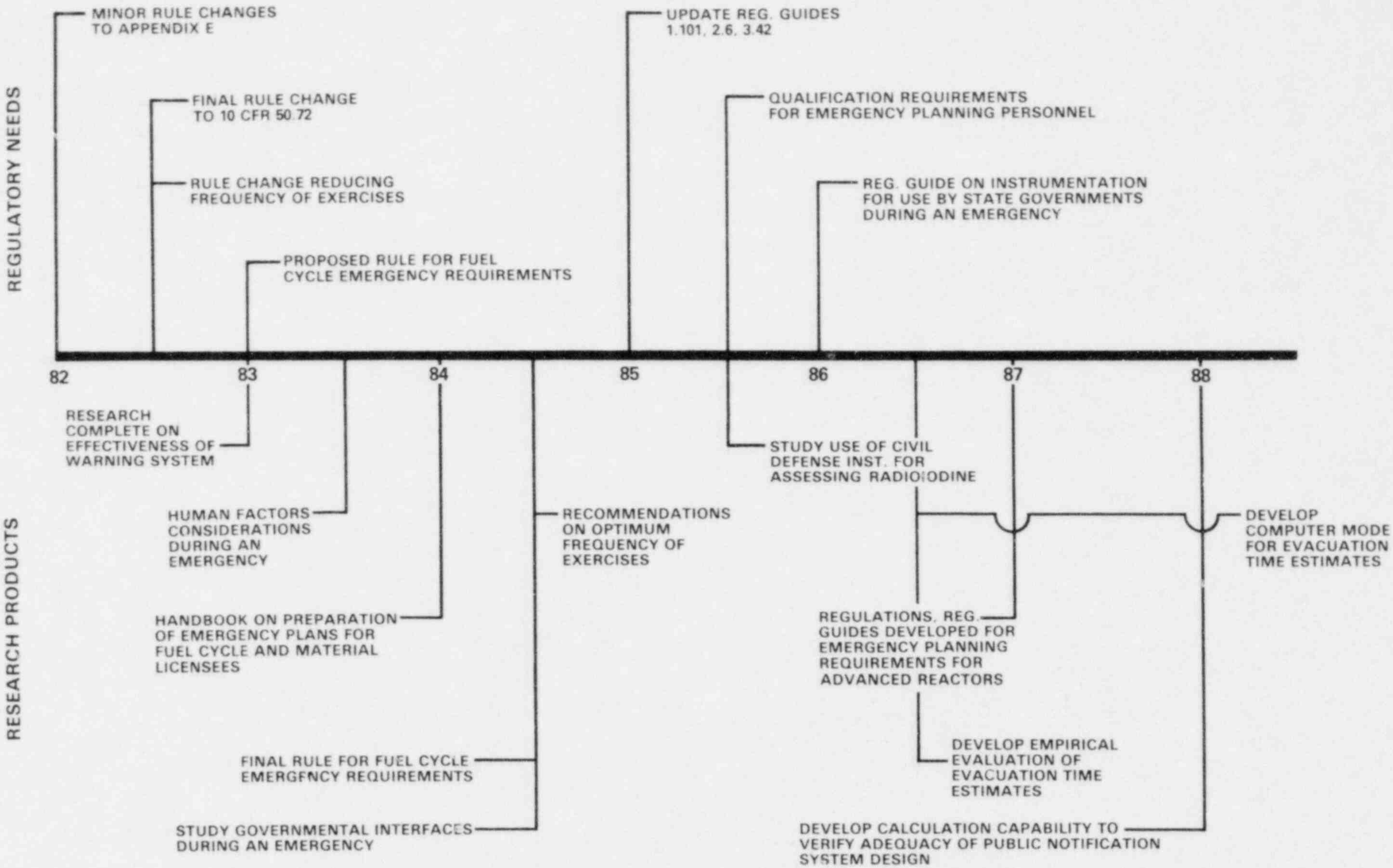

Figure 7.3 Planned availability of research products to current assessment of anticipated regulatory needs for emergency planning

based on governmental observance of numerous exercises at various frequencies and with various degrees of success. A rule change to Appendix E to 10 CFR Part 50 will implement the results of the study. Completion: FY 1984.

Additional emergency preparedness program criteria in the form of regulatory guides, regulations, or national standards based on the review of existing programs will be developed. A study of the types of tasks performed and qualification requirements for emergency preparedness personnel will be performed. Revisions to Regulatory Guides 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," 2.6, "Emergency Planning for Research Reactors," and 3.42, "Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70," will be developed. Completion: FY 1985.

Guidance will be developed for instrumentation systems that can be used by Federal, State, and local governments in responding to offsite radiological emergencies. The use of Civil Defense and commercial types of radiological monitoring instruments for assessing radioiodine by these governmental agencies will be evaluated. A rapid assessment system for measuring radioiodine and particulate fission products in the environment will be developed along with an onsite system to measure and track the airborne plume resulting from an accidental release. The development of the monitoring instrumentation as well as the rapid assessment system will take into account results of research results on fission product sources and behavior. Completion: FY 1985.

Emergency preparedness requirements for advanced reactors will be studied with consideration of different radiological source terms, different reactor designs, and timing of potential release. The results of this study will be implemented in regulations, regulatory guides, and standards. Completion: FY 1986.

Functional requirements in design specifications for an integrated information and data processing system for assessing accident consequences, to be used in the NRC Emergency Operations Center, will be developed. This system will be developed based on the NRC's legislated authorities and responsibilities during a radiological emergency in conjunction with the technical information needs necessary to carry out these functions. Completion: FY 1987.

A calculation capability to verify the adequacy of public notification system designs using licensee design plans and topographical maps will be developed. A computer model for evacuation time estimates that can systemize these estimates for sites with high population densities or with natural disasters or other factors that could complicate the expected evacuation times will also be developed. An empirical evaluation of evacuation time estimates, based on historical events (both radiological and nonradiological) that required evacuation, will be performed, and these estimates will be compared to existing computer code predictions. Completion: FY 1988.

7.5 Safeguards

7.5.1 Issue

Relative to protecting nuclear material from theft and nuclear facilities from radiological sabotage, issues facing safeguards include (1) the need to refine the technical basis used for licensing decisions to provide consistent licensing based on safety significance, (2) the need to develop means of plant-to-plant consistency in the application of all aspects of the safeguards program, (3) the need to develop a method for optimizing the employment of limited licensee and NRC resources while ensuring that an appropriate level of protection is maintained, (4) the need to evaluate alternative methods of providing safeguards that meet current projected performance criteria, and (5) the need to develop and evaluate new safeguards concepts.

Future revisions of this plan will place increased emphasis on such items as new safeguard concepts development, human factor performance, technological developments to support expansion of regional operations, and examination of foreign safeguards technology.

7.5.2 Research Program Objective

The objective of the safeguards regulatory program is to ensure that NRC licensees protect public health and safety and national security by the implementation and maintenance of effective safeguards programs at nuclear reactors and fuel cycle facilities and during storage and shipment of special nuclear material (SNM). To support this general objective, RES is providing support to both the licensing and inspection functions by developing and expanding the safeguards technical data base, implementing the research results in the form of licensee guidance, developing methods to optimize the inspection process, and developing methods to quantitatively measure inspection results.

RES will also review current rules and guidance to improve clarity, to incorporate the results of completed research, and to ensure that a need continues to exist for them.

7.5.3 Relationship to Other Programs

The safeguards research program is closely related to both the licensing and inspection functions within the NRC. Significant effort for the Office of Inspection and Enforcement (IE) is currently under way and is expected to continue at a high level. The link to the licensing program is multifaceted, including the generation of information, as requested by the Program Area Manager, on which to base future policy direction, as well as rule and guidance development in support of established policy positions. Contract work is coordinated through the Safeguards Technical Assistance and Research Group consisting of representatives from NMSS, NRR, IE, CON, and RES. Staff work is coordinated with the requesting office and the Program Area Manager.

Projects are also coordinated with other agencies of the government such as DOE, the Defense Nuclear Agency, the Department of Defense, and the Department of Commerce. This coordination develops normally from information searchs conducted on specific topics and is used in determining the need for further work and the content of that work.

7.5.4 Background and Status

The FY 1982 program will continue work on the development of licensee guidance information. With approximately 60 guides currently effective, a 5-year cycle of review has been established. The total number of guides is expected to be reduced as similar areas are consolidated.

A statistical methods reference manual is to be completed in FY 1982 and endorsed by a regulatory guide in FY 1983. This manual will allow six to ten guides to be reviewed for deletion in FY 1984. A parallel to this program exists in physical security where three guides will possibly be superseded by one addressing locks and hardware, portal equipment, and barriers in a single document. This effort will begin in FY 1982 with publication for public comment in FY 1983 and publication of the active guidance in FY 1984.

Establishment of standard safeguards equipment acceptance criteria, planned in FY 1983, will provide performance-oriented guidance for regulations in FY 1985. Staff review of the safeguards requirements for independent spent fuel storage facilities in FY 1982 will result in recommendations for possible changes to Section 73.50 of 10 CFR Part 73 and supporting guidance.

A long-term program undertaken in FY 1982 for IE will provide quantitative methods of measuring compliance and optimizing limited inspection manpower.

The evaluation methods were previously developed by RES and are currently being reviewed by NMSS. For a decentralized organization, this parallel approach could be important for consistency in both licensing and compliance programs. This program is diagramed in Figure 7.4.

During FY 1982, the majority of work in support of the material control and accounting reform amendments will be completed. In FY 1983, a staff effort will refine and consolidate 3 years of contractor work. The schedule will be consistent with the publication of a proposed and effective rule and is shown in Figures 7.5, 7.6, and 7.7.

The elimination of a backlog of rulemaking actions and petitions is expected to be started in FY 1982 and completed in FY 1983. Major subjects include the authority of NRC inspectors to copy records and test licensee equipment and the review of Section 73.50 of 10 CFR Part 73 for independent spent fuel storage facilities.

A research program to assess the adequacy of human response will be started in late FY 1982. Past assessments have not dealt with the human element and the man-machine interface in depth. This program is shown in Section B of Figure 7.5.

Current research addressing how to minimize the opportunities for insider sabotage at power reactors will be completed in FY 1982; recommendations will be developed for alternative approaches, and the projected costs will be examined. A related project will provide specific cost information for each generic type of nuclear reactor for retrofit of alternative approaches for minimizing the opportunities for insider sabotage. Based on the results of this work, proposed rulemaking or further research may be recommended in FY 1983. Further research will be started, if required, in FY 1983 to address built-in design requirements for power reactors to reduce vulnerability to sabotage by the insider.

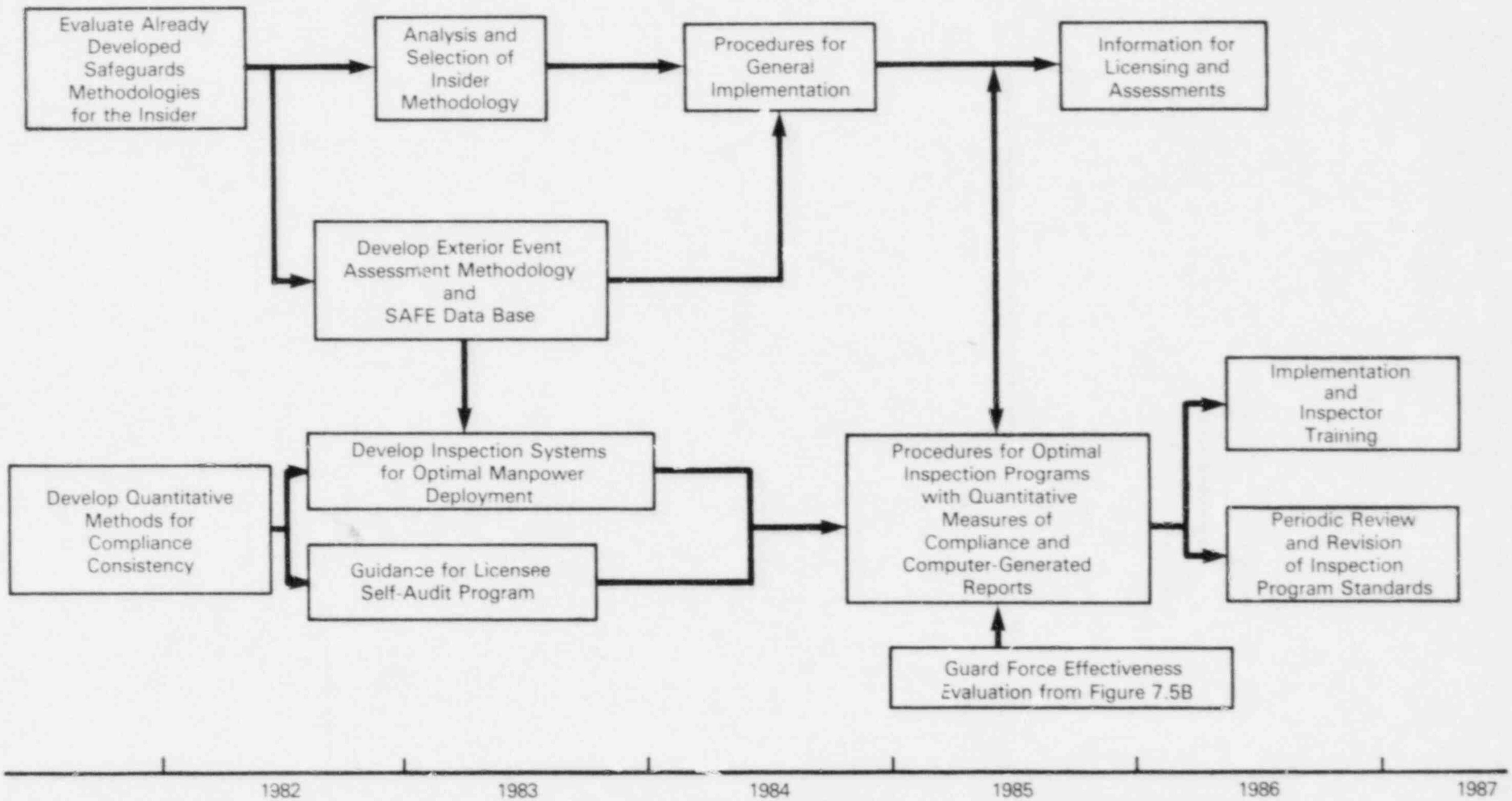

Figure 7.4

SYSTEMS FOR OPTIMAL SAFEGUARDS REVIEW

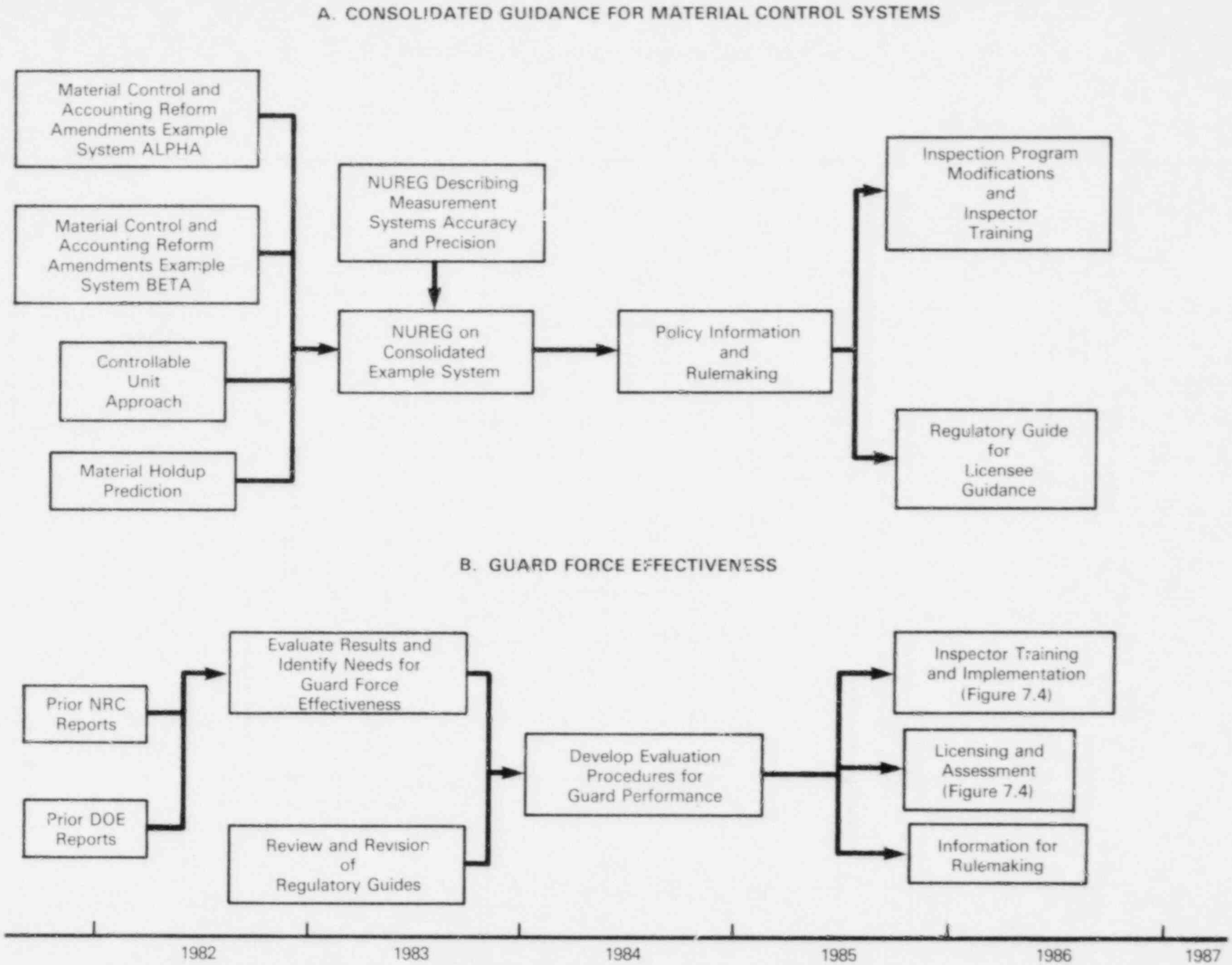
A. CONSOLIDATED GUIDANCE FOR MATERIAL CONTROL SYSTEMS
Material Control and Accounting Reform Amendments Example System ALPHA
Material Control and Accounting Reform Amendments Example System BETA
Controllable Unit Approach
Material Holdup Prediction
NUREG Describing Measurement Systems Accuracy and Precision
NUREG on Consolidated Example System
Policy Information and Rulemaking
Inspection Program Modifications and Inspector Training
Regulatory Guide for Licensee Guidance
B. GUARD FORCE EFFECTIVENESS
Prior NRC Reports
Prior DOE Reports
Evaluate Results and Identify Needs for Guard Force Effectiveness
Review and Revision of Regulatory Guides
Develop Evaluation Procedures for Guard Performance
Inspector Training and Implementation (Figure 7.4)
Licensing and Assessment (Figure 7.4)
Information for Rulemaking
1982
1983
1984
1985
1986
1987

