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UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Information)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: "NUCLEAR PLANT SEVERE ACCIDENT RESEARCH PLAN," NUREG-0900 (DRAFT)

Purpose: Information Paper - Transmittal of Review Draft

Issue: What guidance should be given the staff regarding technical support of formulation of policy on Severe Accidents.

Discussion: The Commission directed on January 29, 1982 that:

"A paper containing plans for producing research information needed to confirm regulatory decisions in the severe accident area, including methodology for comparing the cost of proposed new requirements with their risk reduction, and generalized reduction in the uncertainty of PRA, will be provided to the Commission by February 25, 1982."

Enclosure A to this paper is the plan as requested by the Commission. NRR has performed a preliminary review; its comments are provided in Enclosure B. The attached comments relate to an earlier draft provided to NRR for comments; a number of their comments have been accommodated in the attached revised draft. Enclosure C provides a summary of RES responses to NRR's comments on this earlier draft. There are still some outstanding issues which are yet to be resolved. Key among these is NRR's strong reservation concerning the cost-benefit of Program Element 4 "Behavior of Damaged Fuel." Discussions are continuing. NRR's comments highlight the importance of providing significant information in a two year time period. The revised draft is set up to illustrate what information will be available in two years and what results will require four years, at which time all of the work will be completed, except for any residual confirmatory research. This draft of the plan, along with NRR comments, is being sent to the ACRS concurrently. RES will be meeting with the ACRS Subcommittee on Severe Accidents in a few weeks to discuss their comments on the plan. After ACRS review, and after Commission review and instructions, we will issue a final version of the plan. XA

Contact:
John Larkins, DAE
42-74266, or
Charles Kelber, DAE
42-74442

8206100452

The plan describes the coordinated research programs needed to develop a sound technical basis for Nuclear Regulatory Commission decisions concerning the ability of existing or planned nuclear power reactors to cope with severe accidents, i.e., those which involve damaged or melted fuel. To ensure such a technical basis for these regulatory decisions, two categories of information will be developed: one, a risk analysis process to assess costs and benefits in terms of incremental risk reduction and, two, data related to the behavior of nuclear power plants under a range of severe accident conditions, so that the risk analysis process can be applied knowledgeably.

The goal of the plan is to produce in the first two years:

1. An improved methodology for probabilistic risk analysis, plus a significant extension of the data base for severe accident assessment.

2. Data for a better estimate of the radiological source term used to assess accident consequences.

3. A technical basis for regulatory decisions to add or modify principal design features and operating guides and procedures of existing plants with respect to their ability to prevent and mitigate severe accidents.

Following these short term goals, a more focused program is planned to complete development of the data base, further improve PRA methodology and its applications, and to confirm and render more precise the bases for regulatory decisions and guidance. This continuation is projected to take an additional two years, for most efforts.

The plan provides for regulatory analysis to codify the products of research in such regulatory devices as rules, regulatory guides, standards, or revised standard review plans.

The plan provides for interim evaluations before most of the work is completed, since many of the tasks take four years to complete. In particular, the plan incorporates the Commission's directive on the source term:

"A revised staff estimate of the accident source term, which affects siting, emergency planning and severe accident PRA will be completed in 18 months to 2 years."

The report is organized as follows: Chapter 1 describes the objectives of the work, Chapter 2 discusses the information needs and regulatory issues addressed by the plan while Chapter 3 describes the state of the art. Chapter 4 presents a brief discussion of how the detailed elements of the program are linked and estimates the schedule for production of key results, both interim and final. Chapter 5 describes each of the program elements in detail, and Chapter 6 summarizes the advantages of the approach, as well as some possible pitfalls.

William J. Dircks
Executive Director for Operations

Enclosures: — IN BP
A. Draft Nuclear Plant Severe Accident Research Plan
B. NRR Comments
C. Summary of RES responses to NRR comments on earlier draft

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NUREG-0900
(DRAFT) REV. 2

NUCLEAR PLANT SEVERE ACCIDENT RESEARCH PLAN

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1. INTRODUCTION

1.1 Objectives

The NRC decided, in view of the TMI-2 event, to reflect on the ability of LWRs to cope with degraded cores. This reflection was to include the three-pronged approach of prevention, accident management, and mitigation. Many requirements for LWRs have spawned from the TMI-2 legacy; most have dealt with prevention of severe accidents. A long-term objective, as embodied in Item II.B of the NRC's TMI Action Plan (NUREG-0660) was to look systematically at LWRs and to determine on a more rigorous basis whether further upgrading was needed. This report describes the fundamentals of that investigation.

To ensure a sound technical basis for these regulatory decisions, two categories of information must be developed: (1) a base of data related to the behavior of nuclear plants under a range of severe accident conditions and (2) a risk benefit analysis process to assess benefits in terms of incremental risk reduction (from proposed changes) and the accompanying costs.

These categories of information will be used to provide:

1. Technical bases for more precise appraisal of specific design and operational refinements to permit cost effective risk reduction changes.

2. More accurate estimates of the potential radiological release ("source term") which could accompany severe accidents.

3. More accurate probabilistic risk assessment methods for use in regulatory decisionmaking.

4. Confirmation of the level of safety of plants.

This information will be applied, together with information from the Atomic Industrial Forum's IDCOR project, and related programs of the Electric Power Research Institute to achieve these goals:

In two years:

1. Provide an improved methodology for Probabilistic Risk Analysis of severe accidents, and a significant extension of the data base used by that methodology.

2. Provide the data needed to make a better estimate of the radiological source term used to assess accident consequences.

3. Provide a technical basis for regulatory decisions to add or modify severe accident-related principal design features and operating guidelines and procedures of existing plants. For example, during this two-year period the program will apply the information generated, using probabilistic risk analysis as a tool, to such decisions as:

    a. Should there be hydrogen control and mitigation in all types of containments?

    b. What ways can value impact assessment be used to decide whether or not add on engineered safety features for prevention or mitigation of severe accident consequences are needed?

    c. An improved basis for procedures to enhance the capability of containments to withstand the loads from core melt accidents. We expect the basis to be most satisfactory for large dry containments, but all containment types will be addressed.