Figure 7.5

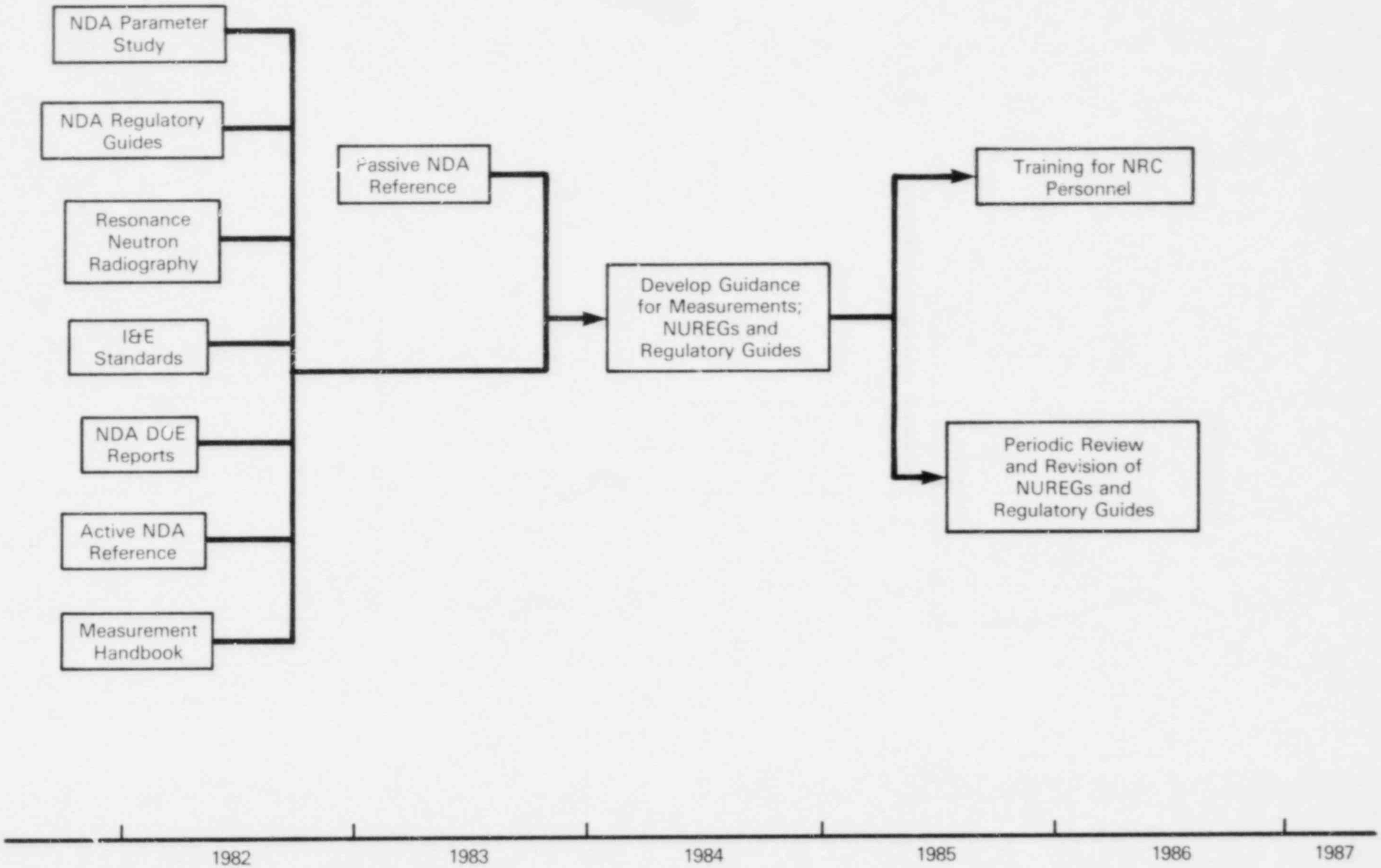

Figure 7.6

MEASUREMENT GUIDANCE PROGRAM

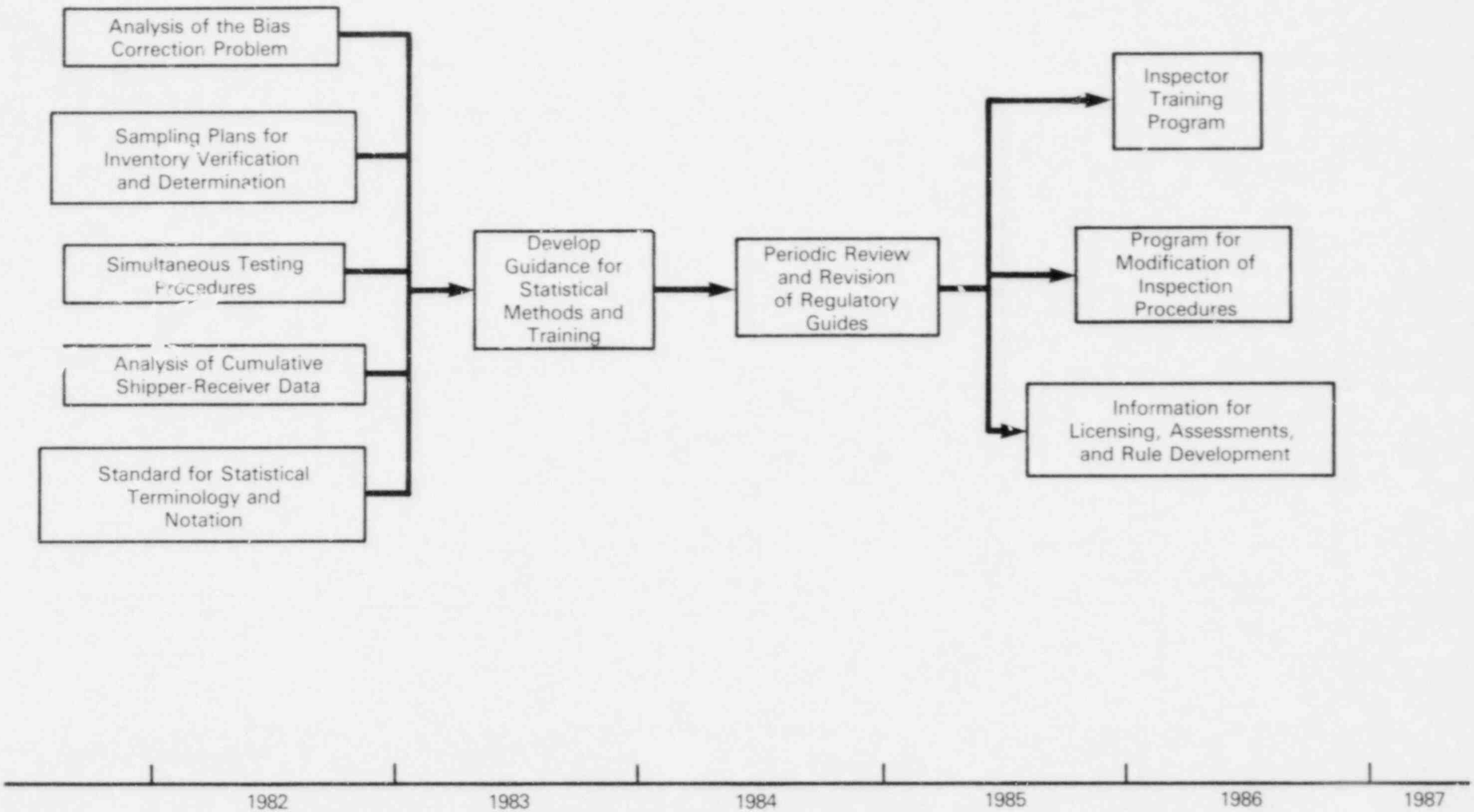

Figure 7.7

GUIDANCE FOR STATISTICAL TREATMENT OF SAFEGUARDS DATA

7.5.5 Research Program Plan

Figure 7.4 shows the planned accomplishments and scheduling related to the full development of a system for safeguards reviews. Each block on the diagram represents a significant and usable product or basis for decisions independent of subsequent products related to it. How the performance of the human factor affects the safeguards system currently in place needs to be examined and evaluated. Section B of Figure 7.5 outlines the program for this issue with completion in FY 1987 followed by integration into the program shown in Figure 7.4.

Figures 7.5A, 7.6, and 7.7 are the established long-term programs currently under way in material control and accounting. These represent the three major areas--measurements, statistics, and integrated control.

A program for updating licensee guidance was initiated in FY 1982 to include guides that have become technically outdated and incomplete as a result of development work completed since their original publication. This is expected to continue until FY 1988. In support of this program, staff effort will include a survey of past work by NRC contractors, DOE contractors, and others. For the FY 1984 to FY 1988 period, this program is expected to occupy fifty percent of the safeguards staff and thirty percent of the contractual effort. Closely tied to this will be the review, rejustification, and technical updating of existing rules.

Included in support of the rule and guidance development process in FY 1984 will be research to assess the need for imposing safeguards requirements on high-level-waste storage facilities and spent fuel storage. Also, for FY 1984, research to review the applicability of current regulations to advanced reactor fuel cycles is to be initiated. Research to resolve those problems identified will continue through FY 1988.

Research on more cost-effective means to protect operating reactors from sabotage by an insider will be continued in FY 1985. This research is intended to evaluate possible tradeoffs among cost, interference with safety functions, and security. This research will support the systematic review of the regulations and will be a basis for licensee guidance.

7.6 Quality Assurance

7.6.1 Issue

Research and standards efforts in the area of quality assurance are to improve the regulatory criteria available for the establishment and implementation of quality assurance program activities at nuclear facilities. Included among the current issues to which these efforts will be applied are:

1. Should the scope of the quality assurance program be expanded to include additional nuclear power plant items, as has been recommended by a number of studies conducted both before and after the TMI-2 accident?

2. What type of quality assurance requirements should be applied to various nuclear power plant items recognizing that factors such as importance to safety, degree of standardization, and ease of replacement indicate that the same quality assurance requirements need not be applied to all items?

3. What should be the qualification requirements for quality assurance personnel?

4. What is the most effective arrangement for reviewing plant activities by onsite or offsite safety review committees?

5. How can the quality assurance aspects of maintenance tasks be improved in order that risks associated with maintenance are reduced?

6. What generic improvements in quality assurance programs can be implemented as a result of evaluating apparently effective programs at existing facilities?

7. Can system reliability techniques that perform a quality assurance function and are used by NASA in the space shuttle program be applied to nuclear power plants? Additionally, can these highly structured reliability analysis techniques improve system reliability to permit their use on a cost-effective basis, and can a handbook be produced to facilitate implementation of these techniques?

8. What should be the criteria for a third-party system of accrediting organizations conducting qualification testing for nuclear power plant equipment?

9. What quality assurance criteria should be applied to the site selection, design, and operation of facilities for storing radioactive wastes?

10. Do quality assurance criteria need to be developed to be used in conjunction with the performance of probabilistic risk assessments?

7.6.2 Research Program Objective

This element's objectives are to provide the technical bases and criteria (which could include national standards and regulatory products) to improve the implementation of effective quality assurance programs at nuclear facilities. The products of these research and standards efforts include detailed regulatory products, including regulations and regulatory guides as well as other documents such as handbooks and NUREG reports. These products will be developed for each of the issues discussed in 7.6.1 above.

7.6.3 Relationship to Other Programs

Detailed coordination of activities in the area of quality assurance usually occurs through research review groups, the standards process, and the exchange of information by the particular task managers. In addition, many research activities in quality assurance are coordinated with the various national standards programs such as those sponsored by the American Nuclear Society and the American Society of Mechanical Engineers.

7.6.4 Background and Status

The TMI-2 Action Plan, NUREG-0660, includes a number of action items under Task I.F, "Quality Assurance." Included in these items are developing guidance and/or requirements to:

1. Expand the listing of items covered by the quality assurance program to include all items important to safety,
2. Ensure the independence of the organization performing the checking functions from the organization responsible for performing the task,
3. Include quality assurance personnel in the review and approval of facility procedures,
4. Include quality assurance personnel in facility activities,
5. Establish qualification requirements for quality assurance personnel,
6. Increase the size of the facility quality assurance staff,
7. Compare NRC quality assurance requirements with those of other agencies,
8. Clarify organizational reporting levels for the quality assurance organization,
9. Clarify requirements for maintenance of "as-built" records, and
10. Define the role of quality assurance in design and analysis activities.

Subsequent to the development of NUREG-0660, additional areas to be included in the quality assurance research and standards activities include:

1. Review and analysis of quality assurance inspection reports to evaluate generic quality assurance deficiencies,
2. Review and evaluate quality assurance aspects of facility maintenance, including a review of maintenance errors, and develop improved quality assurance criteria to reduce maintenance errors,
3. Apply NASA system reliability analysis techniques to nuclear power plant systems, focusing particularly on the quality assurance aspects of such analyses, and
4. Implement the qualification testing laboratory accreditation system.

The program under way in FY 1982 includes revisions to Regulatory Guides 1.28, "Quality Assurance Program Requirements (Design and Construction)," and 1.33, "Quality Assurance Program Requirements (Operation)," in addition to changes to the regulations concerning the applicability of the quality assurance criteria in Appendix B and reporting of changes to quality assurance programs. Additionally, the rulemaking regarding accreditation of testing organizations is being developed.

7.6.5 Research Program Plan

The overall program plan for FY 1984-1988 for quality assurance research efforts is provided in Figure 7.8, which displays the cumulative products of the research efforts. Figure 7.9 displays such products on a single-project basis to identify the project initiation and completion.

An in-depth review of a sample of existing quality assurance programs to determine what generic improvements can be obtained will be completed. A handbook describing a methodology for application to nuclear facilities of system reliability assurance techniques in use at other agencies, e.g., NASA, will be developed. A systematic review of maintenance errors to determine what quality assurance provisions should be invoked will also be completed. Criteria for the implementation of quality assurance programs for the design, construction, and operation of radioactive-waste-storage facilities will be developed. A methodology for identifying and ranking items based on their importance to safety will also be developed. Completion: FY 1984.

Additional quality assurance program criteria in the form of regulatory guides and regulations, based on the review of existing programs will be developed. A study of the types of tasks performed and qualification requirements for quality assurance personnel will be conducted. Revisions to Regulatory Guide 1.28 and the regulations providing requirements for organizations conducting equipment qualification testing will be developed. Also to be developed are criteria for monitoring design activities to ensure that quality assurance provisions are adequately followed. Completion: FY 1985.

The need for a standardized manual for quality assurance practices and procedures will be evaluated. Based on the study of maintenance errors, additional criteria for quality assurance during maintenance activities will be developed. Demonstrations will be performed of the application of system reliability techniques in use at other agencies to nuclear facilities. Also, a revision to Regulatory Guide 1.33 will be developed. Completion: FY 1986.

A new regulatory guide (or revision to an existing guide) on additional qualification requirements for quality assurance personnel will be developed. A regulatory guide incorporating the criteria for monitoring quality assurance during design activities will also be issued. Completion: FY 1987.

Criteria on applying quality assurance program requirements in a graded manner will be developed. Additionally, criteria for improved implementation of quality assurance programs based on reviews of existing programs will be developed. Completion: FY 1988.

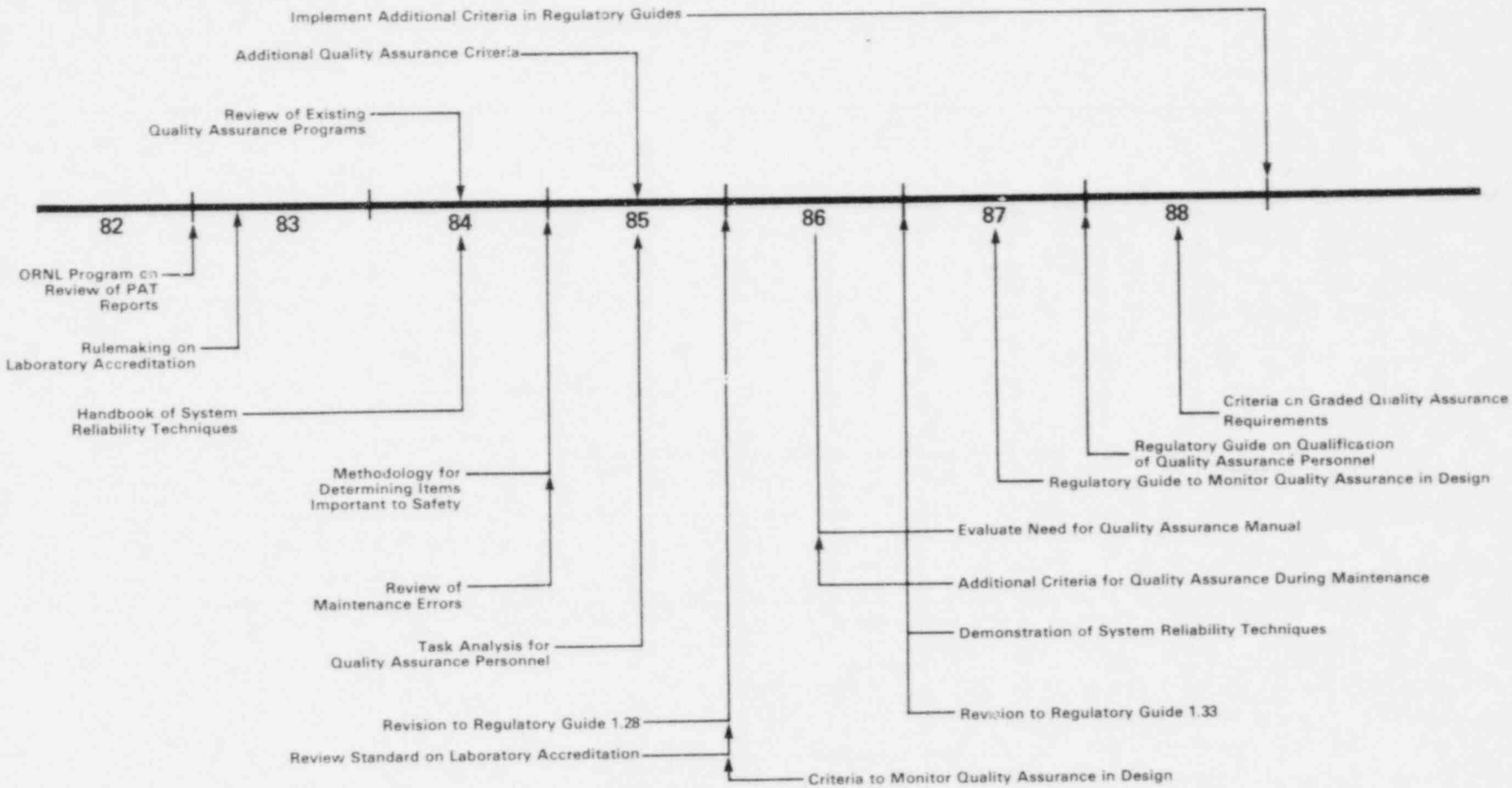

Figure 7-8 Cumulative products expected from quality assurance research

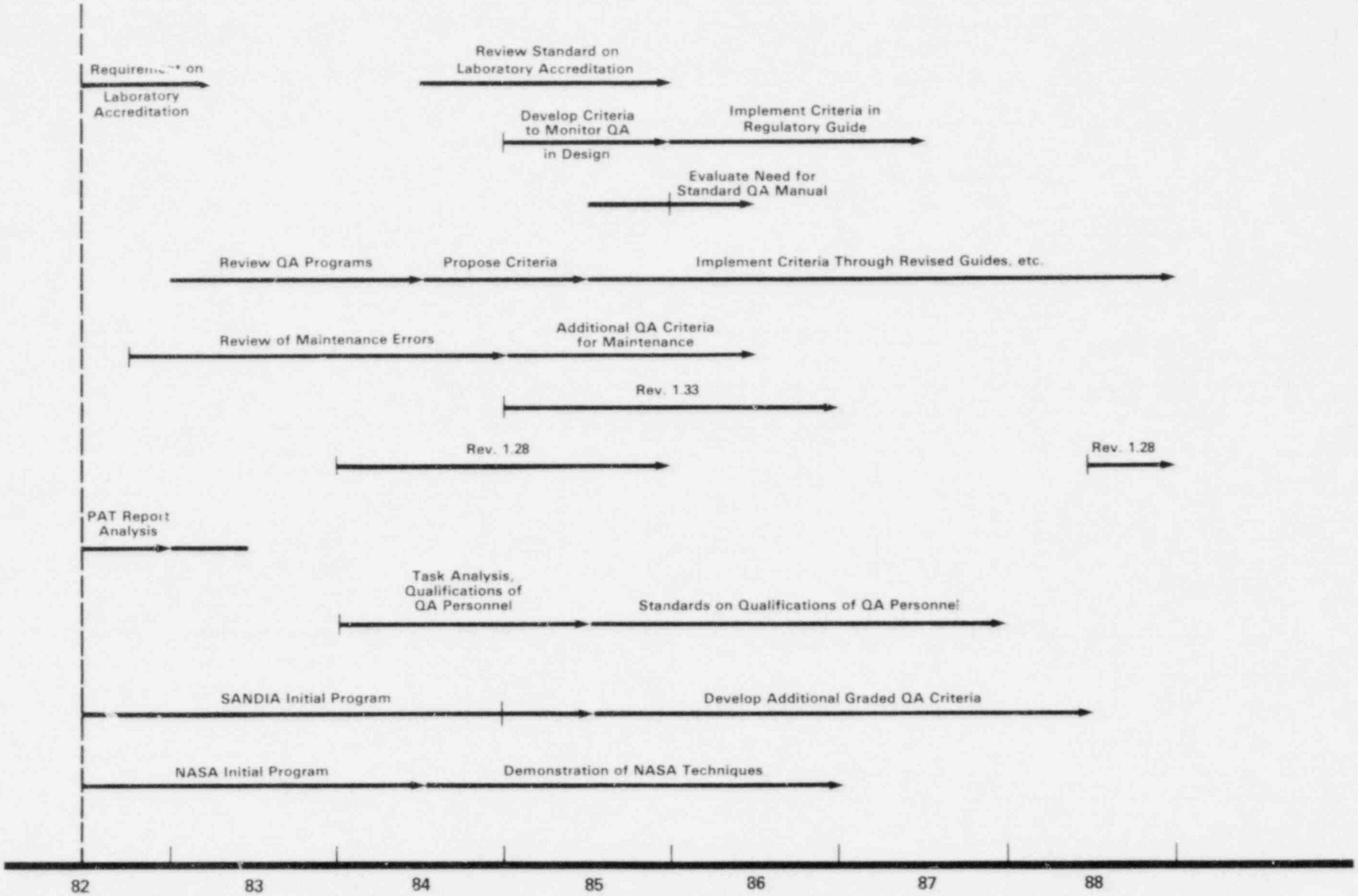

Figure 7-9 Project initiation and completion for quality research

8. WASTE MANAGEMENT

The nuclear waste management program in RES consists of technical research and standards programs to support and improve capabilities for regulating the management of (1) high-level wastes from reprocessing spent nuclear fuel, (2) low-level wastes from nuclear power plant operations and the uses of radioisotopes in industrial, medical, and research applications, and (3) uranium recovery wastes such as mill tailings.

The waste management research program provides information that will be used as technical bases for NRC regulations, criteria, codes, and regulatory guides. The research program results support licensing decisions by providing methods and information needed for the assessment of safety issues and environmental and health impacts. The generic information developed by the research program will improve NRC's capability to evaluate site-specific performance, reliability, and safety of planned or available engineered waste disposal facilities and will improve the capability to assess site monitoring requirements. The waste management research program also focuses on reducing the uncertainties in assessing the risk or safety performance of waste disposal facilities.

Licensing tools such as regulatory guides, codes and standards, and technical directives for the disposal of nuclear wastes are developed in the waste management standards program. Regulations for the construction and operation of nuclear waste facilities are also developed in the standards program.

In summary, the general objectives of the RES waste management program are to:

1. Provide technical bases for protection of the health and safety of the public and workers,

2. Provide technical bases for and develop regulations and standards,

3. Provide technical information and methods to guide and support regulatory decisions, and

4. Confirm the adequacy and reliability of methods and data used by applicants during construction, operation, and decommissioning.

8.1 High-Level Waste

8.1.1 Issue

NRC has the regulatory responsibility for licensing the disposal of high-level waste (HLW) in geologic repositories proposed by the Department of Energy (DOE). This licensing function is carried out in five steps:

1. Reviewing site characterization plan,
2. Issuing construction authorization,
3. Issuing operating license,
4. Regulating during operation, and
5. Decommissioning.

The NRC staff must be able to independently assess DOE's licensing submittals. The HLW research and standards program will provide the tools for those independent analyses. The critical regulatory technical and safety issues for the applicant and thus for the NRC staff to verify are:

1. Prediction and evaluations of:

    a. Ground-water movement,
    b. Interactions of radionuclide with ground water and rock,
    c. Long-term geologic changes and effects on sites, and
    d. Geomechanical changes from construction and operation and their effects on ground-water movement and engineered components.

2. Predictions and evaluations of the extremely long-term performance of engineered components:

    a. Waste package, and
    b. Shaft and tunnel seals.

3. Analysis of operational safety issues:

    a. Safety of operational plan,
    b. Retrievability of waste packages, and
    c. Monitoring.

4. Understanding and reducing uncertainties in evaluating repository performance and risk prediction.

8.1.2 Research Program Objective

The objective of the HLW research program is to provide validated technical information to support an independent assessment by NRC of the proposed site and plan for a deep geologic repository system.

Specific research objectives are (1) to provide information to support an independent assessment of mechanisms and processes that affect long-term waste isolation capability, (2) to identify technical requirements that may be needed to mitigate the consequences of accidental or unplanned movement of radionuclides, and (3) to identify and explore the uncertainties in the data and analytical methods and to develop tools for dealing with such uncertainties, including probabilistic risk assessment.

The objective of the HLW standards program is to provide a regulatory framework for licensing, operation, and closure of HLW repositories.

Specific standards objectives are to:

1. Develop technical directives for construction and operation of HLW repositories; and

2. Develop technical directives for decommissioning.

8.1.3 Relationship to Other Programs

1. Technical Assistance Programs (Sponsored by NRC-NMSS)

   These programs provide state-of-the-art review on various technical issues and identify research needs for licensing and regulation of HLW.

2. DOE Research and Development Programs

   NWTS (Nuclear Waste Terminal Storage), GMIS (Geochemical/Media Interaction Study), and MCC (Material Characterization Center) programs provide a wide range of data on specific site characteristics, geochemical information, and waste packages. These data will provide a major source of information for the licensing evaluation.

3. Domestic Programs

   The Electric Power Research Institute (EPRI) is sponsoring research to develop criteria for and to analyze HLW repositories and to test waste forms.

4. Foreign Programs

   Pertinent foreign programs are French (borosilicate glass performance), German (glass-ceramic waste form, salt repository data), Swedish (multi-barrier waste package design, granitic repository data), and Japanese (borosilicate glass and ceramic waste form) programs. These programs are monitored through information exchange.

8.1.4 Background and Status

The HLW research program provides information and analytical capability required to evaluate DOE's proposed methods for waste treatment and facility design and proposed sites for the storage of HLW. Research is being conducted to assess methods for predicting the occurrence and effects of natural phenomena and long-term geologic processes that may affect the performance of both the natural and engineered components of a geologic repository. During FY 1982 and FY 1983, research will continue on the evaluation of site characterization methods used to evaluate the physical, chemical, and hydrologic properties of rocks in an HLW repository. Research to assess predicting the effectiveness and reliability of methods for plugging and sealing boreholes will be continued and will include consideration of large-diameter shafts and tunnels. Properties of waste forms, containers, and backfill materials will be evaluated with respect to potential long-term containment and controlled release capabilities offered by these materials. The consequences of shaft seal failure on water movement and radionuclude migration under the relatively high-temperature condition in a repository will be assessed. The assessment of the geochemical interactions between waste packages, waste leachates, ground water, and rock will continue, including evaluation of techniques employed to simulate long-term corrosion of waste packages and studies of natural analogues to predict long-term migration of radionuclides from the repository. Also continuing are research to assess methods for predicting long-term ground-water flow through both saturated and unsaturated rock; laboratory-scale tests to predict radionuclide migration through an integrated system, including waste form, breached canister, backfill, and rock; and research on risk methodology to extend the risk methodology developed for bedded salt to include consideration of basalt and welded tuff.

The HLW standards program will complete the revision of the technical criteria of 10 CFR Part 60 during FY 1982.

8.1.5 Research Program Plan

The HLW research programs during FY 1984-1988 will address the following subject areas:

1. Waste form and package performance for the containment and controlled release issues,

2. Near-field migration and geochemistry for the controlled release issue,

3. Repository design, engineering, and monitoring for the occupational safety and waste isolation issue,

4. Site suitability with respect to ground-water transport and geologic stability, and

5. Overall risk methodology.

Ongoing research to understand the failure mechanisms of waste packages and radionuclide release characteristics will be continued. The research will develop the understanding necessary to assess predictions of the longevity of waste package materials under repository conditions. It will quantify the separate important effects in the degradation of waste forms and canister materials. Predictive models that can be used in assessing glassy and SYNROC (titanium-based synthetic rock) waste forms and titanium, copper, and ferrous alloys will be developed during FY 1985. Research to develop information on the reliability and quality assurance of waste package manufacturing will be conducted during FY 1982-1986. Field tests of the models and pertinent parameters will be performed during FY 1986-1988. Studies of long-term physical properties of backfill materials will be extended through FY 1988.

Geochemical research is developing information to support an independent assessment of long-term waste isolation capability. Research initiated in FY 1981 on the importance of irreversible geochemical reactions to the validity of the assumption of equilibrium surface sorption will continue through FY 1985. This research will evaluate the level of conservatism of laboratory measurements of sorption. Current research assesses the validity of using distribution coefficients in radionuclide transport models and assesses uncertainties in the model inputs by comparing retardation coefficients determined from different laboratory methods and from field measurements. Uncertainties in measuring and predicting favorable and unfavorable geochemical parameters in the proposed 10 CFR Part 60 are also being evaluated. This work will continue through FY 1986.

Ongoing research determines the solubility of different chemical species of radionuclides at the temperatures and pressures expected in repository rock and in the engineered structure. The solubility approach is a promising alternative to the retardation (kd) approach for assessing the release rate of radionuclides from the repository. The solubility research will be conducted during FY 1984-1988.

The assessment of the migration of naturally occurring radionuclides from ore bodies over very long time periods (which is a natural analogue to radionuclide migration from an HLW repository) will continue through FY 1987.

Ongoing research through FY 1984 in geophysical methods will produce information to support an independent assessment of geologic and hydrologic site characteristics. A study to confirm the reliability of instruments used to characterize sites and measure rock properties during all phases of repository development will also continue through FY 1984.

Research that assesses the effectiveness of geophysical techniques for identifying geologic structures and hydrologic and stratigraphic boundaries will be largely completed in FY 1986. Related research that will continue beyond FY 1986 will include assessment of the state of the art of methods to characterize faults and joints, including their genesis, surface texture, geometric distribution, nature of infilling, and degree of interconnection. 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 be investigated. Research to identify critical parameters that need to be monitored will be continued through FY 1988. This research will also evaluate the effectiveness of the monitoring instruments that will be used during construction and operation following closure of a repository.

Ongoing engineering geologic research is evaluating the effectiveness of methods for characterizing sites and for assessing mechanical and chemical properties of rock with minimum disruption of the rock. Emphasis will be given to the limitations and uncertainties in extrapolating data to predict repository performance. The effects of heat on rock properties such as strength, rigidity, integrity, chemical reactivity, and adherence to sealing materials will be studied to permit NRC to independently assess the stability of underground openings and assess uncertainties introduced by heat. Evaluations of methods to predict the thermomechanical response of the rock mass, including field tests of their validity, will follow the research on the thermal conductivity of rock and will continue through FY 1987. This research will provide technical bases for verifying thermal-loading criteria for HLW repository designs.

Research will continue through FY 1986 on the measurement of ground-water flow relative to the potential migration of radionuclides under both saturated and unsaturated conditions. This research will include the determination and field-testing of theories governing ground-water movement through fractured rocks. The effects of advection, diffusion, and dispersion on radionuclide transport in ground water, including changes induced by heat, will be evaluated.

Nonradioactive tracers than can be used in hydrologic transport tests and that can be placed in a repository for monitoring will be identified.

The capability for predicting long-term climatic, geologic, and hydrologic changes that have 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.

A small research effort will be initiated in FY 1985 to assess the probable effects of a repository on the economic and social structure of the surrounding local area. The effort will be continued through FY 1988.

In situ verification tests of pertinent parameters for waste isolation will be initiated during FY 1984 and continued beyond FY 1988. This task is contingent upon the availability of DOE's in situ facilities for NRC contractors.

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 1988. The results will be coordinated with other projects in the HLW program to reduce the uncertainties in performance prediction and risk assessment.

8.2 Low-Level Waste

8.2.1 Issue

The regulatory responsibilities of NRC in low-level waste (LLW) are to:

1. License shallow-land burial sites,
2. Assess alternative methods of LLW disposal,
3. Assess waste form and package and performance of waste package,
4. Evaluate public safety and environmental impacts (per proposed 10 CFR Part 61), and
5. Assist Agreement States upon request in the regulation of LLW disposal.

The critical regulatory technical and safety issues for the LLW research and standards program include:

1. Determining appropriate methods for operating and decommissioning LLW disposal sites,

2. Identifying biogeochemical uncertainties that may affect the radionuclide retention capability of shallow-land burial facilities,

3. Characterizing nonradiological toxic substances associated with radiological wastes in LLW disposal facilities,

4. Assessing the characteristics of volume-reduced wastes, and

5. Evaluating methods for predicting the integrity and containment capability of the waste package.

8.2.2 Research Program Objective

The general objectives of the research program are to improve the NRC capability to predict LLW isolation performance and to provide a better technical basis for regulatory standards and risk assessment.

The specific objectives of the LLW research program are to:

1. Assess predictions of long-term performance of LLW disposal sites,

2. Assess methods and establish criteria for short- and long-term monitoring of LLW disposal sites,

3. Assess waste form and package performance,

4. Assess needs and methods and establish requirements for decommissioning LLW disposal sites, and

5. Provide technical assistance upon request to Agreement States.

The objective of the LLW standards program is to provide a regulatory framework for licensing, operation, and closure of LLW burial grounds. This program will develop licensing tools such as regulations, regulatory guides, and technical directives.

8.2.3 Relationship to Other Programs

1. Technical Assistance Programs (sponsored by NRC-NMSS)

   These programs provide state-of-the-art reviews on waste forms and siting issues and also identify research needs. Research programs will follow through on those issues and needs.

2. DOE Research and Development Programs

   These programs provide engineering and operational information on LLW facilities and data on alternative methods.

3. Domestic Programs

   EPRI is sponsoring research on criteria and site selection, on the characteristics of plant wastes, and on the characteristics of LLW forms.

4. Foreign Programs

   European and Japanese programs provide information on waste form performance.

8.2.4 Background and Status

The major emphasis in the LLW research program for waste forms is on assessing the characteristics of solidified wastes arising from reactor accident cleanup operation and from the routine decontamination of operating reactors. This program includes an assessment of the effects of chelating agents found in decontamination agents that may enhance radionuclide migration from shallow-land burial sites. Work will be initiated to compare past test results obtained from simulated waste forms with real full-size forms to evaluate the effectiveness of proposed standard tests of waste forms. Research will be continued to define the characteristics of volume-reduced wastes produced by DOE and industry.

The major emphasis in the LLW research program on engineering design and practices is the evaluation of the long-term confinement capability of containers and of the design, construction, and operating practices of shallow-land burial facilities. Ongoing studies of existing shallow-land burial sites will continue to assess methods for measuring, analyzing, and predicting waste retention and transport in soils. This information will be used to improve both siting and decommissioning criteria. In FY 1983, work will provide for completing tests of solidified wastes generated from two nuclear power plants (one PWR and one BWR) with respect to leachability and compressive strength.

Studies initiated in FY 1982 to assess risks associated with the performance of LLW facilities will be continued.

The LLW standards program will include publishing regulatory guides on waste form, site closure, stabilization and postoperational care, funding for closure, and facility design, operation, and monitoring.

8.2.5 Research Program Plan

Ongoing research to test and improve procedures for evaluating and monitoring the long-term performance of the waste forms will be concluded in FY 1985. Properties of volume-reduced wastes such as incineration ash and acid digestion will be studied through FY 1987. Tests of offnormal wastes such as wastes generated from accidents and decontamination will be conducted through FY 1988. Integrated laboratory-scale tests of waste forms and near-field media will be performed during FY 1982-1987. Results of these tests will serve as laboratory verification of near-field migration models. Testing of proposed high-integrity containers will be continued through FY 1984.

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.

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 and will be continued beyond FY 1988.

By FY 1984, an assessment of geologic alternatives to shallow-land burial will identify those that are feasible and cost effective.

Research on shallow-land burial and geological alternatives to shallow-land burial will include (1) assessing improvements in sites and facility monitoring, (2) assessing the effectiveness of designs and procedures for closing sites, and (3) identifying ways to mitigate the consequences of accidental releases of radionuclides. These efforts will be continue through FY 1988.

In FY 1985, field tests for each disposal method will provide a basis for a comparison of the alternatives. This research will consider site suitability, facility design and operation, monitoring, and facility closure.

During FY 1985-1986, the emphasis will change from research on shallow-land burial to research on burial in deeper geologic media. During FY 1987-1988, a feasibility and cost-effectiveness study for engineered storage will be initiated. The results of research on site characteristics for the HLW repository are expected to make a significant contribution to research on mined cavities for the disposal of LLW.

Field tests at existing commercial shallow-land burial facilities will be conducted through FY 1984 in coordination with supporting laboratory tests and experiments. These tests will include (1) assessment of erosion rates, (2) characterization of degraded wastes in trenches, (3) hydrologic measurements in the unsaturated zone, (4) comparison of distribution coefficients for different chemical species of radionuclides in different geologic media, (5) assessment of nonradioactive components of wastes (such as chelating agents that may influence radionuclide migration), (6) assessment of the effectiveness of monitoring methods, and (7) assessment of the effects of vegetation on radionuclide containment or migration. The data obtained from the field work will be used to test predictive models and to identify characteristics of sites and their environs important to the control of radionuclide migration. This research will be continued beyond FY 1988.

Ongoing research on the effectiveness of geotechnical, radiological, and environmental measurements systems used to characterize and monitor LLW sites will continue through FY 1984.

Research on methods to assess risks to the public and the environment from LLW facilities will be completed in FY 1986. The ongoing risk research concentates on site environmental conditions and waste forms. Future research may be required on the waste package and its effect on the overall risk at an LLW site.

Standards, regulatory guides, and technical directives will be developed during FY 1984-1988 for licensing and regulating engineered LLW facilities.

8.3 Uranium Recovery

8.3.1 Issue

The regulatory responsibilities of NRC are to:

1. License active milling and in situ extraction operations,
2. Concur in the acceptability of DOE remedial-action plans for inactive sites, and
3. License DOE to maintain inactive sites following remedial actions.

The critical regulatory technical and safety issues for the uranium recovery (UR) research and standards program are to assess methods for:

1. Preventing erosion and radon release from tailings piles,
2. Evaluating the effectiveness of interim stabilization techniques,
3. Reducing seepage into ground water and preventing ground-water degradation,
4. Controlling in situ mining solution movement, and
5. Monitoring releases to environment from UR facilities.

8.3.2 Research Program Objective

The objective of the UR research program is to provide validated technical information to support independent safety and site suitability evaluations for licensing active milling operations, inactive tailings, and in situ extraction operation..

Specific research objectives are to test and evaluate methods to assess:

1. The effectiveness, benefits, and costs of various engineering and milling process and disposal alternatives for reducing tailings impoundment seepage of toxic and radioactive materials from deep mine or pit disposal into ground water,

2. Subsurface physical conditions for short- and long-term migration of toxic and radioactive constituents of seepage,

3. The in situ mining solution movement and available restoration techniques,

4. Release of radioactive and toxic materials from facility components, the subsequent environmental transport of the materials, and their impacts,

5. The reliability, durability, and cost effectiveness of interim stabilization techniques for reducing particulate suspension and radon gas exhalation from exposed tailings surfaces,

6. The effectiveness and long-term reliability of clay caps and thick earthen covers for reducing radon gas releases,

7. The practicality, effectiveness, and reliability of rock covers to control surface erosion of tailings,

8. 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. The instruments, techniques, and procedures for verifying the acceptability of site decontamination, decommissioning, and reclamation before license termination.

Specific standards objectives are to develop criteria, technical positions, guides for implementing regulations, and revisions of regulations that may be needed to improve the regulatory management of UR wastes.

8.3.3 Relationship to Other Programs

1. Technical Assistance Programs (sponsored by NRC-NMSS)

   These programs provide state-of-the-art reviews on tailings impoundments and identify research needs through case studies.

2. DOE's UMTRAP (Uranium Mill Tailing Remedial Action Plan)

   This program provides data from closed UR facilities.

3. Domestic Programs

   EPRI is sponsoring a $4.5 million program in FY 1983 to characterize geologic and hydrologic parameters for uranium facilities and disposal methods.

8.3.4 Background and Status

The UR research program provides information and analytical capability required to evaluate (1) license applications from commercial vendors involved in the extraction of uranium from naturally occurring ores and (2) proposals from DOE to perform remedial actions at inactive uranium mill sites under the provisions of the Uranium Mill Tailings Radiation Control Act. Major areas of research for the FY 1982-1983 UR program and beyond are:

1. Mechanics for contamination of ground water from conventional milling and disposal operations and from in situ mining,

2. Interim and long-term reclamation and stabilization of tailings, and

3. Environmental monitoring methods and techniques.

8.3.4.1 Ground-water Contamination

Assessing the interaction of mill tailings with ground water requires research to study methods for detecting and measuring the seeping of liquid radioactive waste from tailings ponds into surface and ground waters and to study methods for controlling such seepage. This research, begun in FY 1981, will continue through the planning period and includes assessment of the effectiveness of various neutralization techniques to decrease the mobility of potential ground-water contaminants. Field tests to determine accuracy of flow models for determining impact of tailings on ground water will be initiated. Research will continue to assess engineering methods for disposal of tailings in contact with ground water.

8.3.4.2 Reclamation and Stabilization

Assessments of methods for stabilizing active operational tailings began in FY 1981 and will continue through the planning period. These efforts include completing assessment on the scope, feasibility, and design of experimental studies on rock resistance to weathering and optimum placement of riprap.

Finally, research on the properties of covers for attenuating radon emanation from uranium tailings piles will be completed.

8.3.4.3 Environmental Monitoring

Research on monitoring methods and equipment, including assessing releases of radioactive and toxic materials, began in FY 1981 and will continue through the planning period. Radon flux comparisons with radium concentrations in surface and subsurface soils will be completed in FY 1983.

The UR standards program will develop and publish regulatory guides on stabilization and erosion protection of uranium tailings disposal piles; on determining compliance with site decommissioning, cleanup of contaminated lands, and remedial action criteria for contaminated structures; and on covers for reclaimed uranium mill tailings disposal sites.

8.3.5 Research Program Plan

8.3.5.1 Ground-Water Contamination

The tailings dewatering and neutralization research projects and the leachate movement project will be conducted through FY 1987. These projects will provide information on (1) the reduction of liquid wastes from milling operations, (2) immobilization of toxic materials in tailings, and (3) movement of waste liquids in soils. Knowledge of these fundamentals are needed to minimize and prevent potential ground-water contaminations.

Evaluation of the linear consolidation theory of tailings dewatering will be completed in FY 1983. Measurements of tailings and soil characteristics and the study of materials and installation procedures for dewatering will also be completed in FY 1983. Tailings neutralization and alternative methods for immobilizing toxic materials in tailings will continue until FY 1986.

The identification of alternative fixation methods and materials and laboratory analyses will be completed by the end of FY 1984. Field tests of alternative fixation methods and materials will then be initiated and continued through FY 1987. Ground-water consequence evaluations, started in FY 1982, will be followed by actual field evaluations of leachate movement in FY 1983 and 1984. Models for predicting such movement will be developed in FY 1984.

8.3.5.2 Reclamation and Stabilization

Ongoing research to select and test candidate soil materials for covering uranium mill tailings will be concluded in FY 1984. The evaluation, assessment, and development of information on the specifications and proper installation of riprap and other materials for long-term protection of mill tailings from water and wind erosion will continue until FY 1985. Results of these research projects will provide the basis methodology and the standardized procedures for calculating the radon attenuation provided by covers placed over uranium mill tailings systems. They will also provide the technical basis for protecting the tailings systems from wind and water erosion from the long-term standpoint.

The interim stabilization research project should be completed in FY 1984. Results of this research will support enforcement by NRC of EPA's environmental standards contained in 40 CFR Part 190 that are generally applicable.

8.3.5.3 Environmental Monitoring

Research of effluent and environmental monitoring methods, equipment, and instrument testing will continue until FY 1985. Initiated in FY 1982, the field comparison of radon progeny measurement techniques should be completed by FY 1985. The evaluation of long-term radon daughter measurements will continue until FY 1984. The foregoing research is expected to result in the development of field-testing protocols for surveys of uranium mills and surrounding environs for decontamination and decommissioning operations.

9. SITING AND ENVIRONMENT

The siting and environmental impacts program includes the development of the technical basis to support NRC regulatory requirements and a standards activity. This program addresses the environmental aspects of the siting of nuclear facilities and the protection of the public from harmful effects of radioactivity in licensed nuclear facilities.

Specifically, this program includes (1) development of methodologies and verified predictive models for the systematic analysis of sites for the protection of public health and safety and the environment; (2) development of generic data on facility siting parameters for site characterization; and (3) development of dosimetric information bases and methodologies for the protection of the public and the worker from potentially harmful effects from ionizing radiation and from the use of radioactive materials in licensed nuclear facilities.

9.1 Siting and Environmental Impact

9.1.1 Issue

Protection of the health and safety of the public and of the environment requires that NRC have the capability to evaluate site suitability for licensed nuclear facilities. The siting standards and criteria program provides information and guidance needed to ensure that both proposed and alternative sites can be systematically analyzed and assessed from the perspectives of public safety and protection of the environment. This involves comparing the environmental and physical characteristics of candidate sites and assessing the effects of nuclear facilities at the site on public health and safety and on the quality of the environment. To date, the siting and environmental program has been concerned principally with nuclear power stations. However, during the time period covered by this plan, issues arising from the siting and operation of other nuclear facilities, especially low-level radioactive waste repositories and major byproduct material and nuclear fuel reprocessing facilities, will assume increasing importance.

Specific issues to be addressed include (1) determining the criteria to be used in the siting of nuclear facilities, including power reactors and low-level waste facilities; (2) assessing the environmental impacts of nonreactor nuclear facilities; (3) using remote-sensing and mapping techniques in site evaluation; and (4) determining the socioeconomic effects of a range of nuclear power plant accidents.

The principal objective of the National Environmental Policy Act (NEPA) of 1969 is to build into the Federal decisionmaking process appropriate consideration of environmental impacts of proposed actions. Implementation of NEPA requirements by NRC necessitates the development of information and methods through research to provide for assessing impacts of the proposed licensing action and alternatives to it. Information and data developed through research are used by the NRC staff to prepare hearing testimony and environmental impact statements.

9.1.2 Research Program Objective

The broad objectives of this program are to ensure (1) that the NRC possesses the methodology to make acceptably accurate, independent assessments of the projected safety and environmental impacts from proposed nuclear facilities and (2) that there is an acceptable body of knowledge upon which to base environmental regulatory evaluations and requirements.

One 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.

Another specific objective of the research program is to support site evaluation and selection as well as standards development. This research will provide methods for comparing proposed alternative sites on the bases of public health and safety and environmental impact and will provide validated technical bases for regulatory decisions in the siting and environmental area.

Another specific objective of the research program is to provide for transfer to the States of the capability to perform economic and power-demand assessments.

9.1.3 Relationship to Other Programs

The NRC siting and environmental research and standards program is coordinated with work being conducted by the Environmental Protection Agency (EPA), the Department of Energy (DOE), the Electric Power Research Institute (EPRI), and State governments. Research on Legionnaires' Disease bacteria in closed-cycle cooling systems is being coordinated with an EPRI project dealing with this topic. DOE is participating in the review of research to determine the impacts of copper on aquatic biota. Need-for-power modeling work is primarily coordinated with State governments and to a lesser degree with DOE and EPRI. Work in the siting area is being coordinated with other programs in RES that are reevaluating the fuel damage and fission product accident source term.

All siting and environmental research sponsored by NRC is published in technical reports and made available to interested public and private organizations for information and comment. In addition, contractors conducting research are encouraged to publish the results of their work in professional journals and to seek other peer review in academic and professional forums. Open conferences and workshops on siting and environmental issues are sponsored by NRC to permit interaction with the technical community. In turn, the NRC staff in the siting and environmental area maintain close contact with relevant work being done outside NRC.

9.1.4 Background and Status

Programs presently under way in siting and environment include (1) demographic studies, (2) societal impacts studies, (3) development of assessment models and measurement methods to characterize environmental impacts, (4) the transfer to States of the economic assessment capability, and (5) improvement in dose assessment.

The technical evaluation of the bases for reevaluating demographic criteria for reactor siting will be issued in FY 1982. After the accident source-term reevaluation on the siting work has been assessed, demographic siting criteria will be prepared as a proposed rule in FY 1983 with a final rule issued in FY 1984. A number of current activities are providing backup for this effort.

Studies are being conducted to evaluate population densities at nuclear plant sites, site availability, population changes in the immediate vicinity of existing nuclear power plant sites, and State and local governmental policies for dealing with postlicensing land-use change. An initial evaluation is being completed to determine if siting criteria for nonreactor facilities are needed.

Other studies to improve the basis for site selection and alternative site review are being carried out. A test of a procedure for obtaining inputs from States on a regional basis in the alternative site review process is being conducted in the Southeastern region, focusing on a low-level radioactive waste facility.

As a part of the transfer of economic assessment capability, work to enhance the technical capabilities of States to participate in need-for-power reviews is under way. Information on the use of the NRC-developed SLED (State-level electricity demand) model is being provided to interested States.