The development of guidelines for operator procedures will require participation by user groups including the reactor vendors, owners' groups, Institute for Nuclear Power Operation, the Nuclear Safety Analysis Center, and, of course, the Office of Nuclear Reactor Regulation.

Following the two year program work will be continuing to complete development of the data base, further improve the probabilistic risk analysis methodology and to render more precise the bases for regulatory decisions.

1.2 Relationship to Other Work

This plan depends upon a wide range of other research work, including efforts that do not address severe accidents directly, but are related. In addition, closely related work, as has been mentioned, is being sponsored by industrial groups. With respect to the first category of work, prominent examples include:

a. Probabilistic risk analyses, including the revision of WASH 1400, the RSSMAP and IREP studies, and the detailed studies of the components of the event and fault trees that appear in the sequences that dominate risk. This work has developed the insight that enables one to set priorities and order the work with some confidence that the issues posed are critical to reactor safety evaluation. Moreover, the studies have developed a good understanding of how the risk assessment methodology is best improved, and then applied to detailed studies.

b. The TRAC and RELAP code development efforts, as confirmed by LOFT, Semi-scale and PBF test programs, established the basic means for predicting how a plant undergoing a potentially severe accident can be brought to a safe condition.

With respect to industry sponsored work, the IDCOR program is an effort devoted to correlating current knowledge, with some methods development, to help the vendors and utilities arrive at a best estimate of the most effective ways to cope with severe accidents, especially in existing plants. An important aspect of the IDCOR program is the independent development of improved PRA methodology for analyzing physical processes; this should furnish a valuable element of quality assurance to the NRC's efforts in this area.

It is doubtful the short-term objectives of this plan can be met without the IDCOR compilation. Equally significant are EPRI programs in such areas as hydrogen control and in radiological source term determination. These programs are being coordinated with NRC work to avoid duplication of effort; this projected schedule of NRC work assumes that EPRI and IDCOR will continue their efforts.

As this discussion reveals, a depiction of all the information inputs would be very complex, and is not attempted. Instead, to aid understanding of the management considerations involved with the plan, an overview of the information flow from the various elements of this plan only has been prepared and is presented in Chapter 4.

Although the dividing line between prevention and mitigation of accidents is not clearly defined, it is true that this plan places substantial emphasis in terms of money on resarch supporting accident mitigation systems and procedures. This is not surprising, as the test facilities for Behavior of Damaged Fuel are quite expensive. However, in other programs, such as IREP and NREP, the Commission sponsors major efforts focused primarily on accident probabilities; from this, information on prevention can be derived. These programs, which are carried out by activities outside the scope of this plan, together with the programs described herein, constitute an appropriate balance of effort in severe accident research.

1.3 Background and Context

1.3.1 Issues

Two major issues have been raised by the accident at Three Mile Island Unit Two. One is the need to bring to bear the techniques of modern technology on the requirements for quality assurance and control in the design and construction of plants, and on their reliable operation.

The second major issue is the extension of regulation to encompass the low likelihood, high consequence accidents that dominate the risk. The early recognition that a hydrogen explosion occurred in TMI 2 gave impetus to a program on hydrogen control and initial results are being factored into current licensing practices, and backfits are being considered in accordance with a Commission Rule.

The dual use of PRA to pinpoint problems affecting reliability as well as to determine risk dominant sequences has afforded that discipline a central role. It has become clear, however, that deficiencies in the data base and the present state of development of PRA methodology carry large uncertainties such that, for example, in coping with hydrogen fires, PRA sometimes does not offer a clear choice among alternative courses of action. The need for improvements in methodology and physical research to narrow these uncertainties is apparent.

Moreover, the problem is only slowly becoming partitioned. The concerns following TMI 2 focused attention on Zion and Indian Point (NUREG/CR-1409, 1410, and 1411). The original concept was that simple solutions to problems of coping with severe accidents, which were judged by PRA to be cost effective, would be clearly good enough that the issues could be quickly resolved. Instead, two different types of findings emerged.

One was that the current state of technology did not allow a value impact judgment to be made. Indeed, it is not even known if the fuel will melt in some putative risk dominant sequences.

At the same time, it was found that the plant containments being analyzed were strong enough that even in the presence of massive uncertainties, the risk appears to be very low, much lower than estimated in earlier assessments. This finding has been formalized in NUREG 0850, "Preliminary Assessments of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," and, by deterministic studies reported in NUREG/CR 1988 (SANDIA 0503), "Analysis of a Hypothetical Core Meltdown Accident Initiated by Loss of Offsite Power for the Zion 1 Pressurized Water Reactor." Thus, the assessment of the problem of effect of gaps in the data base on the risk from severe accidents is beclouded by the fact that the plants most thoroughly studied to date have such strong containments that their risk appears to be on the low side, and the effects of data gaps are thus low in terms of comparative risk. If the strength of the containments were as little as 15% lower, howevre, the uncertainty bands related to data bands would result in significantly larger uncertainties in risk analysis, because early failures would then be predicted as a possibility in a number of sequences.

1.3.2 Technical Problems

There are significant technical problems in meeting the needs posed by the issue of plant operating reliability. Many appear to be focused on the topic of human factors, but they certainly include a host of related efforts. There may be some doubts about the best mode of solution, such as substituting computer errors for unknown human error by utilizing computerized closed loop control appears to be one feasible alternative, since it is used at some plants in France (in particular PHENIX). There appears to be general agreement about the need to attack the problem by developing the needed human factors technology. It is noteworthy that the massive research program conducted by the NRC on plant response to LOCAs and Transients has sufficiently defined the problems facing the operator and operational options during these accidents to facilitate development of improved engineered safety features and operator guidelines. This is being done, for example, via the Severe Accident Sequence Analysis program (cf. element 5.2) in connection with studies of Anticipated Transient Operator Guidelines (ATOG) by NRR. The benefits from this aspect of the LWR safety research program are probably as great as the direct benefit to the licensing review.

After the staff analyses performed for the Kemeny and Rogovin Commissions, there arose a growing understanding of the substantial gaps that exist in the data and methodology to treat severe accidents. It is not known in any significant detail when and how the fuel will melt under stress of an accident, or if it will melt as fast as simple energy conservation calculations would indicate. It is similarly uncertain what fission products will be released and how many transported to the containment. We remain uncertain as to how much hydrogen is generated when, and how it might burn.

In the case of strong containments, these gaps in present phenomenological information may be tolerated to a much greater degree.

In view of the importance of Probabilistic Risk assessment to the regulatory process, and to the problem of arriving at accurate value impact assessments, it is essential that risk be correctly assessed. Sensitivity analysis and the phenomenological component of the research program contribute to this end, as do the planned developments of improvements in risk methodology and risk reassessments.

Correct identification of and subsequent reduction in the uncertainties of predicted risk is a complex process.

One major issue is completeness. The extent to which important safety related phenomena in both system and human response are overlooked by current methods is difficult to determine. Phenomenological research can only indirectly contribute to the reduction of the incompleteness component of uncertainty. But uncertainty can also be caused by a lack of insight into the physical processes governing the course of an accident. Errors can be introduced into predicted risk by unrealistic treatment of the accident phenomenology. These types of uncertainty will be addressed through the program of phenomenological research.

Actual risk is governed by:

a. the probability of an accident involving core damage.

b. the effectiveness of the containment

c. the extent of operator intervention

d. the magnitude and form of the radiological source term and

e. the site characteristics (including meteorology) and

f. the effectiveness by which emergency procedures reduce exposure to the population.

The uncertainties in the overall risk assessments arise from the uncertainties in the individual risk factors. Major uncertainties in factors such as containment effectiveness and radiological release are not so much related to reliability considerations as they are to an incomplete understanding of the physical processes involved. For this reason, efforts have been made in risk analysis to produce estimates that are conservative when faced with significant unknowns; that is, they usually lead to higher estimates of risk than we believe to be the case. Such conservatism can be appropriate for many regulatory purposes but it tends to defeat the attempt, for example, to make a valid analysis of cost effectiveness of risk reducing features.

1.3.3 Priorities and Costs

The relative benefits and projected costs of reducing uncertainties in severe accident phenomenology are a major factor in setting priorities and schedules for research. The relative benefits are adjudged by the three major areas of regulatory concern: (1) Safe plant design and construction; (2) Safe operation and maintenance, and, in the event that an accident does occur, (3) Adequate accident and emergency response procedures.

The key role of Probabilistic Risk Assessment in assessing benefits leads to the assignment of high priority to improvements in PRA methodology to enable better application of knowledge about severe accidents to accident likelihood and risk re-evaluation as well as to value impact assessment. Short-term improvements in basic PRA methodology are being used pending the introduction of improved codes by both the NRC and IDCOR. Since most of the continuing applications of PRA will involve iteration between assessment and improved deterministic evaluations, continued improvements in methodology are needed to maintain the program in focus; this means that PRA assessments will be used to determine changes in the emphasis of our phenomenological research, and the results of this will be fed back into PRA to produce revised risk assessments.

The technical difficulties in much of the phenomenological research pose some difficulties in respect to cost and schedule. The program presented in this plan is based on current knowledge and analysis of uncertainties in risk assessments. We expect that it will be significantly revised in about two years, after initial results are in hand.

With respect to the first of these areas of concern severe accident research will provide information on the physical phenomena that engineered safety features and mitigating system equipment must address during a severe accident. This is the particular focus of the work which addresses how a molten core interacts with the primary vessel and subsequently with the containment basemat, and what ensues from such interactions.

With respect to the second area-improved safety of operation- there are two important factors: (1) Reduce challenges to the safety systems and (2) Mitigate those accidents that do occur. The plant operator needs improved early information of abnormal conditions and better procedures to inhibit the progression of significant core damage or to reduce the release of radioactive meaterials through optimum use of the plant's accident mitigation features. These actions are termed accident management. To establish a basis for sound accident management it is necessary to study how damage progresses when cooling capability is severely degraded, how cooling should be restored as systems again become available and still not overload the system or cause further damage, and how releases to the containment correlate with the progression of fuel damage. This is a significant product of the severe fuel damage program and related efforts, including fission product release and transport, and hydrogen control.

Finally, with respect to accident and emergency response procedures, research leading to an accurate definition of radiological release and consequences, as well as research leading to an improved understanding of accident progression contributes directly to the basis for sound procedures.

The projected costs for the various efforts described in this plan are about $200 million for a four year program. Table 4-2 in Chapter 4 provides a preliminary breakdown on cost for each element for the four year period. Almost half of the total is devoted to understanding the progression of damage when cooling capability is severely degraded. These costs arise from the intrinsic difficulty of research on damaged fuel phenomena; however, the benefits are, high, since they are spread across all three of the areas of concern described above. For example, containment effectiveness is considerably enhanced if a severe accident can be terminated and the damaged, but coolable fuel remains in the primary system as at Three Mile Island, Unit Two. We have to understand core damage progression, however, if we are to determine the extent to which the operator can rely on such enhanced effectiveness.

Similarly, the capacity required of hydrogen control systems and the effective use of sprays to mitigate the release of certain fission products requires more basic knowledge of the way in which fuel damage proceeds as cooling fails. To be effective in promoting safety, value impact analysis requires a realistic basis for assessing the adequacy of equipment to do the job. The fundamental source of the load which mitigative equipment must handle is the severly damaged core.

Finally, proper emergency response arrangements require an appraisal of the potential radiological release. Since the source of the release is the damaged fuel, it is clear that understanding of the progression of fuel damage and its consequences is essential. This knowledge will also lead to an understanding of the time available for various response measures.

In this plan largely analytical elements tend to cost somewhat less than the experimental programs because of the heavy logistical demands of physical research. Timing is important because public safety is better served by early return to regulatory stability.

The work of IDCOR directed at accident prevention and mitigation based on current knowledge, and the work by EPRI on hydrogen control and on the radiological source term will enhance the value of the early results obtained from this program. Our goal is to make early decisions for those systems least sensitive to remaining uncertainties, while developing the data needed to resolve more sensitive issues. At this time those plants having large dry containments as typified by the Zion and Indian Point Plants appear to be least sensitive to current phenomenological uncertainties.

Cost projections by program element are given in Chapter Four.