Research to validate mathematical models for predicting the transport of radionuclides on sediment in rivers is to be completed in FY 1983. Also, research is being carried out on bioaccumulation of phosphorus-32, and the safety implications of aquatic-fouling organisms. Studies are in progress to evaluate the effects of copper releases from nuclear stations on aquatic biota in cooling lakes. Remote-sensing techniques, using aerial and satellite reconnaissance, are being tested for utility in environmental monitoring with emphasis on land use and land cover surveillance.

A study of the socioeconomic impacts of the construction and operation of nuclear power plants is being completed. The probable economic consequences of a variety of types of nuclear power plant accidents are being assessed. A quality assurance program for radiation measurements in the environment is being developed. Revised criteria for defining an "extraordinary nuclear occurrence" will be proposed in FY 1982.

Studies of manmade offsite hazards are being conducted to determine the need for regulatory guidance in this area.

In FY 1982, revisions to incorporate public comments will be made to regulatory guidance on dose assessment methods for uranium milling operations. Efforts to analyze NRC postaccident environmental monitoring requirements will also begin in FY 1982. Efforts have been initiated in FY 1982 to develop criteria for the unrestricted disposal of materials containing very low levels of radioactive materials.

9.1.5 Research Program Plan

9.1.5.1 Site Evaluation and Selection

Implementation guidance on demographics and external hazards will be developed during FY 1984-1986. Following issuance of the proposed demographic siting rule in FY 1983, revisions and maintenance of the rule will be necessary through FY 1988. Revision and maintenance of Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," will be carried out through FY 1988. The alternative site rule for reactors will be completed in FY 1982, with maintenance activity scheduled through FY 1988.

If studies now under way indicate that siting criteria for nonreactor facilities are needed, they will be developed in FY 1983-1984 along with appropriate regulatory guides in FY 1985.

It is required that nuclear power plants have the capability to be safely shut down in the event of severe manmade phenomena. Consequently, safety-related structures, systems, and components must be designed to withstand the effects of such phenomena. The assessment of the contribution to the overall risk to the public from severe manmade-phenomena-induced failures requires information on magnitude, frequency of occurrence, and distribution of severe manmade phenomena. These phenomena include such potential hazards as airplane crashes, flammable liquids and gases, explosives, toxic and corrosive gases, and forest fires.

Programs to advance the methodology for site selection and siting criteria for reactors will be carried out in FY 1983 and FY 1984. Regional approaches to alternative site review and increased use of States' inputs in the environmental aspects of siting nuclear facilities will be emphasized. A NUREG report will be issued in FY 1984. Building upon present activities in the need-for-power reviews for reactors, programs to provide technical bases to assist States to participate in the environmental reviews involved in site selection will be undertaken. Activities related to low-level radioactive waste facilities will also be initiated.

An existing program to gather and evaluate data on postlicensing population and land-use changes at nuclear power plant sites will be continued through FY 1988 to include new plant sites as they become operational. Results will be incorporated into reports, scheduled for FY 1985 and FY 1987, to assist in the implementation and maintenance of nuclear reactor siting policies and standards.

9.1.5.2 Generic Environmental Assessments

While environmental impact work for light-water reactors will be largely completed by FY 1983, present indications are that licensing activities centering on other aspects of the nuclear fuel cycle will be increasing. To accommodate information needs for licensing and to maintain adequate standards, programs to identify and analyze types of environmental impacts stemming from fuel reprocessing and low-level radioactive waste repositories will be undertaken. Where existing environmental impact assessment methodology for reactors is not transferrable, research and standard-setting activities will be started in FY 1984 and completed by FY 1988.

During FY 1984-1988, research will be conducted to resolve discrepancies in existing information on the transport of radionuclides and to address problems identified in the licensing process. This work, as well as literature reviews, will be reflected in revisions to guidance on dose assessment for reactors and fuel cycle facilities. Efforts to provide greater uniformity in approaches and parameters used thoroughout the agency for dose assessment will continue through the planning period. Interim reports will be prepared for use by the licensing staff beginning in FY 1985.

During FY 1984-1988, the NRC environmental monitoring requirements and the data provided by licensees in conformance with these requirements will be reexamined. These data will be analyzed to determine their relationship to facility operations and their value in assessing the radiological impact of these operations. This effort will supplement the review of postaccident environmental monitoring carried out in FY 1982. Reports are scheduled to be available beginning in FY 1984.

9.1.5.3 Societal Impacts

Research on socioeconomic impacts of licensed nuclear facilities has concentrated on the NEPA-related socioeconomic information requirements arising out of the licensing process for nuclear reactors. In the recent past, reactor licensing has generated needs for generic research on the need for power, the socioeconomic impact of construction of nuclear facilities, and related topics. Research projects already completed or to be completed in FY 1982 are expected to fill gaps in knowledge and give NRC the tools it needs.

Much of the research on the socioeconomic impact of reactors will be of use in support of regulatory decisions on low-level waste. However, because of the different characteristics of waste facilities, the impact of these waste facilities on the local economy and public services and the land-use and land-value impacts need to be examined. Licensing hearings on waste facilities will probably be as involved and contested as some prior reactor licensing cases have been. NRC needs such impact information on a timely basis in order to avoid the costly licensing delays that are brought about by lack of technical information on NEPA-related licensing issues. Beginning in FY 1984, research to permit the assessment of the socioeconomic impacts of radioactive waste facilities will be started. Starting in FY 1985, a series of reports will be delivered to the licensing staff documenting the information derived from this research.

A controversial factor in recent reactor licensing hearings has been the need for nuclear power plants and the relative economics of nuclear vs. coal and other forms of electric power generation. NRC has developed several computer codes and models to assess these issues. The SLED modeling system, which forecasts need for power, and the CONCEPT and OMCOST codes will be maintained and data bases updated as necessary during FY 1984-1988. Assistance will be provided to States to acquire and maintain the capability to use these tools, thus encouraging States to assume much of the responsibility of need-for-power assessments now borne by NRC. The current schedule calls for at least ten States to be assisted by FY 1985.

9.2 Earth Sciences

9.2.1 Issue

The technical and safety issues addressed by the seismology/geology research program concern the uncertainties in estimating the occurrence and severity of vibratory ground motion (VGM) at nuclear facilities. Present NRC criteria (Appendix A to 10 CFR Part 100) determine the design basis for VGM using a correlation of seismicity with tectonic structures or using the tectonic province approach. The regulatory issue is that there are uncertainties in estimating ground motion at nuclear facilities. These uncertainties result from ambiguities (1) in determining the correlation between seismicity and structure, (2) in defining tectonic provinces, and (3) in estimating the magnitude of potential ground motion due to a fault. Additionally, there are uncertainties in correlating tectonic province with seismic hazard.

The purpose of the seismology/geology research is to reduce the uncertainties of seismic risks in siting nuclear facilities. Also, the improved data base will support the systematic evaluation program.

The technical and safety issues addressed by the meteorology research program include (1) the uncertainties in existing methods for predicting the movement of effluent plumes; (2) the extension of the distance over which existing models are applicable; and (3) the uncertainties in the spatial and temporal characteristics of severe weather phenomena. Guidance documents incorporating the results of this program will be developed and issued as the information becomes available through FY 1986.

The technical and safety issues addressed by the hydrology research program are (1) the assessment of flood determination methodologies, including the likelihood of failure of flood projections and (2) the evaluation of ground-water transport and interdiction methods for containing releases resulting from postulated core-melt accidents. By FY 1987, the ground-water studies should serve as a technical base for future siting rulemaking actions.

9.2.2 Research Program Objective

The objectives of the seismology/geology research are to provide (1) improved bases for licensing decisions and for development of standards applicable to the safety of nuclear facility sites and (2) authoritative information to assist in revising the siting criteria. These siting objectives apply to new facilities such as nuclear power plants and high-level waste facilities and to the systematic evaluation of existing power plants.

The objectives of the meteorology research program are (1) to obtain effluent dispersion information to reduce uncertainties and (2) to extend the range of plume movement models. Severe weather studies will realistically determine maximum characteristics and spatial/temporal variations.

The objectives of the hydrology research program are (1) to develop the basic information needed to establish pertinent evaluation criteria for the hydrological characteristics of existing or future nuclear facility sites, (2) to address the uncertainties in flood determination methodologies, and (3) to assess protective designs to interdict ground water in the event of a core-melt-type incident.

9.2.3 Relationship to Other Programs

Because of the high density of nuclear power plants in the Eastern United States and the unique seismic characteristics of that region, the NRC has been developing the data base to assess the earthquake hazard there. The U.S. Geological Survey's (USGS) earthquake hazards program concentrates on the Western United States where the general seismic hazards are higher. NRC programs are coordinated with the USGS, the principal NRC contractor for the regional program covering the Southeastern United States, including the Charleston, South Carolina area. A number of State geologic surveys and universities are participating sponsors in the research program as cooperating and contributing partners.

The NRC meteorological research program is being coordinated through the Office of the Federal Coordinator for Meteorological Services and Supporting Research with similar programs in other Federal agencies, particularly DOE, EPA, Federal Emergency Management Agency (FEMA), Department of Defense (DOD), and Department of Commerce (DOC). In addition, the research efforts in the private sector, including those funded by the Atomic Industrial Forum, EPRI, and the American Petroleum Institute are closely monitored.

Related hydrological programs exist in the NRC and DOE waste management programs for understanding ground-water flow and transport mechanisms associated with uranium recovery facilities and shallow-land burial of low-level waste. Related work on flood determination and protection is being performed by FEMA.

9.2.4 Background and Status

The seismology/geology regional program consists of (1) monitoring and interpreting seismicity, (2) collecting and compiling surface and subsurface geological information, and (3) conducting tectonic analyses. The preliminary results from the first 5-year program of regional geologic investigations is a very limited correlation between surficial, pre-Cenozoic (older than 65 million years) geologic structure, and current seismicity, i.e., between geologic structure and seismic hazard. Consequently, future geologic research will investigate the causes of seismicity and the current state of stress in the crust. Topical programs to study specific technical issues related to site-specific spectra, regional seismic wave attenuation, and strong ground motion will be initiated in FY 1982 and FY 1983 as funds become available.

For the meteorology program, field-tracer tests to obtain tracer concentrations and meteorological measurements were begun in FY 1972. These tests are used to assess dispersion models for emergency preparedness and for siting evaluations. A field test in a shoreline environment is planned for FY 1982 and an additional test in an environment not yet selected in FY 1983. Research to characterize severe weather events, in particular tornadoes and lightning, is expected to continue at a reduced level in FY 1982 and FY 1983.

The results of completed hydrology research projects on the probabilistic assessments of flood hazards, flooding effects, and the assessment of ground-water interdiction techniques will be used in writing regulatory guidance during FY 1982 and FY 1983.

9.2.5 Research Program Plan

9.2.5.1 Seismology/Geology

Delineating and characterizing tectonic provinces, as defined in the seismic and geologic siting criteria (Appendix A to 10 CFR Part 100), have been the principal concerns addressed by this research to date. In FY 1982, a review and a reevaluation of the seismology/geology program was initiated. Based in part on this review, specific areas of seismic activity in the Eastern United States that coincide with existing or proposed nuclear facilities are being targeted for further seismological and geological investigations. These specific areas of seismic activity will be investigated with an integrated seismological and geological program and will include:

1. New Brunswick with its two magnitude 5.5 earthquakes of January 1982;
2. Ramapo fault system;
3. Giles County, Virginia;
4. Charleston, South Carolina; and
5. Possible extension of the New Madrid system into the Wabash Valley.

These areas are within the broad regions of previous investigations, i.e., within New England, Southeastern United States, and Central United States (Anna, Ohio; New Madrid; and Nemaha Ridge).

The seismic networks will continue to monitor regional seismicity because it is impractical to make this task site specific. The seismic networks will be upgraded to consider site-specific spectra, strong ground motion, and differences in regional seismic wave attenuation. Programs will be initiated to reduce the uncertainties in estimating site-specific spectra and regional attenuation of seismic waves.

The geological phase of the program will address specifically the geologic and geophysical characteristics of seismically active areas to complement the short seismic record. The geological activities include:

1. Investigation of the state of stress and recent strain within the crust;
2. Investigation of the structure of the crust using geophysical techniques such as seismic reflection profiling, gravity, areal magnetics, and geoelectrics; and
3. Specific geologic mapping directed at understanding the earthquake mechanics.

The final reports for the first 5-year programs in three of the four regions (Northeastern United States, Central United States, and Nemaha Uplift) will be completed by mid-FY 1983. A second USGS Professional Paper on Charleston, South Carolina, is expected by this same date. This work was partially funded by the NRC.

9.2.5.2 Meteorolo /

Transport and diffusion are the two components of the dispersion process that move airborne effluents from nuclear power plants through the atmospheric pathway to potential receptors. The capability to characterize the dispersion of released airborne effluents under a variety of atmospheric and topographical conditions is necessary in the assessment of both public safety and environmental impacts. It is required that nuclear power plants be able to safely shut down in the event of severe atmospheric phenomena. Consequently, safety-related structures, systems, and components must be designed to withstand the effects of such phenomena. The assessment of the contribution to the overall risk to the public from severe natural-phenomena-induced failures beyond the design basis requires information on the magnitude, frequency of occurrence, and geographical distribution of severe atmospheric phenomena.

Research in atmospheric dispersion supports NRC's emergency preparedness program. This research involves the development, execution, and analysis of both empirical, field, and wind-tunnel tracer tests. Such tests are used to obtain the high-quality tracer concentrations and meteorological data needed to verify or evaluate atmospheric diffusion and transport codes. The series of field experiments will be extended to include additional tests in a river valley-rolling terrain environment and in a sea coast environment. The series should be completed in FY 1985. The information will be used to evaluate dispersion models over various distances and for different meteorological and topographical regimes. Such assessments will provide a technical basis for selecting particular models for use in emergency planning requirements under Appendix E to 10 CFR Part 50 and for site evaluation purposes.

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 used a limited amount of tornado data to formulate the current NRC position. Additional data and improvements in data quality are necessary to evaluate the adequacy and accuracy of the information used as a basis for Regulatory Guide 1.76. To complete the research program in this area, a reconciliation of the two existing tornado data sets regarding the size and intensity ratings of tornadoes reported since 1950 will be accomplished. In addition, a regionalization of the contiguous United States, each region with its own design basis tornado, is planned. This regionalization will be based on meteorological, demographic, and topographical factors that modify tornado occurrences, severities, and reportings. This research should be completed by FY 1984.

9.2.5.3 Hydrology

For ground-water transport, the research will be divided into three separate tasks: (1) ground-water flow and transport mechanisms in the vicinity of the assumed core-melt zone, (2) site evaluation studies of ground-water transport potential for a variety of geologic and hydrologic conditions, and (3) assessment of hydrologic design features to interdict releases. This work will be guided by experience gained from licensing and case review and is planned to be accomplished by FY 1987.

Similarly, for flood determination methodology and design protection, the research includes (1) assessing long-term representativeness of flood records, (2) defining the methodology for selecting flood event confidence limits, and (3) assessing the likelihood of failure of flood protection at nuclear power plants. Further, these studies will be used as input for the development of probabilistic risk assessment and consequence models.

To assess the probability of flooding at nuclear power plant sites, it is necessary to subdivide flood records by causative mechanisms to prevent underestimates of probabilities and to minimize residual risks and uncertainties. Flood records at stream, coastal, and lake sites contain information that reflects the causes of individual events.

The majority of operating and proposed nuclear power plants depend on both human intervention and flood protection devices to prevent accidents in the event of severe floods. A large number of flood protection projects that have failed to perform their intended function have been built in the United States. Data on failure rates and related causes are necessary to provide licensees and the staff with better guidance on design and operation criteria.

The research effort in flood determination and prevention should be accomplished by FY 1987.

9.3 Health Effects

9.3.1 Issue

NRC must ensure that its regulations adequately protect the health and safety of the public and workers from the harmful effects that might result from the use and production of ionizing radiation and radioactive materials in NRC-licensed nuclear activities. In order to fulfill this mandate, the health research and standards program must be designed to:

1. Improve the technical bases used to evaluate the potential radiological impact of proposed or licensed activities, and

2. Assess the adequacy of, and develop as necessary, radiation protection standards.

The technical and safety issues in health effects are:

1. The current NRC standards are based on the linear, dose-rate-independent, nonthreshold dose-effect model. Although this model may not be conservative for alpha particles and neutrons, it is generally considered to be overly conservative for low linear energy transfer (LET) radiation such as gamma rays. Any overestimates of possible health damage for low-LET radiation can have major regulatory implications for ALARA programs that depend on total collective dose, particularly when the total collective

dose is based on the exposure of large populations to very low dose rates. Further dose-effect studies are needed to clarify the extrapolated values for both high- and low-LET radiation.

2. Uncertainties exist in the metabolic and dosimetric models used to assess the potential health risks from exposure to various compounds in the nuclear fuel cycle, particularly those radionuclides abundant in the front end of the cycle.

3. The present 10 CFR Part 20, "Standards for Protection Against Radiation," has been amended many times since its initial issuance in 1959. However, none of the amendments has changed the basic structure or fundamental approach of this regulation for radiation protection. Radiation protection has changed in the last several years. National and international bodies now provide assessments of risk as a basis for recommending new standards. A revision of 10 CFR Part 20 will be based on this concept, using the system of dose limitations recommended in Publication 26 of the International Commission on Radiological Protection (ICRP).

4. Rulemaking is needed to simplify, clarify, and reorganize medical licensing requirements in 10 CFR Part 35, "Human Uses of Byproduct Material," because portions of Part 35 are obsolete and unclear. The Part 35 review and rewrite also implement Executive Order 12291, which concerns periodic and systematic review of the regulations.

9.3.2 Research Program Objective

The main objectives of the health effects research program are to:

1. Develop information bases and dosimetric methodologies for assessing individual and population doses and consequent health risks resulting from environmental, occupational, and medical exposures to radioactive material,

2. Provide verified information and models for metabolic pathways and mechanisms of radiation injury, and

3. Improve the understanding of causal links between low-level radiation exposure and human effects through analysis of health effects exposure data.

The main objective of the health effects standards program is to update and maintain NRC radiation protection standards so they are consistent with advances in radiation protection philosophy and methodology and with the latest research findings on the biological effects of ionizing radiation.

9.3.3 Relationship to Other Programs

Efforts of the health effects program to develop public health standards and to support a bioeffects research program are well coordinated with national and international standards advisory scientific bodies (ICRP, National Academy of

Sciences (NAS), National Council on Radiation Protection and Measurements, United Nations Committee on the Effects of Atomic Radiation, and with other Federal agencies (DOE, DOD, EPA, Occupational Safety and Health Adminstration, National Institutes of Health, and Food and Drug Administration (FDA)). NRC membership on the Interagency Radiation Research Committee (IRRC established by Executive Order) ensures NRC input into the overall Federal biological effects research program. The NRC health effects research program is consistent with the biological effects Federal research strategy developed by IRRC (1981) and the recommendations of the NAS (1981) and the General Accounting Office (1981).

9.3.4 Background and Status

The FY 1982-1983 health effects research program includes both the continuation of some research projects initiated in response to specific requests by other program offices and the initiation of several new projects required to resolve the technical and safety issues discussed above. These new projects will include analysis of existing human data, animal experiments, and mathematical modeling. The projects will primarily focus on obtaining information on radionuclide metabolism, dosimetry, and health effects required for more accurate risk assessment.

Additional information will be collected on populations exposed to radioactive materials in order to improve health risk assessments of NRC-licensing activities. A study of thorium concentrations in tissues of former thorium workers will provide new data, by FY 1986, on which to base standards on the intake of thorium compounds. By FY 1984, a study on a population who underwent diagnostic procedures using iodine-131 in childhood is expected to provide improved information on the potential consequences of routine and accidental releases of radioactive iodine.

In FY 1982-1986, somatic effects, generic effects, and life shortening will be studied in animals exposed to neutron energies with doses comparable to present occupational exposures. These data are expected to decrease the present large uncertainties in evaluating the health impact of occupational exposures to neutrons.

Metabolic studies in multiple species will provide information on the patterns of deposition of industrially produced mixed oxides, on the biological and physical characteristics of inhaled yellowcake, and on the patterns of deposition and histopathological effects of uranium ore dust. By FY 1983, a study will be completed to evaluate the behavior of mixed oxides of uranium and plutonium in order to develop a bioassay program for nuclear fuel cycle licensees. Data on the solubility of yellowcake will be used to resolve the discrepancy between the regulatory classification of yellowcake as insoluble and the bioassay data from uranium mill workers that suggests greater solubility of uranium.

A study to determine whether immunological disorders predispose mice to radiation-induced leukemia will be completed in FY 1982. These results will help resolve a controversial theory that maintains radiation-exposure limits should take into account the fact that some individuals have an increased sensitivity to ionizing radiation.

The use of hematological changes as indicators of accumulated radiation dose and predictors of late effects of whole-body irradiation will be explored by a continuing long-term study of late health effects and causes of death in irradiated beagles.

In FY 1983, mathematical models for calculating doses to internal organs from inhaled or ingested radionuclides will be refined. Other models for improving the quantification of predicted health consequences resulting from large releases of radioactivity will also be developed in FY 1983.

In FY 1983, a review of existing human data involving exposures to radon and its decay products will be initiated. This review is necessary because large uncertainties exist in extrapolating the findings of lung cancer in uranium miners, caused by high levels of exposures to radon decay products, to the general population that surrounds uranium milling facilities licensed by the NRC.

In FY 1982-1983, work will continue on the development of a national registry of nuclear power plant workers. The information from the registry could be used in any future epidemiological studies that might be warranted both for tracking transient workers and for facilitating industry compliance with NRC reporting and recordkeeping requirements for occupational exposures.

In FY 1982, the technical and economic impact on NRC licensees of the proposed revision to 10 CFR Part 20 will be evaluated. The proposed revision is scheduled for publication in the Federal Register in late FY 1982. In FY 1983, public meetings will be held and responses to comments prepared.

The rewrite of 10 CFR Part 35 should be completed by FY 1983. This will resolve many outstanding issues within NRC's medical regulations.

9.3.5 Research Program Plan

The epidemiological followup study of former thorium workers will reach the point where only the followup of autopsies will be necessary for FY 1983-1985. Data collected from the thorium workers will provide information on the patterns of deposition and translocation in inhaled thorium in humans. By FY 1986, sufficient additional deaths should have occurred for an update of the 1980 mortality study. The findings may necessitate a change in the standards for thorium in air for occupational exposures.

A study to determine the dose-response relationship for the development of thyroid tumors in a population who underwent diagnostic procedures using iodine-131 in childhood will be conducted in cooperation with the Bureau of Radiological Health (FDA) and the National Cancer Institute. The epidemiological followup study of these children will be completed in FY 1984. The results of this study will provide direct information on the incidence of thyroid diseases and other pathological effects of iodine-131 exposures and will serve as a data source comparing the effectiveness of radioactive iodine vis-a-vis x-ray to induce thyroid abnormalities in children. In radiation protection, children's thyroid tissue is the critical organ controlling effluent limits for radioactive iodine.