1.3.4 Research Programs Quality Assurance and Control

Normal technical review processes carried out by the principal investigators and their peers represent the first level of quality assurance. Control and coordination will be ensured by convening research review groups with consultants from a wide range of associated activities, including IDCOR and EPRI, as appropriate, to determine the completeness and adequacy of the work leading to a milestone. Finally, a senior research review group, representing upper level management, with consultants from a similar level in other organizations, will be formed to review over-all progress and direction periodically, as well as the impact of technical findings on regulatory actions.

1.4 Description of Contents

This report is organized as follows:

Chapter 2 discusses the information needs and regulatory issues addressed by the plan while Chapter 3 describes the state of the art. Chapter 4 presents a brief discussion of how the detailed elements of the program are linked and estimates the schedule for production of key results, both interim and final. Chapter 5 describes each of the program elements in detail, and Chapter 6 summarizes the advantages of the approach, as well as some possible pitfalls.

In the chapters to follow we discuss a number of computer codes in varying stages of development. The codes discussed fall into two classes: those used for carrying out probabilistic risk assessment and those used for deterministic studies to develop technical specifications for regulatory guides and rules. The latter class of codes is composed of computer programs that describe operating phenomena in great detail and that are subject to validation against experimental data. The first class is composed of codes that seek to represent lumped effects or consequences of a series of events in order to understand the progression of events, given an assumed set of faults. These codes, which are used for probabilistic risk assessment, tend to be fast running, obtaining their lumped representation of effects from the more detailed, deterministic codes or from the associated experiments.

A brief glossary of codes by class is furnished in Table 1-1.

Table 1-1

Deterministic Codes	Probabilistic Analysis Codes
SCDAP - Severe Core Damage Analysis	MARCH - Models of the thermal-hydraulics of melt-down event sequences
HECTR - Hydrogen Combustion	CORRAL - Models of fission product transport in event sequences in containment
CORCON - Fuel/Concrete Interaction	CRAC - Ex-plant consequences code
TRAP MELT - Fission product release and transport in primary	MATADOR - Improved CORRAL code
CONTAIN - Detailed prediction of of containment loadings	MELCOR - Long-term replacement to MARCH, MATADOR, CRAC

There is a detailed discussion of the schedules, costs, and interactions of these program elements in Chapter Four. In summary, we expect to maintain program elements at a pace providing timely exchange of information among the elements, with the goal that interim results will be produced to answer immediate needs for information in two years. Ongoing confirmatory results and clarification of the more complex issues needed for regulatory insight will permit the program to be more tightly focused thereafter, to the end that at the conclusion of this Severe Accident Research Plan in 1986, the NRC will be on a sound basis for regulation of all operating and contemplated nuclear power plants.

2. INFORMATION NEEDS AND REGULATORY ISSUES

Resolution of the generic and specific regulatory issues will require a substantial body of organized information. We examine the information needs related to the issues and report these findings in this chapter. We then examine the state of the art in each of the areas to ascertain what we now know. The results are reported in Chapter 3. The difference between knowledge needed and knowledge on hand represents the body of technical material to be developed by the program; details of the program are described in later chapters. The budget decision units and subelements involved are: (1) Probabilistic Risk Assessment (PRA); (2) Accident Management; (3) Behavior of Damaged Fuel; (4) Fission Product Release and Transport; (5) Fuel-Melt Interaction; and (6) Accident Mitigation.

Three bodies of organized information are projected as output of the program:

1. Data for guidelines for refinements to system design, operating procedures, and instruments;
2. Verified methodology for accident load phenomena and system responses;
3. Information for decisions on potential risk reduction add-ons and refinements.

The plan provides for transformation of these products into regulatory end products (i.e., regulations, guides, and revisions to the standard review plan).

Most current questions about severe accidents result from consideration of the accident at Three Mile Island Unit 2 (TMI-2), with some additional questions arising from other accidents with potentially serious consequences such as the Brown's Ferry fire. The need to focus on severe accidents was documented in the Reactor Safety Study (WASH-1400), but detailed technical questions were not adequately framed until the accidents provided numerous focal points of inquiry.

The accident at TMI-2 on March 28, 1979, was a severe reactor accident. Although the accident produced virtually no offsite radiological consequences, it did great damage to the reactor and raised serious questions about the adequacy of the regulation of nuclear power plants in the United States. In the process of regulation, practice had been to test the adequacy of nuclear plant design against a set of design basis accidents that were believed to constitute a sufficient envelope of credible scenarios. System reliability was "assured" of meeting regulatory requirements by using a postulated single failure criterion in the safety analysis, quality assurance procedures, and inservice inspection and testing. The acceptability of reactor sites was tested by a hypothetical accident dose calculation that combined the most serious design basis accident with postulated nuclear core damage and a radioactivity release level believed to represent severe accident phenomena to an adequate degree.

The TMI-2 accident challenged the validity of many of these practices. The events of the accident did not fit the envelope of design basis accidents (DBAs). Events did not follow the simple binary logic postulated in the DBA in which things either worked or they failed. At TMI-2, core cooling was not completely lost but severely degraded. The core was badly damaged, but there was no significant core melting. Large quantities of hydrogen were formed, released, and burned during the prolonged core damage sequence, rather than the small amount prescribed in §50.44 of 10 CFR Part 50 for design basis accident analysis. Large quantities of radioactive fission products were found in the coolant water, greatly restricting the ability to circulate cooling water for safe shutdown. The released fission products so pervaded the plant that personnel access was made very difficult. The operating crew committed repeated and persistent errors, failing to diagnose the accident causes. In sum, a host of questions were raised about the adequacy of plant design and operation and of NRC regulations for dealing with severe accidents.

In particular, three key questions representative of the major concerns, were raised in the report of the President's Commission (the Kemeny report):

1. Page 15 - How can we identify and analyze the possible consequences of accidents leading to severe core damage? "Such knowledge is essential for coping with the results of future accidents."
2. Page 15 - How can we prevent such accidents and minimize the potential impacts on the public health and safety?
3. Page 72 - What are the consequences and probabilities of such accidents, including the consequences of meltdown?

Our aim is to see what information is needed to answer these questions. The information we seek is categorized by NRC budget decision units and sub-elements because of their relationship to the Long-Range Research Plan, but this categorization is otherwise arbitrary.