Following completion of the FY 1983 review of human epidemiological studies involving exposures to radon and its short half-life decay products, possible additional followup studies of these populations will be considered. Laboratory investigations will be supported that provide needed information on the association between exposure to radon and increased lung cancer. Other studies will also be initiated to provide additional information on the quality factors for, and dose-rate dependence of, high-LET radiation in general.

The large-scale study of mice exposed to either fission neutrons or gamma rays will be completed in FY 1986. The results will be useful in resolving questions concerning the effectiveness of neutrons to produce harm at low doses. The application of a variable quality factor dependent on dose would be required for neutron exposure, a change from current practice.

The dose-pattern and toxicity study of yellowcake materials inhaled by multiple species will be completed in FY 1985. The information will be considered for possible revision of metabolic and dosimetric values used in assessing the chemical and radiological toxicity of uranium. In addition, long-term health effects will be determined by a study to be initiated in FY 1984, involving chronic exposure of animals to yellowcake aerosols.

A DOE-supported long-term experiment, involving whole-body, chronic, low-level exposures of beagles, is partially supported by NRC. The study will continue until statistically valid endpoints are determined. The study is attempting to characterize hematological changes as a function of dose and as predictors of late effects of whole-body irradiation.

In the area of accident consequences, work will be initiated to develop health risk estimators for emergency actions.

In FY 1984, neptunium toxicity studies will be initiated to improve radiological assessments of waste disposal sites. Predicted pathways for accumulation and transport of neptunium will be verified when it is administered via injection or ingestion.

Following resolution of public comments, the final rule for 10 CFR Part 20 will be published. NUREG reports and regulatory guides will be issued, as required, for implementing the revised rule.

In FY 1984, a measurement improvements program for teletherapy and brachytherapy calibrations will be initiated.

In coordination with FDA, guidance for the safe operation of nuclear pharmacies will be developed and issued. Rules, guidance, and environmental statements on exemptions from licensing of certain medical products with low levels of radioactive material will also be developed and issued.

10. SYSTEMS AND RELIABILITY ANALYSIS

The mandate of the NRC is to ensure that the uses of nuclear technology that it regulates pose no undue risk to the public health and safety. The basic issues the NRC must face to satisfy that mission are: (1) what constitutes undue risk? (2) what are the risks associated with each aspect of regulated nuclear technology? and (3) if undue risk is identified, what should be done about it? The systems and reliability analysis decision unit includes programs that deal directly with these basic issues.

Systems and reliability analysis entails the application of reliability engineering techniques to nuclear safety issues throughout the nuclear fuel cycle and other uses of nuclear technology. The technical research disciplines involved include system reliability analysis, accident sequence analysis, severe accident consequence analysis, and risk assessment. The work uses results from phenomenology research (particularly on severe reactor accidents described in Chapter 4) to model accident consequences and to develop methods for analyzing the reliability of safety systems and the risks to the public from regulated nuclear activities.

Research on systems and reliability analysis serves several purposes. It strengthens NRC's capability to analyze risks and understand the relative importance of various safety issues. It helps NRC to evaluate alternative methods for resolving safety issues and to select effective strategies for regulation. This work also coordinates the drafting of selected rules. Another purpose is to help NRC identify and set priorities on safety research. It is the function of this decision unit to develop current, comprehensive assessments of risk for principal nuclear activities, including systematic appraisal of the significant sensitivities and uncertainties associated with that risk. Risk analyses for nuclear power reactors are further discussed in Sections 10.1 and 10.2, and the risks of all other regulated nuclear activities are addressed in Section 10.3.

Finally, this activity trains selected NRC staff members who need to understand risk analysis to perform their regulatory duties, for example, staff members who review probabilistic risk assessments prepared by licensees.

Following the TMI accident, the NRC markedly increased research on systems and reliability analysis. This long-range research plan proposes further increases to develop effective methods for stabilizing the regulatory process. The plan then proposes sharp cutbacks in this research as the results become available and are applied.

The systems and reliability analysis research program is described below in terms of three program elements: reactor risk analysis, risk methodology and regulatory analysis, and transportation and materials risk.

10.1 Reactor Risk Analysis

10.1.1 Issue

The fundamental issues dealt with by this element are: (1) By what means should NRC measure risks associated with nuclear plant operations? (2) How should risk-measurement codes be improved to ensure accuracy of risk assessments? (3) What engineering insights (human or physical) can be gained from applications of risk-assessment techniques to resolve generic and specific nuclear plant safety questions? Since these issues are so fundamental, it is natural that they be addressed by this element, in part, in the context of support for other decision units and support for broad-based rulemaking activities and, in part, by addressing particular subissues, for example, determining and implementing modifications to specific risk assessment codes. Regardless, in what follows, each activity can be related either directly or indirectly to one or more of the foregoing three fundamental issues.

In the near term, it is expected that the products of efforts addressing one issue could be used as information affecting another in a somewhat cyclic fashion. However, by the late 1980s, means by which we measure risk should be fairly well established. Also by then, risk-assessment codes used in this element should have well-developed models and structures requiring minimal maintenance. Thus, by the end of the 1980s, the third issue would be virtually the only issue remaining and would eventually become a relatively minor one as the industry moves to standard plant designs reflecting previous lessons learned. Applications of probabilistic risk analysis (PRA) would continue to identify relatively significant uncertainties to guide whatever further research seems warranted in concert with periodic restatements of risk.

10.1.2 Research Program Objective

The NRC has now sponsored eleven PRA studies through the completed Reactor Safety Study (RSS, WASH-1400), the nearly completed Reactor Safety Study Methodology Applications Program (RSSMAP), and the ongoing Interim Reliability Evaluation Program (IREP). Lacking among these studies is an assessment of a BWR Mark II design. Thus, one objective is to fill this gap with an assessment of that design by an additional IREP study. The study objectives will be to exercise the IREP procedures guide, developed out of lessons learned from the previous studies, and to test additional methods for addressing concerns such as common-cause failures, pressurized thermal shock, etc. within the framework of a PRA for ultimate use in the National Reliability Evaluation Program (NREP).

The objectives of IREP extend beyond those of the two former studies (provision of base technology and limited use of that base) to include preliminary identification (within the limitations of PRA) of dominant accident sequences, development of foundations for subsequent more intensive PRAs, expansion of the cadre of experienced risk-assessment practitioners within and without the NRC, and the evolution of procedures codifying the competent use of these techniques for use in IREP studies in the mid-1980s and as the basis for the NREP procedures guide. In addition to these eleven studies, industry-produced PRAs are also being reviewed under this element for usable accident sequence information such as the identification of contexts in which operator error is particularly important to risk. An IREP-like PRA of the Clinch River Breeder Reactor (CRBR) is being undertaken in this element to assess the comparative risk of the CRBR vis-a-vis that associated with LWR plants. This project would provide a study, completely accessible for useful interim results, whose objectives and direction of emphasis were controlled by the NRC to ensure the end product conformed in nature and timeliness to agency needs. Areas requiring emphasis from an NRC standpoint would include those requiring accurate sensitivity studies to guide future research. The study would be beneficial to the NRC as it views CRBR from a licensing aspect by providing an independent basis of judgment on many unusual risk issues inherent in advanced reactor technology. In addition to providing an independent assessment of CRBR risk, the study will exercise IREP procedures on a plant not yet built. This should provide insights into problems associated with performing PRAs on standard plant designs for future plants. Limited resources are being applied to source-term analysis and siting criteria development for high-temperature gas-cooled reactors (HTGRs) in anticipation of increased interest in that type of plant.

A major objective of this element spanning the full term of this plan is to see a number of studies performed to reassess and refine the predictions of severe accident sequences and likelihoods made in PRAs. This reassessment and refinement will be made based on the availability of new data and PRAs, the reconsiderations of previously produced event trees and accident sequences with greater emphasis on potential common-cause failure mechanisms, the investigation of possible "precursor" events in operating LWRs, and the consideration of the relative likelihood of "TMI-like" accidents distinguished from full core meltdown accidents. In concert with and in part guided by the refinements identified through these efforts, a revision of the RSS is scheduled for FY 1984-1988. Like the original, the revision is intended to develop the most accurate possible assessment of the risks posed by externally caused accidents (excluding sabotage) at the commercial LWR plants. The objective of this intensive study will be to identify the strengths and weakness of less intensive risk assessments such as IREP.

This element has a second major objective: the development of computer codes for use in PRA to analyze the phenomenological processes associated with severe accidents. Because of the need in PRA studies for the analysis of many accident sequences, these codes will be relatively simplistic and fast running. They will thus be the more approximate and quick counterparts to the more mechanistic codes being developed in parallel in other decision units. Two generations of codes are to be developed in this element; each is described in Section 10.1.4.

In this element, analyses are to be performed of the risk-reduction potential and costs associated with a spectrum of possible plant modifications. The first round of these analyses is scheduled for FY 1982-1983. Followup studies in FY 1984-1988 will capitalize on additional completed PRAs and refined value/impact analysis methods in development under the risk methodology and regulatory analysis element of Section 10.2. Included in these possible modifications are filtered-vent containment systems, alternative shutdown heat removal systems, and stronger containments. The objective of such analyses is to identify those modifications that appear to present the most cost-effective risk reduction. Since such results will vary with the specific plant design being considered, analyses will be performed for all major design types (PWR large dry and ice condenser containments and BWR Mark I, II, and III designs). To be effective, such and similar efforts require a periodic restatement of risk and identification of significant uncertainties that warrant research. This up-to-date statement of risk by reactor and containment type must be a usable reference for considering the risk significance of regulatory activities and for setting priorities for research. This is the thrust of the major objectives of the two preceding paragraphs, which makes PRA refinement and performance a continuing obligation throughout the years of this plan.

A variety of research and standards development projects are under way or planned to support reactor safety standards development and the development of staff aids and guides for use in the area of severe accident risks. These include final and proposed rules for the control of hydrogen during severe accidents and proposed rules for dealing with anticipated transients without scram (ATWS). A notable change in regulatory approach is the recently issued final rule for near-term construction permits and manufacturing licenses, which explicitly requires containment modifications for dealing with severe accident forces and requires the use of PRA as a design evaluation tool.

The continued accuracy of a nuclear plant PRA could be significantly enhanced by an effective plant reliability assurance program that not only ensures periodic updating of the PRA itself but also monitors items critical to the PRA assumptions to lend assurance of design adequacy and removal of blind spots in surveillance testing. To this end, the reliability assurance program was initiated in FY 1982 to tailor to the nuclear industry an approach for continuous and effective safety management based on experience in the aerospace and defense industries.

Also under way in this element, are support studies for NRR, the objectives of which are to resolve the station blackout and pressurized thermal shock unresolved safety issues and to produce models and a computer code for assessing the risks to nuclear plants from offsite transport of hazardous materials.

This type of staff assistance to other organizations is expected to continue indefinitely, although it is difficult to predict the exact nature of the need. However, in anticipation of continued requests for assistance, a small amount of Reactor Risk staff effort will be expended to maintain a high level of expertise and liaison with other practitioners in all fields involved in the area of nuclear reactor reliability engineering and risk assessment.

10.1.3 Relationship to Other Programs

Research conducted under this element is related to research under other decision units or organizations in the table presented below. Additional discussion on each item follows the table.

Decision Unit	Coordinated Activity
1. Accident Evaluation and Mitigation, Section 4.3	Code development efforts on MARCH, CORRAL/MATADOR, and MELCOR
2. Advanced Reactors, Sections 5.1 and 5.2	CRBR PRA and HTGR source-term analysis and siting criteria development
3. Reactor and Facility Engineering, Section 6.2	IREP/NREP and SSMRP program interface development

4. Facility Operations and Safeguards, Section 7.1	Identification of human contributions to risk
5. Ibid., Section 7.4	Accident consequence modeling

<u>Other Organizations</u>

6. Office of Nuclear Reactor Regulation, NRC	Pressurized thermal shock analyses
7. Electric Power Research Institute	Technical review of IREP/NREP guide documents and NRC-produced PRAs
8. American Nuclear Society and Institute of Electrical and Electronics Engineers	Industry standard on station blackout
9. Institute of Electrical and Electronics Engineers	Development of reliability assurance program guides and standards
10. Organization for Economic Cooperation and Development	Improvement in techniques for estimating consequences and risk
11. Federal Emergency Management Agency	Accident consequences modeling

1. Efforts on MARCH, CORRAL/MATADOR, and MELCOR will use the data base being developed under the fission product release and transport element in model development and improvement to reduce uncertainties in prediction of fission product release and transport.

2. In cooperation with the advanced reactors program, a risk assessment of the CRBR was initiated in FY 1982 with completion scheduled by the end of FY 1983. HTGR source-term analysis work began in FY 1982 with siting criteria development scheduled to follow in FY 1983.

3. Discussions with Seismic Safety Margins Research Program (SSMRP) personnel in FY 1982 indicated that IREP/NREP fault trees might be used in that program. In particular, the Browns Ferry Unit 1 fault trees might be modified to provide an earlier BWR seismic analysis. Continued liaison is expected.

4. The Reactor Risk Branch is providing accident sequence and human error contribution information to the Human Factors Branch for use in program planning and possible use in simulator development. This modest staff effort will be replaced by products from the accident sequence evaluation program in FY 1983 and beyond.

5. The reactor risk analysis element is a resource of information regarding potential scenarios, source terms, and consequences for reactor accidents as required for emergency planning and response.

6. This element is providing to NRR a set of scenarios leading to possible pressurized thermal shock situations. Prevention techniques and engineering fixes will be explored and recommended.

7. EPRI has requested involvement, in a review capacity, in the production of IREP/NREP procedures guides and in the review of design deficiencies revealed by probabilistic risk assessment.

8. A joint committee (ANS/IEEE) has requested and obtained participation by Reactor Risk Branch personnel to aid in the development of an industry standard on station blackout.

9. Reactor Risk Branch personnel are participating as members of the IEEE Nuclear Systems Reliability and Safety Committee in the investigation of approaches to reliability assurance program development.

10. To better understand the differences in consequence modeling techniques, a comparison study was organized by the NRC under the auspices of the Committee on the Safety of Nuclear Installations. About 30 organizations from 16 countries are participating in the study. The primary focus of the exercise will be to improve the techniques of estimating consequences and risk and to develop a methodology for estimating uncertainties in the models and data.

11. The reactor risk analysis element is a resource of the information regarding potential scenarios, source terms, and consequences for reactor accidents required for emergency planning and response.

10.1.4 Background and Status

10.1.4.1 Reactor Safety Study Methodology Applications Program (RSSMAP)

The RSSMAP program is intended to apply the methods and insights of the RSS to a somewhat broader spectrum of LWR designs. Thus relatively limited event tree and fault tree analysis has been performed on four designs: a Babcock and Wilcox plant; a Combustion Engineering plant; a Westinghouse four-loop plant with an ice condenser containment; and a GE BWR plant with a Mark III containment. The final product of each plant study is a discussion of the likelihood of experiencing serious core damage and of having particular magnitudes of releases of radioactive material from the plant (the RSS "release categories") and an explanation of what types of accidents (e.g., station blackout, ATWS) contribute importantly to these likelihoods of releases. Three of the final reports on these studies are already published and the fourth is to be published early in FY 1982. It is planned that the Mark II analysis will be completed by late FY 1983 or early FY 1984 if not opted for under the IREP program.

10.1.4.2 Interim/National Reliability Evaluation Program (IREP/NREP)

Activities under IREP began in FY 1980 with Phase I, which consisted of a pilot study of the Crystal River Unit 3 plant. This phase was completed during the second quarter of FY 1982 with publication of NUREG/CR-2515, "Crystal River-3 Safety Study." Phase II, which began in FY 1981, consists of studies of Browns

Ferry Unit 1, Calvert Cliffs Unit 1, Millstone Unit 1, and Arkansas Nuclear One Unit 1. The production of these studies made use of insights gained from the Phase I activities. Phase II will be completed during the last quarter of FY 1982 with publication of safety studies on all four Phase II plants and a procedures guide for performance of future studies. FY 1983 will see initiation of an additional plant safety study that will exercise the procedures guide and explore incorporation of in-depth common-cause failure analysis, both qualitative and quantitative. The possibility of using IREP fault trees in the SSMRP (Section 6.5) will be explored in FY 1982. As of this writing, closer ties between the two programs appear quite beneficial, and they should be developed and implemented during FY 1983 and beyond.

In the IREP program, PRAs are being performed on a set of plants using advanced methods and data. Detailed fault trees and event trees are being generated and quantified for the purpose of yielding estimates of the likelihood of various serious accidents, an overall likelihood of severely damaging the core, and the likelihood of significant radioactive releases. While this product will provide a measure of safety of the particular plants, it will also provide a basis for developing general PRA procedures and techniques for use on a larger scale (i.e., on all U.S. nuclear plants). The NREP program is intended to be the vehicle for the larger-scale effort.

The plant analyses now under way in IREP are scheduled for completion in FY 1982. NREP studies to a standard model have not yet been initiated; however, many plants have undertaken independent PRA analyses.

10.1.4.3 Advanced Reactor Safety Research Program

In anticipation of, but separate from, CRBR licensing activities, this element initiated in FY 1982 an IREP-like study of the Clinch River plant. The study is scheduled for completion by the end of FY 1983. HTGR source-term studies were initiated in FY 1982. Site selection criteria development for HTGRs will be completed by the end of FY 1983.

10.1.4.4 Industry PRA Reviews

In addition to the plant PRAs being performed under RSSMAP, IREP, and NREP, licensees have initiated (for various reasons) PRAs on specific plants. It is planned that, as such PRAs become available, reviews will be undertaken and the results will be incorporated into the overall accident sequence likelihood reassessments being performed in this element if the review so warrants. By mid-FY 1982, licensee-produced PRAs for Big Rock Point, Limerick, and Zion will have been reviewed. The dominant accident sequences and human error contributions from the Limerick PRA have been identified for use by the Human Factors Branch.

10.1.4.5 Accident Sequence Evaluation Program

In this program, reviews will be made of the accident sequence (event tree) evaluations in plant-specific risk assessments such as the RSS, IREP, and RSSMAP. NREP and industry-produced PRAs will be included as they become available. These risk assessments and the reevaluated event trees from this program will be used as the foundation from which the risk analyses of plant modifications will depart. The objective of the event tree reevaluation will be to consider the need for (and make, as needed) modifications to the event trees to incorporate new information and make them more appropriate for use in the value/impact analyses. More specifically, modifications will be made to differentiate between sequence variations not previously necessary, but important for the value/impact analyses; to permit differentiation between core damage and full core melt sequences (and to assess their relative probabilities); to make modifications to account for (probabilistically) poorly understood events such as fires, sabotage, and operator error; and to attempt to make the event trees more generic than originally established.

There are a variety of useful products that can be derived from a thorough review and delineation of reactor accident sequences in current PRAs. Applications using the event tree or similar methodology and the fully delineated reactor accident sequences include (1) establishing the feasibility and utility of real-time diagnosis and prognosis of accidents in progress for the NRC Emergency Operations Center, emergency planning, and operator training; (2) developing screening aids that would be useful for IE/AEOD incident review, NRR risk outlier plant identification, evaluation of SRP revisions, and IE inspection/reporting requirements; (3) identifying reactor safety research areas (systemic and phenomenological) where available resources could be most effectively used; and (4) identifying potential flaws in PRA estimates of reactor accident probabilities that may not agree with precursor experiences.

This program is now in the process of updating estimates of sequence likelihood; completion of this phase is planned for mid-1982. The program will continue as long as PRAs are being produced.

10.1.4.6 Accident Sequence Precursor Program

In the accident sequence precursor program, events in operating LWRs are being examined for their potential, when combined with other events, to lead to a severe accident. After an initial screening to define the more important events, estimates of the likelihood of these events resulting in a severe accident will be made. The screening of events has now been under way for more than a year. Initial likelihood estimates are now available; further estimates will be developed over the duration of this plan.

10.1.4.7 Design Alternatives Program

As a result of the accident at Three Mile Island and research in reactor accidents, the NRC has come to recognize that the current regulatory requirements governing the acceptable design of containment systems in commercial light-water-cooled power reactors need to be revised. The current body of requirements are founded on the design basis large loss-of-coolant accident (LOCA), the postulation of a release of radioactivity to the containment described in TID 14844, and the dose criteria of 10 CFR Part 100. Several changes are called for: (1) the large LOCA is no longer thought to be an enveloping accident that can be taken as the surrogate of the full spectrum of challenges to reactor containment systems for purposes of design and safety analysis, (2) the TID 14844 source term is no longer thought to be an acceptable model of the containment atmosphere conditions to be expected of challenges to containment systems (see, for example, NUREG-0771 and -0772), and (3) the design and safety evaluation criteria for engineered safety features intended to mitigate accidents entailing core damage are to be decoupled from reactor siting considerations (see, for example, the Report of the Siting Policy Task Force and "Action Plan on Degraded Core Cooling and Related Rules," USNRC, April 1, 1981).

The NRC is undertaking rulemaking to correct these deficiencies in the body of reactor safety regulatory requirements. The design alternatives program is one of several research programs intended to provide the technical basis for rulemaking. The objectives of the design alternatives program are to explore the technical feasibility of a variety of concepts for the mitigation or prevention of accidents entailing damaged or melted cores, to assess the risk-reduction value of these concepts, to assess the impacts in cost and in attendant risks of these concepts, and to develop design criteria for the more promising concepts suitable for use as regulatory requirements. The work will entail conceptual design, applied risk assessment, and accident analysis.

10.1.4.8 Risk Assessment Code Development Program

The primary codes for use in NRC reactor accident consequence analysis are MARCH, CORRAL, and CRAC, which deal, respectively, with fission product and hydrogen release from damaged cores, fission product transport and plateout within containment, and possible property damage and health consequences to the public. These codes had their origins in the analyses performed for the RSS. Following release of WASH-1400, work was initiated to correct inefficiencies in the codes and to incorporate revised and improved models. During FY 1982, MARCH-2 will be produced by revamping the phenomenological models and computational methods of MARCH-1. CORRAL will be intensively revised into a new code, MATADOR. These revised codes will suffice while MARCH-2, MATADOR, and CRAC are being merged into one code, MELCOR. During both FY 1982 and FY 1983, experimental data and code development approaches will be reviewed and analyzed for selecting models for physical phenomena and mathematical techniques for MELCOR. Uncertainty analysis techniques will be developed concurrently with a new data management approach. The merging of MARCH-2, MATADOR, and CRAC into MELCOR will begin in FY 1983.

10.1.4.9 Reliability Assurance Program

A program was started in FY 1982 to explore the reliability assurance management practices and reliability engineering techniques developed in other industries for possible application to nuclear safety assurance. The aerospace, weapons, and electronics industries have pioneered management and technical analysis techniques to attain and maintain during operation the reliability of complex systems. System reliability analysis and prediction methodology were invented in these industries. Many of their techniques have never been tried in the field of reactor safety; however, the FAA has adapted many of these techniques to regulation and system procurement. These approaches to safety assurance and reliability assurance will be studied to determine if they can help to sharpen the focus of reactor safety assurance requirements to establish a higher level of risk-limitation effectiveness while avoiding overregulation.

10.1.4.10 Station Blackout Studies

Station blackout (i.e., the loss of all ac electric power at a plant) has been identified by NRC as an "Unresolved Safety Issue" because, among other things, it has a relatively high predicted probability in some plants of leading to high-consequence accidents. As such, it has been the subject of considerable study over the past several years, including a significant amount of research as to its likelihood in operating LWRs. As this information is being compiled, it is being incorporated in this element's overall evaluation of severe accident likelihoods.

10.1.4.11 Pressurized Thermal Shock Program

The pressurized thermal shock program was begun in FY 1982 to support NRR in resolving this issue. During FY 1982 and FY 1983, the program will have developed a set of event sequences leading to possible pressurized thermal shock situations for reference PWRs. Sensitivity studies will be performed as part of the analysis, and the effectiveness of the possible corrective actions will be gauged.

10.1.4.12 Transport Hazards Analysis

The transport hazards analysis program was begun in FY 1981 to provide means for assessing the hazards to nuclear plants from offsite transport of chemical and toxic materials. FY 1982 saw completion of model development and initial code development. Completion of code development is expected in FY 1983.

10.1.5 Research Program Plan

Many reactor risk assessment activities will continue through the period FY 1984 through FY 1988. IREP-like studies will continue to be used as a test bed for improved PRA methodology intended for use by licensees and NRR. For example, we expect to expand the scope of NREP procedures to include fires, floods, earthquakes, better prediction models of human error, and design or surveillance adequacy problems affecting safety function reliability. The draft procedures by which licensees will be expected to incorporate such risk contributors in their NREP studies will be debugged in trial IREP studies.

From time to time the reference reactor risk assessments commonly used in environmental reports, studies of generic safety issues, value/impact evaluations of regulatory changes, etc., will be revised to incorporate improved PRA methodology. The six reactors for which a PRA was done in the RSS and its followup program, RSSMAP, will be subject to these revisions. By FY 1984, one round of improvements will be completed using the improved accident likelihood estimation developed in the accident sequence evaluation program. The phenomenology of the accident processes and consequences in these PRAs will also be updated with the improved codes MARCH-2, MATADOR, and CRAC-2. In and following FY 1984, the improved code MELCOR will be available for the next round of updating the reference reactor risk assessments. As new phenomenological information is developed by the Division of Accident Evaluation (DAE) and the Division of Engineering Technology (DET) and new human reliability models are developed by the Division of Facility Operations (DFO), we plan to incorporate these findings in improved PRAs of the six reference plants and in a PRA of a seventh reference plant, a BWR with a Mark II containment.

The investigation into design alternatives will continue throughout the years of this plan if new and substantially different designs are proposed. The scope of the research is to embrace the current and planned population of commercial light-water-cooled reactors in the United States. Consideration will be given to backfits to operating plants, to plants under design and construction, and to requirements of new generations of LWRs. The research is to embrace each of large dry atmospheric and subatmospheric containments, ice condenser containments, and the Mark I, II, and III BWR containment designs.

The work is to be done iteratively, i.e., by successive refinement. The choice of an iterative program plan is dictated by several considerations:

1. A large number of combinations of design improvements are to be evaluated. Successive refinement allows the more promising combinations to be resolved from the less promising options without massive investment in research that later proves to be unimportant.

2. An iterative approach allows preliminary results to be reported early. The NRC staff needs frequent opportunities to evaluate the regulatory implications of the emerging research results and to provide for the coordination and midcourse correction of the research direction.

3. Several parallel programs will be developing improved tools and perspectives that can be fed into the later iterations of the conceptual design and value/impact analyses of this research program. Among these parallel programs are the phenomenological research projects of the DAE, the Indian Point and Zion studies by and for NRR, the siting rule research, a study to reexamine the accident sequence analyses in the reference risk assessments, other reactor risk assessments, and a study of the economics of reactor accidents for use in value/impact assessment.

The first round of studies will be completed prior to the span of this plan and will produce reports on alternative decay heat removal concepts, alternative auxiliary feedwater designs, and alternative containment designs.

Reactor risk assessment will be employed as the framework for evaluating the value and impact of the hypothetical improvements in the prevention or mitigation of core damage accidents. The event sequence analysis contained in the reference PRA studies will be employed to postulate the spectrum of challenges to prevention or mitigation options. In the first phase of the research, the accident frequencies in the reference PRA studies will also be used as guides to the value/impact study. However, the second and subsequent phases (iterations) of the research will consider the possibility that the several plants of each type may differ from the reference PRA in either the frequency or the phenomenology of accidents. Accident frequencies may differ because of differences in the design or operation of the plants, or because of scope limitations in the reference PRA studies, i.e., through sabotage, fire, floods, human error, design flaws, or other common-cause failure mechanisms not fully accounted for in the reference PRAs.

As the later iterations of these analyses are completed, it is also planned to use the results to help formulate design criteria for the most advantageous alternatives. These criteria could then provide technical support to the process of requiring such alternatives, if such requirements are forthcoming. Beginning in FY 1984 and continuing thereafter, approximately half the program will be focused on newer and planned facilities.

The Accident Sequence Evaluation Program (ASEP) consists of two phases. The first phase of ASEP provided a "first-cut" effort at defining a generic and comprehensive set of accident sequences for PWRs and BWRs. Major system, phenomenological, human, statistical, and other differences and uncertainties that are of potential importance to reactor accident sequences were identified. In the second phase of ASEP, a thorough review and further expansion of the Phase I work will be conducted. The results of this ASEP Phase II effort will be a major input to the design alternatives program for the evaluation of accident prevention, accident mitigation, and value/impact assessments.

A review of operating experience information and analysis programs will be performed to relate the insights developed in the Phase II evaluations to actual field experience. Operating experience programs to be reviewed will principally involve licensee event report (LER) evaluations performed for the NRC involving accident sequence precursors, system failure experience, and component failure data. This review will look for subtle but potentially important operating experiences associated with dominant accident sequences, component failure rates, including those induced by common causes or human error, or heretofore unreviewed interactions that may differ in expected frequency or underlying cause from that observed in current PRAs. As appropriate, LER searches will be conducted to enhance information obtained from these programs.

A sensitivity analysis and evaluation will be performed to identify and bound the importance of hypothetical interactions, dependencies, and common causes of failure that can increase the probability or risk of potentially dominant accident sequences significantly above estimates based on current PRAs and operating experience reviews. Such factors as human error, support system dependence, safety system interaction with nonsafety systems, operating and accident environment, external events, and other aspects that may couple initiating events to subsequent failure or multiple subsequent failures will be analyzed for potential importance. Where possible, potential causes will be identified for the sensitivity factors that significantly increase accident probability and risks.

Based on insights gained from activities of the previous two paragraphs, one or more accident sequence delineation schemes will be developed. The organizational approaches will be at a gross conceptual level depicting an information hierarchy of increasing detail that can accommodate accident descriptions in terms of causal, systemic, phenomenological, symptomatic, consequential, chronological, and probabilistic considerations. At this stage of development, only information levels and logic of the hierachy will be provided.

A multidisciplinary team will be employed to critically examine the accident sequence evaluation, operating experience review, and sensitivity analysis. The generalized accident sequence delineation schemes will also be evaluated.

The purpose of this effort will be to obtain varied and expert opinion on the completeness and apparent accuracy of the accident sequence findings. This effort will focus on the potentially important interactions, dependencies, couplings, and current PRA limitations that may preclude the identification of significant accident sequence considerations. The approaches used will involve a three-to-five-day working meeting with appropriate premeeting preparation, including distribution and review of accident sequence evaluation (ASE) documents.

A final ASE report for use in the design alternatives program will be prepared with due consideration of the critical examination. This report will provide a representative set of reactor accident sequences suitably delineated for use in evaluating core damage accident prevention and mitigation concepts and performing value/impact assessments. Insights obtained regarding the cause, probability, and uncertainties of the accident sequences will also be provided in the report.

One or more of the conceptual accident sequence delineation schemes will be expanded to evaluate its desirability for more detailed development. An evaluation will be made of the benefits and limitations associated with increasing levels of detail regarding the probability, cause, and progression. Standardized or equivalent terminology, symbology, acronyms, and initialisms will be used for each generic plant and accident sequence type. Event tree methodology will be used initially, although other accident sequence delineation methods may be investigated and used at various levels of information definition as appropriate. For instance, "reduced" fault trees or block diagrams may be useful in describing subsystem interactions as a subelement of a larger accident sequence event tree.

A final accident sequence delineation scheme will be developed. Generic "tree" structures (or equivalent) and descriptive text will be provided. The accident sequence evaluation will be converted to the format of the final accident sequence delineation scheme as a test application. Expert review and comment on this work will be obtained as appropriate, and a final accident sequence delineation report, including insights developed during this program, will be prepared for publication.

The accident sequence precursor program will have developed in FY 1983 a compendium of potential precursors from LERs submitted during the period 1969 through 1981. The process of LER review and screening will continue through FY 1988. It is anticipated that some developmental efforts will be required through FY 1985 on models suitable for use in computer-assisted time analysis of LERs for particular LWR designs. Boolean representations of potential core damage sequences from LWR plants assessed as part of the RSS, RSSMAP, IREP, and selected industry-produced PRAs will be used in this regard. Statistical techniques for prediction of core melt frequency, trend analysis, and importance calculations will be firm by the end of FY 1985. Beginning in FY 1984 and continuing through FY 1988, a subsidiary program will be carried out to examine in depth those precursors that deserve to be labeled "close calls." Insights gained from this analysis, indeed the entire precursor program, will provide invaluable guidance to ASEP, IREP, and, in particular, the reliability assurance program.

The development and use of improved risk codes for the modeling of severe accident physical processes will continue through mid-FY 1985. In FY 1982-1983, certain modifications will be made to the existing risk codes - MARCH, CORRAL (MATADOR), and CRAC. In parallel with these modifications, development will begin on a code (MELCOR) that will eventually replace these codes. The latter code is to be developed in such a way that it can be used to perform uncertainty analyses on these physical processes (as well as best-estimate calculations) and can be readily modified as new experimental results and models become available. The latter requirement permits the use of MELCOR as a means for incorporating new information from other RES programs (e.g., radionuclide source-term experiments) into risk studies.

In FY 1982-1983, activities in the MELCOR development effort include the review of existing information on core-damage/core-melt processes, containment system challenges, containment failure modes, radionuclide transport, and deposition modeling, including particulate and moisture through both air and liquid pathways. Models for consequences outside the plant, including those for property damage and cost of decontamination, will be reviewed. Methods and advanced computer programming techniques (e.g., data management systems) for uncertainty analysis will also be reviewed. Based on these reviews, recommendations will be made regarding each of these areas and code development will follow.

In FY 1984, it is expected that a first version of MELCOR will be completed. Documentation of this version and sensitivity and validation exercises will then be performed. As this version becomes available, it will also be used to reevaluate the predicted consequences of accident sequences previously analyzed in PRAs such as the RSS and RSSMAP.

Work will also continue in FY 1984 and into FY 1985 to upgrade models within MELCOR to reflect the completion of ongoing experimental programs. It is planned that in mid-FY 1985 this upgrading will be at a point that a second version of MELCOR can be made available. This version will then undergo similar documentation, sensitivity, and validation exercises. In addition, the development of methods for uncertainty analysis is expected to be completed by this time, so this code version and the developed methods can be used to perform quantitative uncertainty analyses on predictions of severe accident physical processes.

Beyond mid-FY 1985, it is expected that work in this area of code development will begin to diminish in magnitude. Principal activities in FY 1986-1988 are expected to be maintaining the code and updating it to reflect new data. Application of MELCOR is expected to be a part of other PRA programs.

The CRBR risk assessment and HTGR siting criteria development will be completed in FY 1983. However, this plan will allow for modest continued research efforts through FY 1988 contingent on interest and activity in the advanced reactor area.

Review of industry-produced PRAs is expected to require considerable staff time commencing about FY 1984 and continuing through FY 1988. Ad hoc staff effort for solving miscellaneous and unforeseen systems analysis problems can also be expected to continue.

The reliability assurance program will consist of two phases. The first phase, to be completed about the end of FY 1984, will focus on development of a pilot program limited to addressing problems associated with anticipated transients without scram (ATWS). A multidisciplinary team consisting of individuals from the nuclear, aerospace, electronics, and other industries will work in cooperation with a volunteer utility to develop a program to minimize surprises caused by design errors, blind spots in surveillance testing, errors in test and maintenance, and the conduct of various plant operations and evolutions. The program will incorporate time-tested procedures for quality assurance and quality control from other industries that can be adapted to nuclear plants. Mechanisms for experience feedback will be an integral part of the program.

The second phase (FY 1985-1986) will see the program enlarged to encompass all known initiating events with the goal of lending continued assurance that the assumptions made in the assessment of risk remain valid for the lifetime of the plant.

The principal new initiative for the FY 1984-1988 period will be a new attempt to develop a state-of-the-art assessment of the risks posed by accidents at light-water reactors, a Reactor Safety Study II. Like the original, the new study is intended to develop the most accurate possible assessment of the risks posed by externally and internally caused accidents (excluding sabotage) at commercial LWR plants. However, it will not be modeled on WASH-1400. There will be three principal tasks in the new RSS. One of these will be an intensive risk assessment of a single plant that has been the subject of a previous IREP study. The objective of this intensive study will be to identify the strengths and weaknesses of less-intensive risk assessments like IREP. This task will use very thorough accident-sequence analysis, including 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 fortreating 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 corrections or additions to prior, less-intensive risk assessments that are found to be necessary for the plant may be applicable to other reactors.

The second principal task in the new RSS study 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 has a risk profile characterized by a set of estimable 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 (only the historical record), 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 new RSS study 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.

Prior to the initiation of such a large program, however, the first step would be the performance of feasibility studies, methods selection, and cost/benefit analyses to determine the most cost-effective scope of the project. These activities would occupy the first half of FY 1984. Lagging these activities somewhat would be program formulation, planning, and scheduling for the surviving programmatic elements to be performed in FY 1985 and beyond.

The program schedule for reactor risk analysis is shown in Figure 10.1.

10.2. Risk Methodology and Regulatory Analysis

10.2.1 Issue

The principal issue being dealt with in risk methodology research is overcoming the limitations of probabilistic safety analysis that stand in the way of more trustworthy risk assessments and PRA applications in licensing and standards development.

The NRC safety regulations are suspected of harboring some loopholes in safety-significant areas, some instances of overregulation, and some features that pose unnecessary constraints or disincentives to creativity for licensees. The principal issues before the regulatory analysis efforts in research and standards development lie in the identification and correction of these off-target features in the agency regulations.

10.2.2 Research Program Objective

The objectives of the risk methodology and regulatory analysis programs are to:

1. Develop analytic methods to improve the completeness and accuracy of system reliability analysis techniques.

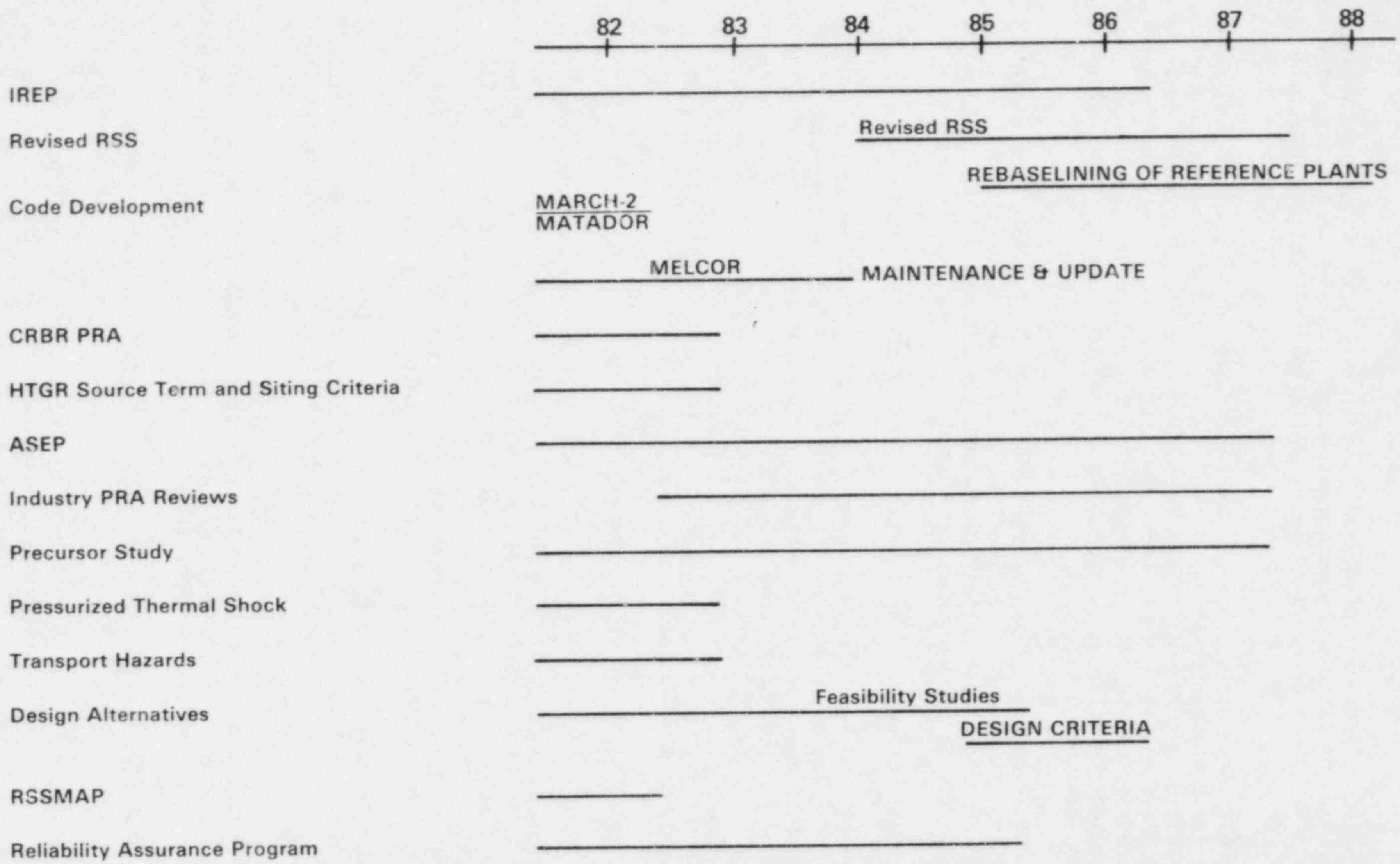

Figure 10.1 Reactor Risk Analysis Program Schedule

2. Develop methods and software with which to translate raw data on reactor experiences (initiating events, equipment failure, test and maintenance records, and precursors to accidents) into a reliability data base for use in PRA.

3. Develop strategies, tactics, and procedures for regulatory decisionmaking in the context of risk-based safety goals for use in major rulemaking, standards development, licensing casework, evaluation and feedback of licensee operating experience, and setting priorities in regulation, standards development, and research. In particular, develop procedures for cost-benefit, value/impact, and regulatory analysis and determining compliance with safety goals.

4. Identify and close loopholes in PRA methodology involving identification of initiating events, accident sequence delineation, common-cause failures, including external events and sabotage, and design adequacy or surveillance adequacy considerations.

5. Perform periodic and systematic review of the NRC regulations.

6. Develop and implement a program of technology transfer to other NRC divisions. The objectives of the program are to (1) train a cadre of PRA practitioners capable of evaluating PRA submittals and applying PRA techniques to regulatory problems, (2) apprise NRC management and selected staff of the capabilities and limitations of PRA and provide insights from past applications of PRA, and (3) provide technical assistance on statistical methods, probabilistic safety or reliability analysis, and standards development.

10.2.3 Relationship to Other Programs

Research and standards development in risk methodology and regulatory analysis is intimately coupled with reactor risk. Improved methods for risk assessment and value/impact analysis are employed in reactor risk research. Reactor risk research also provides input to regulatory analysis and related standards development. New advances in PRA methods intended for direct use in licensing are first debugged in trial use in the Reactor Risk Branch. They are the prime users of advances intended principally for research applications.

Policy developments in safety goals for reactors, the use of PRA in licensing (NREP), regulation of severe accident risk, regulatory reform, and value/impact analyses for new requirements provide direct input to research in risk methodology and regulatory analysis.

The technology transfer function supports personnel development and work on licensing issues in NRR, and it will do so increasingly in IE, AEOD, and the Committee to Review Generic Requirements (CRGR).

Work in statistical methods and data analysis complements the more deterministic work of AEOD.

Work on improved PRA techniques will draw upon the human reliability research of DFO, the research on thermal shock and cold repressurization of the reactor vessel and on containment failure modes of DET, and the accident phenomenological research of DAE.

The work of the Regulatory Analysis Branch is not confined to reactor safety regulation. In the broader area of materials licensing, this research and standards development will draw upon research in transportation and materials risk and upon the work of the Division of Health, Siting and Waste Management. It will also serve NMSS.

10.2.4 Background and Status

By the fall of FY 1982, the following reactor risk assessments will be available to aid in regulatory decisionmaking and value/impact analyses:

Plant	Reactor	Containment	Sponsor	Reference
Surry*	Westinghouse	Large dry	NRC	WASH-1400
Peach Bottom*	GE	Mark I	NRC	WASH-1400
Sequoyah*	Westinghouse	Ice condenser	NRC	RSSMAP**
Sequoyah	Westinghouse	Ice condenser	TVA	--
Oconee*	B&W	Large dry	NRC	RSSMAP**
Oconee	B&W	Large dry	NSAC-Duke	OPRA
Grand Gulf*	GE	Mark III	NRC	RSSMAP**
Calvert Cliffs	CE	Large dry	NRC	RSSMAP**
Calvert Cliffs	CE	Large dry	NRC	IREP-II
Crystal River	B&W	Large dry	NRC	IREP-I
ANO-1	B&W	Large dry	NRC	IREP-II
Millstone 1	GE	Mark I	NRC	IREP-II
Browns Ferry	GE	Mark I	NRC	IREP-II
Browns Ferry	GE	Mark I	TVA	--
Big Rock Point*	GE	Large dry	CPC	--
Limerick*	GE	Mark II	Phil E	--
Zion*	Westinghouse	Large dry	Com. Ed.	ZPSS
Indian Point	Westinghouse	Large dry	PASNY-Conn. Ed.	--

*Published as of this writing.
**NUREG/CR-1659.

These studies have been or will be checked for completeness and accuracy by (1) critical peer review, (2) comparison with the historical record of severe accident precursor events in light-water reactors (close calls, initiating events, and instances of safety system failure), and (3) comparison between different PRAs of the same plant. We expect to have four plants each with two or more PRAs by the end of FY 1982.

Our understanding of the strengths and weaknesses of individual PRAs and of state of the art will mature rapidly over the next year. The number of PRAs available to the NRC is rapidly expanding. These are being subjected to peer review. The accident sequence precursor report is available in draft and will soon be published (spring 1982) and will be updated and improved in subsequent years. Updates and improvements of the data base on component failure and human error rates are being completed. Many new perspectives on severe accident phenomenology and source terms are emerging. A major effort to benchmark consequence analysis codes is soon to be completed.