2.1 Probabilistic Risk Assessment

The TMI-2 accident dramatized the inadequacies of traditional regulatory treatment of severe accidents. The elements of the TMI-2 accident scenario seemed to affirm the principal factors of accident risk as described in the Reactor Safety Study (WASH-1400), which used Probabilistic Risk Assessment (PRA) to obtain as realistic as possible a description of severe accident behavior and risks. The Reactor Safety Study utilized the risk assessment of only two plants as temporary surrogates for the first hundred. More plant specific risk assessments are needed, and are being done, to develop a technical basis for regulatory decisions regarding severe accidents.

It was realized that the two Reactor Safety Study plants are not apt surrogates for the variety of plant system and containment designs that exist. We need more representative PRA models of each basic type of plant. If we are to use these models for regulatory decisions regarding severe accidents, we must assess the level of severe accident risk as well as the relative risk reduction benefits of changes in plant design or operation. The many questions raised are:

1. What are the probabilities of specific accident sequences?

2. What are the consequences of these individual accident sequences, and how do they contribute to overall risk?

3. What are the risk reduction effects of changes in plant design or operation?

4. What are the dollar savings possible from averted losses?

5. What are the costs of changes to reduce risk or avert loss?

6. How do current regulatory practices aid in allocating risk, and can improvements in regulatory practices also improve the control of risk allocation?

The physical data necessary to apply to improved PRA techniques will be acquired by a program of physical research comprised of five technical elements. These elements correspond in general to the areas of difficulty encountered in casework such as the Zion-Indian Point Study (NUREG/CR-1409, -1410, -1411) and later reviews. The Zion-Indian Point Study was an initial attempt to coordinate the use of PRA and best-estimate physical modeling to determine the potential value of methods for reducing residual risk from a specific set of nuclear power plants.

The five remaining elements that have been identified and now appear as budget subelements are: Severe Accident Sequence Analysis (Accident Management Guidelines); Behavior of Damaged Fuel; Fission Product Release and Transport; Fuel-Melt Interactions (Containment Failure Processes), and Accident Mitigation research. The program of physical research is narrowly defined to produce data for PRA and to be capable of defining objectives more precisely as better PRA results become available. Therefore, it is important that intermediate products become available to make possible better use of PRA before the program is largely completed. The plan of physical research is designed to allow this, with major intermediate results in the second and third years.

We next address the information needs and regulatory issues associated with the five elements that make up the physical research program.

2.2 Severe Accident Sequence Analysis

The Research budget subelement addressing this particular technical area is called Severe Accident Sequence Analysis (SASA). As pointed out above, the examination of the accident at TMI-2 raised a host of questions about plant operation, among other things, with respect to the tactics for dealing with severe accidents. Actually, the potential for improving accident management techniques to reduce risk was first recognized after studying operator actions during the Brown's Ferry fire. Subsequent events at TMI-2 reinforced the idea that systematic studies of accident management will yield useful guidelines for emergency procedures under multiple failure conditions. The regulatory issues raised by these considerations are:

1. Should guidelines be established for operator response during severe accidents?

2. Should there be additional instrumentation and information on requirements to assist the management of severe accidents?

3. Should the operator be required to take actions to interdict fission product transport and mitigate containment failure during severe accidents?

4. Should the regulations involving emergency response reflect emergency procedure guidelines?

5. Should the design bases for handling major fission product releases, as well as the corresponding equipment qualification standards, be revised?

The SASA program has developed a detailed program plan that is condensed within this report. This program will complete major milestones by the end of FY 1983 with respect to management of accidents to reduce the likelihood of progression of significant fuel failure. Also included in the plan are accident studies extending beyond the point of significant fuel failure. These studies are scheduled for completion at the time when more comprehensive data about the behavior of cores with severely degraded cooling have been acquired in the Severe Fuel Damage research program. In general, accident management is an attempt to prevent significant core damage. Existing procedures fill this function under the single failure criterion, and SASA is attempting to extend the process to multiple failures; management guidelines are required for the optimum use of accident mitigation features in a presumed case of large-scale core melt.

2.3 Severe Fuel Damage

The Severe Fuel Damage program is designed to deal with difficulties identified in the analyses of the TMI-2 accident. A key issue raised during the considerations of TMI-2 is: How does one deal with a severe accident once it occurs or how does one keep a minor accident from progressing into a major one? It is clear that all early efforts should be focused on early termination to minimize damage and release; this requires, among other things, correctly interpreting data and identifying problems. Crucial to this is avoiding actions which can make the accident worse. For example, it has been argued that regardless of the state of progression of a severe accident, the first act should be rapid reflood of the core as soon as emergency cooling is available. The analysis of TMI-2 suggests that if the internals are very hot and the primary system steam starved, such actions will product steam pressure spikes (from rapid quench) or even steam explosions, which might fail the reactor vessel, leading to full core melt and possibly even the containment. Another concern from such a scenario would be the very large hydrogen production that would arise in the steam rich atmosphere following the sudden reflood. Too-rapid reflood might also fragment the core into debris too fine to remain coolable by the reflooding. Hence, proper operator action at this time could mean the difference between catastrophic release of radioactivity to the environment caused by early containment failure or safe termination of the accident.

Even supposing emergency reflood is not available, it may be possible to cool a degraded core. To make decisions about how to cool the core one has to know the conditions of coolability. A major problem is lack of knowledge of the likely damage state of the core as the accident progresses. Estimates using the available fast-reactor debris-coolability data and models, unverified for specific LWR accident conditions, indicate that, for damage states less severe than fragemented pellets, an entire core is coolable by simple reflooding at low pressure after the first day, while with fragmented pellets, high pressure or inlet flow are necessary to achieve coolability at this time.

The current state of knowledge in these areas is addressed in Chapter Three; it is clear that a significant extension of existing technology is required to meet information needs regarding the behavior of severely damaged fuel in order to address the following regulatory issues:

1. What are the water inventory, pressure, and inlet flow rate needed to achieve coolability of a severely damaged core, as a function of its state of damage? How do these parameters compare with the capabilities in place at plants?

2. What guidelines are there to indicate the optimum method of restoring cooling so as to minimize the potential hazard to public health and safety?

3. What is the rate at which the fission products and hydrogen are being produced and transported to the containment?

It is planned that major portions of data from this program will be available in FY 1983, with still more available in FY 1984. The Severe Fuel Damage program plan is provided in more detail in Section 5.4.

2.4 Fission Product Release and Transport

An observation growing out of the TMI-2 investigations and subsequent studies of better estimates of accident consequences is that the radiological source term* generated by nuclear plant severe accidents may, in some cases, be very conservatively characterized by the assumptions used in guides or in WASH-1400. Since both siting rules and risk evaluations depend on the technical details of the radiological source term, a research program to trace the formation of the components of the radiological source term and their transport within the primary system and containment has been established. The Fission Product Release and Transport program (FPR&T) will meet its first major milestones in FY 1983.

The radiological source term issues are:

1. What is the composition and magnitude of the radiological source term corresponding to each of several dominant accident sequences,

2. What design features significantly affect the composition and magnitude of the radiological source term, and

*By "source term," we mean the radioactive material in the nuclear power plant that can leak out or can be released by containment failure and thus pose a hazard to the public. Although the actual composition of the source term in an accident will depend on details of the accident, it is common practice to correlate a hypothetical composition with a given accident or set of accidents. Such a hypothetical composition is then called a "source term." The source terms used currently attempt to model, in crude ways, processes that transport the fission products from the fuel to the containments and processes that tend to remove the fission products from the containments.

3. To what extent should these details of the source term components be reflected in equipment qualification, plant design (shielding), siting, and emergency procedure regulations?

2.5 Fuel-Melt Interaction (Containment Failure Process)

In risk analysis, another major concern is that radioactivity, as characterized by the radiological source term, might be released early in an accident sequence as a result of containment failure. The Zion-Indian Point Study encountered major problems in determining the likelihood of early failure and found that the provision of engineered safety features such as vented-filtered containment that might mitigate such failures must depend on the details of how a molten core would attack the vessel and subsequently the containment. Processes that serve to attenuate fission products suspended in the containment are slower to develop; however, at the same time there are processes (e.g., increased containment pressure) that may threaten long-term containment capability. The details of these processes are important to proper classification of the release category for PRA, which is used in siting and consequence considerations.

The regulatory issues that are considered in this context are:

1. Under what circumstances can processes such as hydrogen burning, steam explosions, and basemat attack lead to containment failure, and

2. Are there modes of containment failure that affect the magnitude of the release of radioactive material?

The Fuel-Melt Interaction research portion of this program is designed to develop relevant data to resolve these issues. Major milestones are planned to be met in FY 1983 and 1984.

2.6 Accident Mitigation

As expressed in the Commission's Construction Permit/Manufacturing License (CP/ML) rule, there is a need to anticipate features to mitigate the results of severe accidents that threaten the containment.

The Accident Mitigation element of this program supports the physical research that develops technical feasibility and engineering design criteria appropriate for such engineered plant features. An early result (in late FY 82) of the PRA effort will be a tentative ranking of such features to identify the worthwhile features and to thereby help organize and give priority to the study, as well as limit the scope of the work. It is expected that major milestones will be met by FY 1983 with respect to important classes of features such as hydrogen fire suppression, and other milestones will be met by FY 1984.

Regulatory issues addressed by this element are:

1. What are the relative costs and benefits of such features, and

2. What are the design criteria for features that are assessed to have the most promising value/impact in preventing or mitigating containment failure?

3. STATE OF THE ART

This chapter summarizes the current capabilities for providing the information needed to resolve the issues listed in Chapter 2. To the maximum extent possible, recent reports that summarize the technology involved are used. In most cases, the reports cited have received extensive peer review.

3.1 Probabilistic Risk Assessment

By the fall of 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 condensor	NRC	RSSMAP**
Sequoyah	Westinghouse	Ice condensor	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	Cons. Power	
Limerick *	GE	Mark II	Phil. E.	
Zion*	Westinghouse	Large dry	Com. Ed.	ZPSS
Indian Point*	Westinghouse	Large dry	PASNY-Con. Ed.	IPPSS

*Published as of this writing.
**NUREG/CR-1659

There are several ways in which these studies have been or will be checked for completeness and accuracy. These are (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) by comparison among 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 the 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 international 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 and from changes in siting or emergency response practices are fairly reliable; they can be made far more reliable by carefully examining the approximations and assumptions in the PRA model to which such evaluations are sensitive and selectively upgrading these as necessary. Likewise, evaluations of the importance of contributors to risk, such as classes of initiating events, accident sequences, system reliability, etc., are sometimes reliable and can be made more so by careful examination.

The picture of PRA usefulness emerging today is similar to that of econometric models purporting to predict the evolution of the national economy. PRAs are built upon a coherent, causal framework on which is stretched a fabric of sometimes simplistic assumptions made in order to render the vast complexity of accident susceptibility amenable to analysis. 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 setting severe accident regulatory policy. Rather, we anticipate that PRA models will be used to help establish the technical foundations for severe accident regulatory policy in a number of other ways that are less sensitive to the completeness and precision of the models:

1. Value-impact analysis of alternative designs, procedures, siting policy, and emergency planning options.

   Since PRAs are more accurate in calculations of risk differences than of absolute risk, cost-benefit assessments are more trustworthy than measures of compliance with risk threshold values intended to delineate acceptable risk.

2. Identification of the risk-limitation effectiveness of regulatory requirements.

   Sensitivity studies and importance-to-risk evaluations share the characteristic reliability of risk-difference calculations.

3. Identification of the attendant risks of altered 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 improved or rebaselined to strengthen the case for and against each regulatory option.

A number of limitations in many or all of the published PRAs that will need to be addressed in these applications have been identified and research projects established to upgrade the reference 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 developed. In addition, more complete event sequence delineation is needed for the variety of possible interfacing systems LOCAs, vessel rupture in conjunction with thermal shock and/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 better 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. Blindspots in testing through which safety system faults could escape detection and repair for long periods of time.

It will not be practical to revise the PRAs to incorporate quantitative, causal models of the likelihood and severity of these failure mechanisms for every sequence or system. Rather, the 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.* Thus, for example, consideration will be given to the potential impact of the probabililities of seismic events and satotage on the overall likelihood of melting and risk. In addition, the capability of possible plant modificaitons to withstand or cope with such events will be considered as part of the risk reduction benefit and cost analyses to be performed.