We can, however, identify a number of strengths and weaknesses of reactor risk assessments today. It is clear that none of the individual plant PRAs give fully reliable estimates of the bottom-line risk posed by severe accidents at the subject plant, although the range of risks posed by commercial nuclear power plants can be bounded. Evaluations of the change in risk resulting from changes in safety system design or operation, in siting, or in emergency response practices are fairly reliable and can be made far more reliable by carefully examining the approximations and assumptions in the PRA model to which such evaluations are sensitive and by selectively upgrading these as necessary. Likewise, evaluations of the importance of contributors to risk, such as classes of initiating events, accident sequences, and system reliability, are sometimes reliable and can be made more so by careful examination.

Insofar as checks of the accuracy of PRAs have been made to date, we find that the majority of qualitative and quantitative features of the PRA models are approximately correct, but a small percentage are wrong, questionable, or seriously incomplete. These discrepancies are sufficient to cast serious doubt about the accuracy of bottom-line risk predictions, but the discrepancies are not so common as to invalidate many applications of PRA models as investigative or analytic tools.

In deference to these limitations of PRA, we anticipate that little weight will be placed upon the conformance of PRA bottom-line risk predictions with absolute safety goals in the period FY 1982-1984. Rather, we anticipate that PRA models will be used in a number of other ways that are less sensitive to the completeness and precision of the models. These include value/impact analysis of alternative designs, procedures, siting policy, and emergency planning options; the identification of the risk-limitation effectiveness of regulatory requirements; and the determination of priorities in regulation, standards development, and research. Since PRAs are more accurate in calculations of risk differences than of absolute risk, cost-benefit and importance-to-risk assessments are more trustworthy than measures of compliance with risk threshold values intended to delineate acceptable risk

We also expect PRAs to be used to identify some of the attendant risks of designs, procedures, etc. PRAs not uncommonly reveal vulnerabilities to systems interactions or common-cause failures that can degrade the risk-reduction effectiveness of designs or procedures.

In each of these applications, the PRA models will not be accepted at face value. Rather, they will be employed to identify which of the many modeling assumptions are particularly important to the principal findings. These assumptions and approximations will be examined on a case-by-case basis, and the reference PRAs will be improved or updated to develop the case for and against the validity of PRA-based inference.

A number of limitations in many or all of the published PRAs that will need to be addressed in these and future applications have been identified and research projects have been established to upgrade the reference PRAs and the state of the art for future PRAs. These are:

1. Better resolution of accident sequences

   Altered event trees to enable distinctions to be made between damaged core and molten core accidents will be drawn. In addition, more complete event sequence delineation is needed for the variety of possible interfacing system LOCAs, including steam generator tube rupture, vessel rupture in conjunction with thermal shock or cold repressurization, anticipated transients without scram, and post-LOCA failures of containment heat removal.

2. Causal analysis of initiating events

   Pipe breaks triggered by transients, active-failure LOCAs, and failures of auxiliary systems that can precipitate an initiating event as well as narrow the options for mitigating the event need to be modeled.

3. Additional sources of equipment failures

   A number of causal mechanisms of safety system failures have been neglected or poorly approximated in the published PRAs. Among these are:

   a. Environmental effects: fire, flooding, earthquakes, etc.,
   b. Sabotage,
   c. Some common-mode failures,
   d. Operator misdiagnosis leading to errors of commission during accidents,
   e. Design adequacy issues, and
   f. Blind spots in testing through which safety system faults could escape detection and repair for long periods of time.

With the methods and data available today, it is not practical to revise the PRAs to incorporate quantitative causal models of the likelihood and severity of these failure mechanisms for every sequence or system. In the next few years, reference PRAs will be selectively modified to enable sensitivity studies to be made on the likely effect of these failure mechanisms on value/impact results obtained with the PRA models. However, research to close these loopholes* in PRA is planned for the FY 1984-1988 period.

The background of the regulatory analysis function lies in the requirement for the periodic and systematic review of the regulations,** requirements for

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*The prospects of closing these loopholes in PRA to measure compliance with risk-threshold safety goals will be dealt with in the forthcoming staff paper on safety goal implementation.

**TMI Action Plan Section IV.G.2, NUREG-0660.

better value/impact analyses for regulatory initiatives, the prospect for technical as well as administrative regulatory reform, and the need for a center of expertise in the processing of new rules and regulations.

Prior work on the relevance to risk of the 133 generic safety issues in 1978 and the value/impact assessment of the pre-TMI standard review plan demonstrate the feasibility of analyses of the risk-limitation effectiveness of many kinds of regulations. In addition, recent progress in translating risk-assessment results into projections of the present worth of projected losses makes it clear that major strides are possible in measuring the value of regulatory changes to reduce risk. It is also clear that the disciplines of operations research and engineering economics offer as yet untapped resources to measure the impact of regulatory changes. Thus, the prospects are bright to develop the research foundations for technically substantive reactor safety regulatory reform which could enhance the assurance of safety while diminishing the direct and indirect burden of compliance on licensees.

The technology transfer function has been served in a variety of ways. Actual reactor risk assessments, methodology, and data have been transmitted to the licensing offices and licensees. Technical assistance has been provided to the regulatory offices on a wide variety of issues ranging from hearing testimony to bases for the resolution of generic safety issues. PRA training courses open to all NRC personnel have been sponsored. However, the trend toward the adoption of PRA as a licensing or regulatory tool has intensified the need to upgrade the training of NRC personnel in the results to date, the strengths and weaknesses, and practitioners' skills.

10.2.5 Research Program Plan

10.2.5.1 Technology Transfer Program

Initially, this program will concentrate on the application of PRA techniques to LWR. Principal activities will include formulating a curriculum for training selected NRC staff members in the use of PRA techniques, developing training aids, conducting training courses, and developing PRA training documentation for future reference and use.

In the initial phase, training will be given to approximately 250 management and staff members each year. Training methods will include management briefings, formal classroom instruction, and instructor-student intermediate and advanced workshops. 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, human reliability analysis, system interactions, and common-cause analysis. Training in improved techniques in PRA methodology will be incorporated throughout the training program as these techniques are developed and proved successful.

Beginning in FY 1984-1986, training will be expanded to cover other major aspects of the nuclear fuel cycle (including milling through fabrication, transportation, reprocessing, and waste management) and other reactor types such as HTGRs and liquid metal fast breeder reactors (LMFBRs). This phase should be completed during FY 1988-1989.

Technical assistance to NRR, NMSS, IE, AEOD, and CRGR will continue during FY 1984-1988.

10.2.5.2 Human Reliability Research

This project is budgeted in this decision unit but performed in DFO (see Section 7.1).

10.2.5.3 Fire Risk Assessment Methods

Prior efforts to develop models of reactor accident initiation and propagation via in-plant fires will result in two procedure guides. The first, to be largely completed in FY 1983 and finished in FY 1984, is intended for use in IREP-like studies. It will provide for the efficient screening of plants to identify whether fires may constitute a dominant or significant contribution to risk. The second, to be refined in FY 1984-1986, is intended for intensive, comprehensive PRA efforts in which accuracy and thoroughness is pursued at the expense of economy of effort. The latter technique will be employed in the revised Reactor Safety Study.

10.2.5.4 Flood Risk Assessment Methods

Prior work on flood risk screening and risk evaluation will be supplemented and continued as follows:

1. All proposed or real reactor sites will be characterized by the types of threat posed by floods or severe storms, i.e., threats to offsite power, to the ultimate heat sink and circulating water system, to the turbine building, and to the auxiliary buildings by, for example, riverine floods, tsunamis, hurricanes, and tornadoes. The thresholds at which these entities are threatened by floods or storms will be compared with the maximum credible events for the site. The objective of this task is to identify the aspects of the problem that are important to risk at current or proposed plants.

2. An analysis of warning times and of conditional probabilities of core melt given the loss of the threatened functions and a study of mitigation strategies will be developed for the more important threats found in item 1.

3. For the more risk-significant flood or storm scenarios at particular sites, the frequency vs. severity relationship for the initiating event will be researched. A value/impact assessment will be made of the value of research to narrow the uncertainties in the initiating event frequency and of the prospects for success in such research as a function of cost. Where such value/impact studies suggest that additional research on the frequency of occurrence is plausibly cost effective, a research program will be instituted.

4. Improvements will be made in the risk screening methods for both in-plant and external flood hazards, leading to a standardized, efficient method for incorporating flood risk consideration in future IREP-like reactor risk assessments.

5. The tasks in 1, 2, and 3 will result in determining in which plants, if any, do offsite floods or storms plausibly pose a dominant contribution to the risk. The mitigation strategies proposed in task 2 will be subjected to refined value/impact assessment using the results of task 3. The value of a dedicated shutdown cooling system, which will have been assessed in the severe accident research programs in FY 1983, will be reevaluated in light of the more incisive information developed in this program, the fire risk program described above, and the sabotage study described below.

10.2.5.5 Risk-Limitation Effectiveness of Sabotage Safeguards

Although a synthetic reactor risk assessment considering sabotage initiators is made impractical by the absence of predictive models of sabotage likelihood, PRA methods are not without resources to evaluate some aspects of sabotage risk. By the end of FY 1983, the ASEP will have identified which accident sequences in the many reactor PRAs then available might plausibly be caused by insider or external sabotage. This preliminary work will be expanded by an assessment of the possible intent or objectives of saboteurs, an analysis of the risk-limitation effectiveness of current or proposed strategies for reducing sabotage risk, and a study of the attendant risks associated with these strategies. Products of the research will include recommendations for regulatory changes, if any, and a value/impact assessment of the options.

10.2.5.6 Operator Errors of Commission

Confusion or misconceptions of ongoing incidents at nuclear power plants can lead to operator error that is not accounted for in current PRAs. In addition, faults or alarms that do not directly threaten the principal safety functions may nonetheless distract operators and reduce the likelihood of focused diagnosis and repair of those faulted conditions that may actually threaten the principal safety functions.

Prior PRAs that detail the evolution of the symptom profile, the severe accident sequence analysis program, and the human reliability research in DFO will be used to provide the basis for methodology development to enable future PRAs to consider operator errors of commission or degraded repair rates. The products of these efforts will be of two kinds: detailed procedures for a thorough analysis of such effects on human reliability in the RSS-II, and efficient screening procedures for use in IREP-like studies to identify which sequences may be made more likely by operator errors of commission.

10.2.5.7 Improved System Reliability Analysis

Research to improve upon current capabilities in system reliability analysis, system interaction analysis, common-cause failure, 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 system interactions and common-cause failures in licensing. In addition, the advent of quantitative safety goals and the plan to conduct a new state-of-the-art reactor safety study necessitates a coordinated attack on the methodological problems in accident sequence likelihood analysis. A focus of the research into improved system reliability analysis techniques will be research to meet this need. Various reliability engineering techniques that show promise for system 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 individual and common-cause 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. Other system faults or design errors may escape detection and repair in surveillance testing. 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 IE in focusing on the more effective ways of reducing reactor accident risks, as well as improving safety analysis and risk assessment.

10.2.5.8 Reliability Data Analysis

In the current rulemaking on revisions to the requirements associated with LERs, the agency has elected to leave to the Institute of Nuclear Power Operations (INPO) the design and implementation of a reliability data system to replace the defunct Nuclear Plant Reliability Data System (NPRDS) and LER-based reliability data analysis efforts previously supported by RES.

In light of the increasing emphasis the Commission is placing upon PRA and probabilistic system reliability analysis, the availability of a trustworthy reliability data bank will become increasingly important to agency decisionmaking.

We can envision two extreme-case scenarios and a continuum of intermediate possibilities:

1. INPO succeeds in creating a sound reliability data system, and the voluntary participation of utilities is high enough that the data base is trustworthy. In this event, RES will maintain a small-scale liaison and quality verification role.

2. INPO fails to create a sound reliability data base or the voluntary reporting is as poor as that recorded by NPRDS. The NRC concludes that it

is essential to reconstruct a reliability data base by auditing plant log books and to simultaneously define mandatory reporting or reliability-related experience. In this event a massive commitment of staff and program support funds would be required. Past experience with the use of plant log books to obtain failure rate data suggests costs in the neighborhood of $100,000 per plant for pump and valve data alone.

In either event, continuing analysis of failure rate data will be necessary to accommodate the needs of new methods and models for the study of common-cause failure, to evaluate anomalies in the reported data, and to pattern analysis for failure trends throughout the FY 1984-1988 period.

10.2.5.9 Safety Goals and Value/Impact Research

By FY 1984, considerable experience will have been accumulated in the use of safety goals, risk-based licensing criteria and risk-based value/impact assessments. PRA techniques will have matured significantly in the interim, and the agency may be ready to allow risk-based safety analysis to take a larger role now filled by some of the deterministic requirements. For example, the selection of allowable outage times, the evaluation of surveillance test and maintenance procedures, and the granting of exemptions from deterministic design requirements may be based on PRA.

We anticipate a continuing need for the developm[illegible] of decision criteria to harvest the fruits of risk assessment developments and experience with qualitative safety goals.

10.2.5.10 Reverse Risk Assessment

Reactor risk assessments build up from models of the frequency of individual component failures a coherent model of accident likelihood, course, and consequences. The PRA provides an integrated summary of what we think we know about accident origins, likelihood, phenomenology, and consequences. PRAs provide an excellent point of departure for a study of which factors might give rise to far more risk than we expect or that PRAs predict.

This research project will trace the logic of several reference PRAs in the reverse direction, starting with consequences, fission product dispersion, source-term calculations, etc. Its objective is to catalog those features, assumptions, models, etc., that might harbor unpleasant surprises that could result in far higher risk than PRAs predict or the Commission's safety policy accepts.

The project complements PRA uncertainty and sensitivity analysis by providing an alternative way of formulating questions about completeness and accuracy in PRAs. One is more likely to discover serious errors in phenomenological modeling or incompleteness in the accident sequence catalog if one asks pointed questions of the kind that could be suggested by the reverse PRA logic.

The product will be a catalog of the factors that would result, if realized, in revising upward the PRA-predicted risk to well over the Commission's safety goals. It is anticipated that several divisions of RES will participate in a followup evaluation of the strengths and weaknesses of the evidence that these high-risk factors are not present in reactor accident phenomenology, safety function design adequacy, or safety function reliability.

A second phase of the reverse risk assessment will catalog the ways in which accident phenomenological factors might limit the risk to a value far less than that predicted by the PRAs or accepted in the Commission's safety goals. This work will help put to use results of the experimental research planned to support the severe accident rulemaking.

The end uses of these catalogs of risk-limiting or risk-increasing factors will be in research priorities, in revisions to our understanding of absolute reactor risks (RSS-II), and in the validation of regulatory positions with respect to severe accidents.

10.2.5.11 Periodic and Systematic Review of the Regulations

The NRC has committed to the periodic and systematic review of the regulations (TMI Action Plan, IV.G.2, NUREG-0660). By FY 1984, a study of the risk limitation effectiveness of the general design criteria, other selected parts of 10 CFR Part 50, the risk-related regulatory guides, and the current standard review plan will be completed.

An early draft of a study to catalog the ways in which a power reactor might pose risks well in excess of the Commission's safety goals or findings of PRAs will also have been prepared in FY 1983. The reverse risk assessment project will further develop this approach.

In FY 1984, these studies will be updated in light of the maturing library of reactor risk assessments and will be supplemented by a study of the effectiveness with which industry practices in complying with the regulations detect and correct potentially risk-significant problems in detailed design, procurement, construction, testing, surveillance-testing, and other forms of operational experience feedback.

The results of the early (pre-1984) studies will be used in the severe accident rule, a draft of which is expected in a Commission paper at the end of FY 1983.

Followup research and standards development will track the rulemaking into FY 1984 and result in an action plan to develop and evaluate options for more extensive reactor safety regulatory reform in the mid-1980s.

10.2.5.12 Risk-Based Priorities for Research

This research program will develop methods for, and conduct pilot-study applications of, setting research priorities based on risk assessment. Methods will be developed for use at four stages of research planning: (1) identification of subjects warranting research, including early definition of research objectives; (2) risk-based value/impact analyses of existing research conceptions and facility utilization; (3) risk-based value/impact analysis of the need for standards development, and (4) risk-based midcourse corrections and peer review of ongoing research and standards development projects.

Among the approaches to be explored will be (1) translation of uncertainties in PRAs into value assessments for the technical unknowns; (2) estimation of the cost effectiveness with which uncertainties can be reduced; (3) decision theoretic assessments of the desirability of NRC research, research requirements placed upon licensees vs. regulations that render the unknown factors unimportant to risk; and (4) options for RES implementation. The reverse PRA research (described above) is expected to be a particularly fertile source of perspectives on the importance to risk of weak spots in our understanding of accident likelihood and phenomenology.

Products of this research will include options for consideration by office management for several types of risk-based setting of research and standards priorities with example case studies.

10.2.5.13 Methodology for Accident Sequence Precursor Studies

Several problems in statistical inference and risk modeling constrain the accuracy of inferences about reactor safety from a statistical analysis of "close call" events. These include the assessment of the conditional probability of a severe accident given a precursor event, the evaluation of precursors at different levels of accident development, the treatment of repair prior to the critical fault duration time, the inhomogeneity of the population of plants from which the accident precursor records are drawn, and the sampling biases in the precursor identification. These will be under study and methodology development in FY 1983-1986.

10.2.5.14 Initiating Event Data Analysis

In addition to the studies of initiating events in the fire and flood risk research projects, a broad-scale effort will be conducted in the FY 1982-1984 period, with subsequent followup, to develop a data base for reactor accident initiating events including transients, LOCA, transient-induced LOCA, transients initiated by auxiliary system failures, and plant conditions conducive to common-cause failure.

10.2.5.15 Improvements on the Single-Failure Criterion

The research to support the severe accident rulemaking will partially address the inadequacy of the single-failure criterion, particularly as it is reflected in insufficient severe accident prevention and mitigation in operating plants. This work is expected to be concluded at the end of FY 1983. It will include the development of options to rectify safety-significant deficiencies through design requirements, system performance standards, and reliability/diversity requirements. The principal focus of this research will be on the necessity or desirability of backfits on operating plants or those in the pipeline.

In FY 1984 and 1985, this work will be used in draft regulations intended for plants not yet designed. The principal objective of this research and standards development will be to close loopholes in the regulations, particularly the single-failure criterion applied to design basis accidents, that might permit unacceptably high susceptibility to severe accidents to slip through the licensing process.

10.2.5.16 Assessment of Foreign Reactor Safety Requirements

A library will be assembled of the reactor safety requirements in other nations dealing with design basis accidents and system reliability assurance requirements. These will be assessed for applicability to the domestic problem of regulating severe accident risk.

10.2.5.17 Options for Regulatory Reform

In FY 1984-1985, a study will be conducted of the options for, and pros and cons of, more far-reaching reactor safety regulatory reform than that emerging from the severe accident rulemaking or its equivalent. This study will be updated, and the more promising options will be further developed in subsequent years. A particular focal point of this effort will be the use of results from studies of reliability assurance practices in nonnuclear industries (see Section 10.1.5).

10.3 Transportation and Materials Risk

10.3.1 Issue

The principal general issue associated with regulation of the nuclear fuel cycle and the uses of radioactive material is the adequacy of the regulatory system for protecting the public health and safety and the public understanding of this system and its technical basis. More specifically, the issues addressed here are:

1. What are the risks associated with all aspects of the nuclear fuel cycle (except reactors) and with industrial, medical, academic, and public uses of radioactive materials, and how can PRA and reliability assurance methods improve the regulatory process?

2. Can research be performed to develop data, analysis results, and PRA methods that will either verify that current regulations are adequate and appropriate or identify needed improvements associated with transportation, manufacture, or use of radioactive materials?

3. Does experience in transportation, manufacture, and use of radioactive materials under both normal use and accident conditions indicate a need to revise existing regulations?

10.3.2 Research Program Objective

The objectives of the transportation and materials risk program are to:

1. Employ PRA and reliability assurance methods for all nonreactor operations in the nuclear fuel cycle and with uses of radioactive materials (both licensed and exempted) as a basis for setting priorities for future research and regulation development activities.

2. Develop a technical basis for judging the adequacy of the existing regulations. This will involve collection, analysis, and validation of

operational data on fuel cycle facilities and on transportation and uses of radioactive materials and the development of new data and analysis methods as needed.

3. Develop and promulgate improved regulations, standards, and guides as appropriate. Analytic methods for licensing safety evaluation and inspection methods will be developed as needed.

4. Coordinate activities with other Federal agencies (e.g., DOT, DOE, EPA), national organizations (e.g., ANSI, ANS), States and regional organizations, international organizations (e.g., IAEA, OECD, NEA, individual government agencies), and with commercial organizations (e.g., EPRI). The objective is to ensure to the greatest possible extent that:

   a. NRC regulations are compatible with regulations of other organizations, where appropriate,

   b. Standards development by expert groups is encouraged in areas of interest to NRC and adopted by NRC when appropriate, and

   c. Research results and relevant information on design and operational experience are exchanged and there is cooperative planning of future work.

The objectives identified above apply in part or in entirety to three NRC-regulated activities, i.e., fuel cycle, transportation, and radioactive materials as shown below.

Activity* Objective	Fuel Cycle	Transportation	Radioactive Material Uses
Risk Assessment and Reliability Assurance	X	X	X
Technical Basis for Regulation		X	X
Develop Regulations		X	X
Coordination		X	X

*Each activity is described separately in Section 10.3.3.

10.3.3 Relationship to Other Programs

10.3.3.1 Fuel Cycle

An ongoing NRC research program, "Fuel Cycle Facility Accidents," can provide input to these programs. The objectives of that program are to develop selected in-plant consequence models and experimental data for selected LWR fuel cycle elements. The study is strictly a consequence study that does not cover all the components of a PRA study. It is described in Section 6.9.

Important programs on PRA methodology development for high-level waste (HLW) disposal in deep geologic media are described in Chapter 8, "Waste Management." The NRC is also exploring the possibility of the regulation of processes by which licensees ensure the public health and safety and are pursuing the development of reliability assurance methods to apply to reactors. Some of these methods may be applicable to other elements of the LWR fuel cycle.

10.3.3.2 Materials

The fuel cycle consequence program identified in 10.3.3.1 will be extended to develop in-plant consequence models and experimental data for selected facilities using radioactive materials. This program is closely related to the radioactive materials risk program, and the two programs are coordinated to start at the same time. The consequence study is described in Section 6.9.

10.3.3.3 Transportation

Several programs addressing the transportation safety issue are included in the Reactor and Facility Engineering Decision Unit. Specifically, the objectives of these programs are to (1) ensure that structural performance standards for packages are based on realistic transportation environments and (2) verify existing analytical methods or develop new analytical methods for application to structural, thermal, criticality, or shielding evaluations of shipping packages. Coordination with established national groups and other regulatory agencies in these specific technical areas is accomplished under this decision unit.

10.3.4 Background and Status

10.3.4.1 Fuel Cycle Risk

The fuel cycle risk assessment program was initiated in FY 1981 for both uranium and plutonium recycle and the once-through LWR cycle with disposal of spent fuel.

The program consists of two phases. The first phase, a scoping analysis to be completed in FY 1982, will use existing methods and data where available. This analysis will estimate risks for each fuel cycle element and rank the risks on a consistent basis. Along with a determination of the adequacy of existing analytical tools, the risk ranking will provide a basis for setting priorities for the fuel cycle elements to be assessed in the second phase of the program scheduled to begin in FY 1983.

In 1982, the reference LWR fuel cycles will be identified and described. The risks of the elements will be ranked on a consistent basis using information from previous fuel cycle risk and safety studies. Priorities will be based on the risk rankings of the elements and the immediate needs of NMSS. The results of the scoping analysis will direct the research for FY 1983 through FY 1988.

In 1983, the effort will be concentrated on the first LWR fuel cycle element identified in the scoping analysis. A comprehensive PRA method will be developed with the following PRA components: data collection and analysis, external events, system description, human factors analysis, accident scenario analysis, probabilistic modeling, model development for in-plant and out-of-plant consequences, uncertainty and sensitivity analyses, and risk measure analysis. The products will be methods and computer codes, and their usefulness will be demonstrated in developing risk-based regulatory standards and in the licensing process. PRA method development for this element will extend through FY 1984.

10.3.4.2 Materials

During FY 1982-1983, the development of standards and guides related to the use of radioactive materials will continue. The development of these standards and guides necessitates studies of the bases for regulatory actions.

Consistency of regulatory action with current experience in using radioactive material and the identification of unnecessarily burdensome regulations will continue to receive priority treatment. Development of improved requirements for the manufacturers, vendors, and users of sealed sources and devices will continue. The process by which petitions for rulemaking are filed and handled will be studied, particularly regarding the contents of the supporting statement that must be filed with the petition. Coordination of NRC regulations and guides with those of national and international groups will continue. Among these are suggested standards for irradiators being developed by ANSI and standards for well-logging being developed by the Conference of Radiation Control Program Directors.

A study will be initiated in FY 1983 to evaluate risks associated with uses of radioactive materials. The first phase of a scoping study will be performed, principally to determine if relevant risk information already exists, to tentatively assess the relative risks among the wide range of materials activities, and to develop a plan for performing the final risk assessment.

Finally, a study of the generic issue of recycling scrap from operating and decommissioned nuclear operations, e.g., nuclear power plants and fuel cycle facilities, will be initiated. This study will identify the type and quantity of scrap, the type and quantity of contamination, and the potential hazards to the biosphere for recycle options. This study will culminate in the development of agency policy in this area.

10.3.4.3 Transportation

Specific areas being addressed in FY 1982-1983 include collection and analysis of shipping and incident data on radioactive materials. Work will be initiated during the period on improving NRC/DOT incident reporting requirements;

evaluating risks and other impacts of the transportation of radioactive material, especially spent fuel and large-quantity shipments through urban areas; reevaluating the adequacy of current package testing and acceptance requirements in light of potential environments associated with severe transportation accidents; developing guidance for responding to radioactive material transportation emergencies; and changes in regulations governing shipment of low specific activity (LSA) materials, particularly LWR resins.

10.3.4.4 Risk Validation/Verification and Information Programs

A continuing peer review of all the aforementioned efforts on analyses of transportation, material, and fuel cycle risk analysis efforts will be conducted. Peer review is considered especially important for PRA because professional and expert opinion plays such a key role in all aspects of PRA, including scenario selection, data evaluation and selection, model development and deployment, and interpretation of results.

10.3.5 Research Program Plan

10.3.5.1 Fuel Cycle Risk

The objectives of this program are to:

1. Provide perspectives on relative risks and uncertainties associated with all elements of the fuel cycle (scoping analysis),

2. Provide PRA methods suitable for use as licensing safety analysis tools where appropriate,

3. Provide perspectives and techniques for use in regulatory standards development, and

4. Develop a cadre of fuel cycle risk-assessment practitioners.

The elements of the LWR fuel cycle to be considered in this program are mining and milling, refining and conversion, enrichment, fabrication, storage, reprocessing, refabrication, waste disposal, and transportation. It is anticipated that the advanced fuel cycles will have the same elements. All development effort on PRA methods will consider existing PRA and deterministic studies when appropriate.

PRA method development for the second and third LWR fuel cycle elements will be initiated in 1984 and will be completed in 1985. The level of research for these elements and subsequent elements will be the same as that described in 10.3.4.1 for the first element. Development will be completed in FY 1985. In addition, research on the advanced fuel cycles will be initiated. The reference LMFBR fuel cycles will be identified, and their elements will be described. A scoping analysis similar to that for the LWR fuel cycles will be performed. This is particularly important for LMFBRs because DOE has chosen to use safety design techniques closely related to PRA for LMFBR research and development.

PRA methods will be developed for two LWR fuel cycle elements each fiscal year from 1985 through 1987. Each development will take 2 years with completion of all PRA methods development in 1988. Also during this period, PRA methods will be developed or modified on the basis of the LMFBR scoping analysis. Completion of the LMFBR PRA methods development will extend beyond 1988. In 1988, the reference HTGR fuel cycle will be identified and a scoping analysis similar to the LMFBR and LWR programs will be performed.

10.3.5.2 Fuel Cycle Reliability Assurance

The objectives of this program are to:

1. Develop evaluation methods to assess the reliability of fuel cycle systems;

2. Provide reliability assurance (RA) methods for use in regulations and standards development; and

3. Provide techniques for use in the fuel cycle licensing process.

To ensure that public health and safety goals that may be established for fuel cycle operations are met, it will be necessary to be able to evaluate the reliability of the elements of each fuel cycle to determine whether these goals can be met. In addition, it will be necessary to determine whether the system costs required to achieve the reliability goal would favor a particular fuel cycle.

In FY 1984, the alternative fuel cycles to be evaluated will be identified. Evaluation of fuel cycle elements will be considered in the order of risk significance and NMSS needs. Available RA methods will be assembled and evaluated for potential applicability to the fuel cycle. Those RA tools most appropriate for evaluating facility design and operation, facility control systems, processes, human factors, and costs will be identified. Preferred analysis methods and the basis for their selection will be discussed.

In FY 1985, a detailed analysis of the most important fuel cycle element will be conducted. Stated NRC safety goals will be the basis for establishing reliability goals for the selected fuel cycle element to be analyzed. A complete system description, event sequence descriptions, and failure-mode descriptions will be developed. System performance equations will be developed using these descriptions and will be tested analytically with available or assumed component reliability data. Reliability requirements for system elements to meet the safety goals will be determined.

In FY 1986 and FY 1987, the reliability assurance methods that have been tested analytically and have been determined to be the most suitable will be used to develop computer codes that can be exercised for each element of the fuel cycle. Those elements of the fuel cycle that will be unable to meet the desired reliability safety goals will be identified. During FY 1987 and FY 1988, each of the alternative fuel cycles will be completely evaluated and compared to determine which ones can meet or exceed the reliability goals.

10.3.5.3 Materials

During FY 1984-1988, tasks started during FY 1982-1983 will be completed and additional tasks in the areas of standards development, data collection, and environmental impact studies will be initiated.

Major areas of investigation and accomplishments during this period will be:

1. Completion of development of PRA methods for assessing risk and consequences associated with uses of radioactive material that involve high risk or potentially large consequences.

2. Evaluation of alternatives and their impact related to unlicensed nuclear products, i.e., recycling contaminated scrap from nuclear operations, exempting neutron-activated products, and exempting contaminated building materials;

3. Systematic review of 10 CFR Parts 31, 32, and 33 based in part on studies being carried out by NMSS;

4. Development of environmental impact assessments and regulations as needed for proposed new uses of radioactive materials.

5. Development or modification of regulatory guides as needed. For example, we anticipate the need for new or updated guides on such subjects as nonmedical uses of sealed gamma ray sources, classification of containment properties of sealed radioactive sources, design and performance standards for radiographic source changers, design of high-level radiator installations, and design and production of gauges; and

6. Review of procedures for filing supporting statements accompanying petitions for rulemaking.

10.3.5.4 Transportation

1. Regulation Adequacy

In ensuring the continuing adequacy of existing regulations and licensing procedures, several tasks will continue into, or will be initiated in, FY 1984-1988. These tasks fall into two distinct classes. The first of these classes attempts to define transportation requirements in terms of the ALARA concept. Studies in this class are under way or planned to review allowable external contamination and radiation levels for shipping packages for radioactive materials. In addition, assessment of operational methods that have the potential to reduce occupational radiation exposures to transport workers is also planned in 1985. Rulemaking activities that support regulations suggested by these projects will be initiated. The second class of tasks is related to the upgrading of NRC's current estimates of the impacts of transportation operations. This assessment will continue over the near term to form bases for the Commission's regulations. Specifically, an upgrading of the generic risk evaluation of transportation (NUREG-0170) is scheduled for FY 1983-1984. Key issues in the transportation of radioactive materials are the consequences of severe accidents and the credibility of large impacts on public health and safety. To date, risk analyses have had limited success because, among other things, there is a large amount of uncertainty associated with key technical information, e.g., package failure threshold and dependence of release fraction on accident severity and the probabilistic characterization of the accident environments. A research program to reduce uncertainties in the current risk methodology by developing this needed information will be undertaken. The revision of NUREG-0170 will use an updated shipment data base that is scheduled for completion prior to 1984 and applicable information from the NRC transport incident data collection system that was initiated in FY 1982.

A project will be initiated in FY 1986 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.

2. Value/Impact of Regulatory Change

Programs are under way to assess the need for and effectiveness of possible modifications to the current regulations. One ongoing effort, the "Modal Study," was initiated in FY 1980 with the objective of assessing improvement to public health and safety that could be effected through changes in current package performance tests and posttest acceptance criteria. The proposed changes would make the package performance tests for large-quantity shipments (e.g., spent fuel) more representative of conditions expected in the unlikely event of extremely severe road, rail, air, and marine accidents. Posttest package acceptance criteria (release rates) would also be reviewed with the intent of limiting accident consequences or risks. By 1983, performance tests representative of these extremely severe accident environments will be defined, and testing of generic packages will be started. These test results should provide information necessary to determine potential ALARA criteria. In the 1984-1987 period, generic package testing will be completed, and appropriate cost/benefit analyses will be carried out to establish cost-effective posttest package criteria. These studies will consider the impacts of alternative package designs as well as operational and administrative controls on the transport system.

It is expected that, by 1985, shipping requirements for advanced fuel cycle materials will be sufficiently well defined to enable RES to initiate confirmatory research studies in this area. 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, leakage characteristics, and safety implications of the physical and chemical forms of the material shipped

3. Shorter-Term Licensing and Compliance Assistance

In support of shorter-term licensing and compliance activities, RES issues regulatory guides that (1) define acceptable methods of analysis to assess package designs against regulatory criteria and (2) define acceptable methods of complying with specific regulatory requirements. Continuing annual staff commitment is planned to be directed toward modifying existing guides, developing new guides, and supporting rulemaking activities throughout the FY 1984-1988 period.

4. Coordination and Periodic Review

The RES review of transport regulations and the coordination with other agencies and standards bodies is carried out both on a regular and on an ad hoc basis. An OMB-mandated review of all NRC regulations is required at 3-year intervals. Separate periodic reviews of data-gathering requirements (e.g., NRC's transportation incident reporting requirements) is also planned. Major rulemaking proceedings planned for FY 1982-1985 will cover the use of DOT specification packages, a class of packages authorized by DOT regulations before NRC's certificate of compliance system came into being. This effort will require research to assess the benefits and impacts associated with discontinued use of DOT specification packages. Additional periodic rulemaking takes place every 10 years (next revision scheduled for 1985) to make NRC regulations compatible with those of DOT and IAEA.

10.3.5.5 Risk Validation/Verification and Information Programs

The objectives of the program are to provide on a generic basis (1) administrative procedures designed to ensure the adequacy, accuracy, and applicability of the PRA methods developed and deployed and (2) tools, methods, and data bases needed to perform or improve PRA practices. Currently the validation/verification programs consist of:

1. Continuing systematic peer review,
2. Validation of PRA,
3. Development of economic cost consequence models, and
4. Dissemination of research results to the public in a clear, understandable form.

The peer review program will begin in FY 1982 with the development of a procedures plan. In each subsequent fiscal year, a significant PRA program subelement will be given global peer review as appropriate.

As risk analytical tools and their application to nuclear activities begin to acquire a degree of maturity, validation and verification of the methodologies will begin. This will require a thorough review of the new data bases and analytical tools and their application to the specific fuel cycle problems addressed.

A series of research programs will be initiated to develop methods for evaluating the economic costs associated with accidental release of radioactive material to the environment. The development of the method for modeling economic cost will begin in FY 1984 with a scoping evaluation of the current state of the art. A large effort model development will follow in FY 1985, with a closeout effort in FY 1986 consisting of final documentation of the model and verification of the computer code.

Results from the risk, reliability assurance, and economic consequence studies will be summarized and presented in information packages that are easy to understand.

A program will be initiated in FY 1984 that will make available to the public factual information concerning fuel cycle risk, transportation, and radioactive material uses through public presentations and written material. Topics have not been selected, but could include:

1. Summary of normal radioactive shipping data, population exposures estimated for these activities, and estimated risks,

2. Transportation incidents, their consequences and frequency of occurrence, and a comparison of the total risks involved in transport of radioactive material vs. transport of other hazardous material,

3. A comparison of the risks throughout the LWR fuel cycle, definition of the areas producing the greatest risk, and a summary of research programs proposed to reduce risk or uncertainty in these areas,

4. Recycle of contaminated materials and definition of the proposed approach and alternatives, and

5. Exempt uses of radioactive materials, definition of the uses, radioactive materials used, number of units being distributed, estimated population risk, and safety requirements.

GLOSSARY

Acronyms and Initialisms

ACRR	Annular Core Research Reactor
ACRS	Advisory Committee on Reactor Safeguards
AE	Acoustic emission
AECL	Atomic Energy of Canada, Ltd.
AEOD	Office of Analysis and Evaluation of Operational Data, NRC
AIF	Atomic Industrial Forum
AIPA	Accident Initiation and Progression Analysis
ALARA	As low as is reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOT	Anticipated operational transient
ASEP	Accident sequence evaluation program
ASTM	American Society of Testing Materials
ATOG	Abnormal transient operator guideline
ATWS	Anticipated transient without scram
BCL	Battelle Columbus Laboratory
BE	Best estimate
BMFT	Bundesminister fuer Forschung und Technologie
BNL	Brookhaven National Laboratory
BOP	Balance of plant
B&W	Babcock and Wilcox

CAIS	Coded aperture imaging system
CCTF	Cylindrical Core Test Facility
CDA	Core disruptive accident
CDC	Control Data Corporation
CE	Combustion Engineering
CEA	Commissariat a l'Energie Atomique, France
CHAP	HTGR system transient analysis code
COMMIX	Three-dimensional, transient, thermal-hydraulics code
CONCEPT	Cost-estimating code for nuclear power plants and coal-fired steam supply systems
CONTAIN	Containment analysis code
CORCON	Code that models interaction between molten core materials and concrete during core-melt accidents
CORRAL	Code that models behavior of fission products in containment atmosphere
CORTAP	HTGR system transient analysis code
CRAC	Calculation of Reactor Accident Consequence (code)
CRBR	Clinch River Breeder Reactor
CRC	Corrosion-resistant cladding
CRGR	Committee to Review Generic Requirements (NRC)
CSTF	Containment Systems Test Facility
DBA	Design basis accident
DCC	Degraded core cooling
DEI	Dose equivalent index
DOC	Department of Commerce
DOD	Department of Defense
DOE	Department of Energy

DOT	Department of Transportation
DTPA	Diethylene triemine pentaacetic acid
ECCS	Emergency core cooling system
EDF	Electricite de France
EEI	Edison Electric Institute
EM	Evaluation model
EMI	Electromagnetic interference
EPA	Environmental Protection Agency
EPR	Ethylene-propylene-rubber
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
ESFAS	Engineered safety feature actuation system
ESSOR	EURATOM'S 50mw(th), organic-cooled, heavy-water-moderated, experimental reactor at Ispra, Italy
EXMEL	Model for fuel and clad melting developed at Stuttgart
FAA	Federal Aviation Agency
FAST	Fuel aerosol simulant test
FASTGRASS	Code that models release of fission products from fuel during transient heating events
FDA	Food and Drug Administration
FEIS	Final environmental impact statement
FEMA	Federal Emergency Management Agency
FFTF	Fast Flux Test Facility
FIST	Full integral simulation test
FITS	Fully instrumented test series
FLECHT-SEASET	Full-length emergency cooling heat transfer-separate effects and system effects tests

FM	Factory Mutual Research Corporation
FRAP	Fuel rod analysis program (code)
FRAPCON	Steady-state analysis of fuel rod response during normal reactor operation (code)
FRAP-T	Fuel rod analysis program - transient (code)
FRG	Federal Republic of Germany
FSAR	Final safety analysis report
HAARM	Mechanistic aerosol behavior code (uses log-normal aerosol-size distribution)
HAZARD	Code that assesses seismic hazard and develops information to generate time histories
HDR	Heissdampfreaktor (a decommissioned steam reactor in FRG where reactor safety experiments are conducted)
HECTR	Improved multicompartment deflagration code
HEDL	Hanford Engineering Development Laboratories
HIACA	High adjustable cobalt array (facility)
HLW	High-level waste
HSST	Heavy section steel technology
HTGR	High-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
IBLOCA	Intermediate-break loss-of-coolant accident
I&C	Instrumentation and control
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IDCOR	Industry Degraded Core (program)
IE	Office of Inspection and Enforcement, NRC

IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular stress-corrosion cracking
IHSI	Induction heating stress improvement
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IREP	Interim Reliability Evaluation Program
IRRC	Interagency Radiation Research Committee
ISD	Instructional system development
ISI	Inservice inspection
ISP	International Standard Problems
JAERI	Japanese Atomic Energy Research Institute
JPVRC	Japanese Pressure Vessel Research Committee
KAVERN	German code for melt/concrete interactions
K-FIX	Thermal-hydraulic component code
Kfk	Kernforschungszentrum, nuclear research center in Karlsruhe, FRG
LANL	Los Alamos National Laboratory
LER	Licensee event report
LET	Linear energy transfer
LEU	Low-enrichment uranium
LLW	Low-level waste
LMF	Large-Melt Facility
LMFBR	Liquid-metal-cooled fast-breeder reactor
LOCA	Loss-of-coolant accident
LOFT	Loss-of-Fluid Test

LOF/TOP	Loss of flow/transient over power
LSA	Low specific activity
LSP	Lead standard plant
LTS	Low-temperature sensitization
MAEROS	Mechanistic module of CONTAIN code
MARCH	Code that analyzes core meltdown phenomena
MATADOR	Code that models fission product behavior in LWR containments (replaces CORRAL code)
MELCOR	Planned code to model meltdown accident assessment (will include CRAC-2 and MATADOR codes)
MSLB	Main steamline break
NAS	National Academy of Sciences
NAUA	German model for aerosol behavior in containments
NDE	Nondestructive examination
NDT	Nil-ductility temperature
NEA	Nuclear Energy Agency
NEPA	National Environmental Policy Act
NESC	National Energy Software Center
NMSS	Office of Nuclear Material Safety and Safeguards, NRC
NPRDS	Nuclear Plant Reliability Data System
NREP	National Reliability Evaluation Program
NRR	Office of Nuclear Reactor Regulation, NRC
NRU	Natural-uranium, heavy-water moderated and cooled test reactor, Chalk River, Ontario
NSF	National Science Foundation
NSPP	Nuclear Safety Pilot Plant

NUFREQ	Code that analyzes BWR stability
OBE	Operating basis earthquake
OECD/CSNI	Organization for Economic Cooperation and Development/ Committee for the Safety of Nuclear Installations
OMCOST	Code that estimates capital and nonfuel operating costs of nuclear power plants
ORECA	HTGR system transient analysis code
ORNL	Oak Ridge National Laboratory
ORTAP	HTGR system transient analysis code
OT	Operational transient
PBF	Power Burst Facility
PCI	Pellet/cladding interaction
PCM	Power cooling mismatch
PCRV	Prestressed concrete reactor vessel
PHENIX	French fast-breeder demonstration plant
PIE	Postirradiation examination
PNL	Pacific Northwest Laboratory
PORV	Power-operated relief valve
PPG	Policy and planning guidance
PRA	Probabilistic risk assessment
PSAR	Preliminary safety analysis report
PSI	Primary system integrity
PTS	Pressurized thermal shock
PTSE	Pressurized Thermal Shock Experimental (facility)
PVRC	Pressure Vessel Research Committee

QUICK	Detailed containment aerosol behavior model
RA	Reliability assurance
RCS	Reactor coolant system
RELAP	Detailed model for thermal-hydraulic behavior in reactor coolant system during transient and loss-of-coolant accidents
RFI	Radio frequency interference
RFT	Runaway feedwater transient
RIA	Reactivity insertion accident
RIL	Research Information Letter
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Applications Program
SAFT-UT	Synthetic aperture focusing technique for ultrasonic testing
SASA	Severe accident sequence analysis
SASCHA	German facility where fission product release experiments are conducted
SBLOCA	Small-break loss-of-coolant accident
SCALE	Standardized computer analyses for licensing evaluation
SCDAP	Severe core damage analysis package
SCTF	Slab Core Test Facility
SEISIM	Code that estimates probability of failure of structures and components and seismic risk
SFD	Severe fuel damage
SGTR	Steam generator tube rupture
SIMMER	Code that analyzes course of core-melt accidents in LMFBRs
SLED	State-level electricity demand (model)

SMACS	Code that estimates seismic responses and uncertainties of structures and components
SNM	Special nuclear material
SRP	Standard review plan
SPRG	Senior Research Review Group
SRV	Safety relief valve
SSC	Super-System Code that analyzes system transients in LMFBRs
SSE	Safe shutdown earthquake
SSI	Soil-structure interaction
SSMRP	Seismic Safety Margin Research Program
SSTF	Steam Sector Test Facility
START	Module of TRAP-MELT code that models fission product release from fuel
THE	Thermal-Hydraulic Experiment
THTF	Thermal-Hydraulic Test Facility
TLTA	Two-Loop Test Apparatus
TPFL	Two-Phase Flow Loop
TRAC	Code that models core reflood and quenching
TRAC-COBRA	Code that analyzes PWRs with upper-head injection
TRAP-MELT	Code that analyzes fission product behavior within LWR primary system under accident conditions up to and including fuel meltdown
UHI	Upper-head injection
U.K.	United Kingdom
UMTRAP	Uranium Mill Tailing Remedial Action Plan
UPTF	Upper Plenum Test Facility
UR	Uranium recovery

USGS	United States Geological Survey
USI	Unresolved safety issue
USSP	United States Standard Problems
VGES	Variable Geometry Experimental System
VGM	Vibratory ground motion
WECHSL	German melt/concrete interaction code
WIPS	Structural finite element code that evaluates pipe whip and impact
ZONE	Mechanistic aerosol behavior code with multiple region capability

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The Long-Range Research Plan (LRRP) was prepared by the Office of Nuclear Regulatory Research (RES) to assist the NRC in coordinating its long-range research planning with the short-range budget cycles. The LRRP lays out programmatic approaches for research to help resolve regulatory issues. This year's plan reflects the new organization resulting from the consolidation of RES and the Office of Standards Development. The plan will be updated annually.

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