Such techniques were used in the Reactor Safety Study, WASH-1400, to enable sensitivity studies to be made of ECCS effectiveness in large LOCA events. An artificial event labeled "E" was inserted in the LOCA event trees whose probability was then unknown but which modeled the possibility that ECCS might fail to cool the core even though the ECCS system delivered water in the intended amounts and response times. It turned out that ECCS

*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.

effectiveness for large LOCAs did not matter because the risk was dominated by small LOCAs and transient sequences in any case for the two plants studied. Similar techniques will be used to explore the range of effects of these known limitations in the reference PRAs upon the severe accident regulatory tradeoff evaluations.

4. Phenomenology of severe accident processes.

Most PRAs which include analyses of containment challenge and attendant radiological releases have used rather simplistic models of the phenomenology of containment challenge by damaged or molten cores. PRAs such as those done in the Reactor Safety Study and the Reactor Safety Study Methodology Applications Program are suspected of being nonconservative in, for example, the assumption that core debris and aerosols do not foul active containment heat removal systems. They are also suspected of being conservative in their failure to resolve core damage from core melt outcomes, in their treatment of plateout mechanisms, in the containment failure modes, in the presumed inoperability of sump recirculation following containment failure, etc.

In the earliest round of PRA-based value-impact assessments of alternative designs for core damage or melt mitigation, these simplistic PRA's will be used. These scoping studies will be reported in the summer of 1982. In the next round of studies, improved codes will be used and sensitivity studies performed to evaluate the effects of these known or suspected biases in the phenomenological analyses in PRA. This work will be reported in the summer of 1983. Subsequent refinements of these studies scheduled for 1984 and thereafter will employ a third generation risk assessment code, MELCOR, which is intended to provide a more realistic, state-of-the-art evaluation of the phenomenology of severe accident processes.

3.2 Severe Accident Sequence Analysis (Accident Management Guidelines)

This effort uses a combination of risk assessment methods, best-estimates codes and human factors methods to study the interrelationships between the man and the machine to provide the operator with guidelines for controlling the plant under accident conditions. Accident sequences that contribute significant risk in probabilistic risk assessments are studied in detail using state of the art thermal-hydraulics codes in the Severe Accident Sequence Analysis program. Codes such TRAC and RELAP provide a relatively precise evaluation of transients up to the start of significant core damage. Beyond that, the limitations of the phenomenological data base on fuel damage and relocation in severe accidents make credible modeling extremely difficult.

The MARCH code is intended to model accident progression following the onset of core damage; however, the code has limitations which make its results less than ideal. Details of accident sequences after the core materials penetrate the primary pressure vessel can be treated using the methods described in Section 5.7. Principal codes used are CORCON to describe the fuel-concrete reaction and CONTAIN to predict the loads on the containment. The data base to improve this part of the accident analysis depends on the more basic work described in Sections 5.5, 5.6, 5.8, and 5.9.

Current applications of SASA provide insight into accident management guidelines so long as limitations in modeling fuel damage and relocation are recognized. These studies have analyzed small-break LOCAs, large-break LOCAs, interfacing-systems LOCAs, loss of AC power, and loss-of-feedwater transients. The studies have evaluated numerous accident strategies for each of the sequences.

The studies have also directly supported the resolution of unresolved safety issues (USI) and the evaluation of Abnormal Transients Operator Guidelines (ATOGs).

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

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

3.3 Behavior of Damaged Fuel (Severe Fuel Damage)

An assessment of the very limited current state of knowledge in this area is given in NUREG-0840, "Report of NRC Fuel Testing Task Force." This report received extensive industry and international peer review. A summary of this assessment in NUREG-0840 is given in the following sections.

3.3.1 Damage States

Knowledge of the physical and chemical state of a severely damage core is the major prerequisite for determining the ultimate coolability of the core and for determining the potential radiological source term to the public. Since those scenarios that lead to core melt represent the greatest contributors to risk, a determination of whether or not a damaged core is coolable upon reflood at any time during a severe accident sequence is important in determining the risk to the public. Section 2.6 of the Executive Summary of the Reactor Safety Study (WASH-1400) states that: "The only way that potentially large amounts of radioactivity could be released is by melting the fuel in the reactor core." It goes on to state that: "Thus, for a potential accidental release of radioactivity to the environment to occur, there must be a series of sequential failures that would cause the fuel to overheat and release its radioactivity." The methodology used in WASH-1400 was based on event tree and fault tree analysis that determined the probability of failure of certain systems. After failure, no allowance was given for their ultimate return to service. However, during slow accident sequences, such as that which occurred at TMI-2, complete or partical recovery of such systems is possible by proper operator intervention. Operator action at TMI-2 while contributing to the severity of the accident, did ultimately prevent massive core melt so that were was no appreciable radiation injury to the public. Therefore, when one considers the risk to the public for a given series of equipment failures, one must also consider the effect of the return to service of those systems and the proper mitigating actions of the reactor operators. In order to compute such effects on risk calculations, one must know the chemical/ physical/thermal state of the fuel during the accident sequence.

The current state of knowledge on severely damaged fuel is based on very limited experiments and analyses, which provides a poor basis for understanding and evaluating severe fuel damage and core melt behavior for accident management and risk assessment studies. The only work that was focused on the early stages of severe fuel damage (as opposed to core melt behavior, steam explosions, etc.) was performed at KfK in the Federal Republic of Germany by S. Hagen from 1976 through 1978. These experiments showed clearly that the damage state of electrically heated fuel rod simulators heated in steam to temperatures in excess of 3600°F (2255°K) depends primarily on four major parameters: (1) the final temperature reached, T_{max}; (2) the heating rate, dT/dt; (3) the rate of cooling, dQ/dt; and (4) the pressure difference between the interior of the rod and the reactor coolant, ΔP. Current knowledge can be summarized in terms of these parameters by defining "damage regimes" in terms of T_{max} and expressing the effects of the other three parameters on the phenomena that occur. The following paragraphs discuss each regime in detail by focusing on (a) the physical/chemical phenomena involved and (b) the safety issues to be addressed (if any). Figure 3-1 gives a simplified schematic illustration of the damage regimes discussed. Finally, the current state of knowledge and the information needs on the coolability of severely damaged fuel are discussed at the end of Regime V.

1. <u>Damage Regime I (T_{max} < 1700°F (1200°K); Δ P negative 100-1200 psi; any dT/dt)</u>

a. Physical/Chemical Phenomena - Cladding buckling, collapse, and "wasting" of the fuel stack. These phenomena were studied extensively for the NRC LOFT program and are well correlated with data. Very little additional data are needed and modeling of the effect can proceed with confidence.

b. Safety Issues - Reactor behavior in this regime is covered by current licensing practice for DBA and is not considered to be "severe" fuel damage.

2. Damage Regime II (T_{max} < 2200°F (1475°K): ΔP positive; and dT/dt)

a. Physical/Chemical Phenomena - Cladding ballooning and burst. This phenomenon has undergone extensive study in the last 10 years. Plentiful data are available and preliminary models have been developed. Final resolution of the effect on core coolability awaits completion of the NRU ballooning experiments in FY 1982 and future tests at KfK in the FRG.

b. Safety Issues - The ballooning process may affect the coolability of the core because of the partial closure of coolant channels. If such blockage is near 100 percent, partial localized melting may occur. Current evidence indicates that the latter possibility is very unlikely. In any case, the programs mentioned will fully investigate the possibility. As is the case for Regime I, damage in this area is covered under current regulatory practice and is not considered to be severe fuel damage.

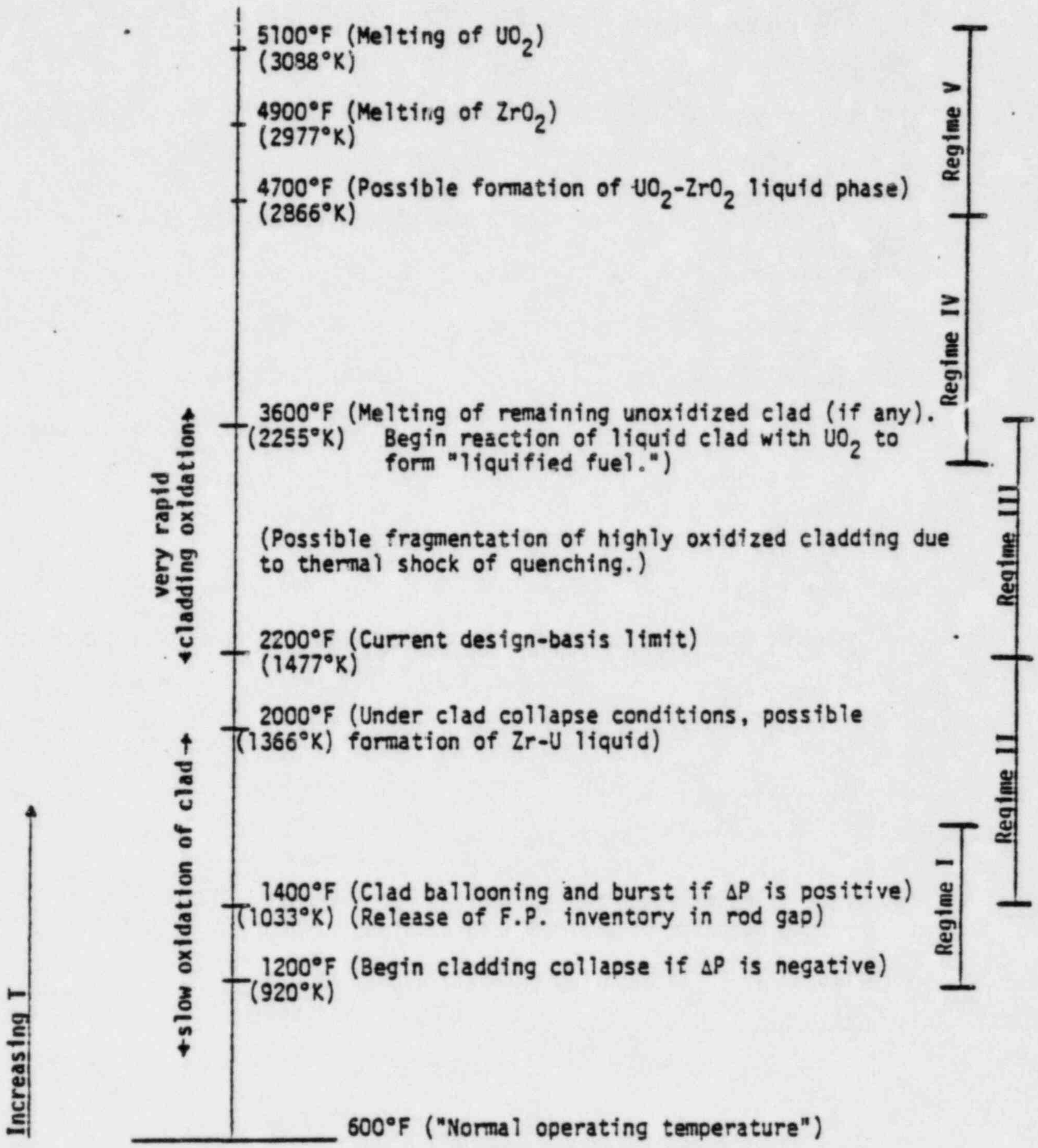


FIGURE 3-1

Schematic (not to scale)

3. Damage Regime III (T_{max} <3400°F (2140°K): any ΔP; any dT/dt)

a. Physical/Chemical Phenomena - Very rapid oxidation of the Zircaloy cladding. This results in severely embrittled cladding that will fragment on reflood quenching. The embrittlement and fragmentation of highly oxidized Zircaloy has been studied extensively for the NRC at Argonne National Laboratory (ANL). The limits on the maximum time-at-temperature which will not result in fragmentation due to thermal shock from reflooding have been determined and can be used in our current models. No additional work is needed or is planned in this area. However, the oxidation kinetics of Zircaloy are not well known above 1800°K (2800°F), and high-burnup fuel may experience considerable swelling due to fission product release.

b. Safety Issues - If the accident is terminated below approximately 3400°F, the issue becomes related to the coolability of a core containing fragmented pieces of oxidized and embrittled Zircaloy-clad fuel rods. Another issue is the extent and amount of fission product release at these higher temperatures.

4. Damage Regime IV (T_{max} < 4700°F (2870°K); any ΔP; any dT/dt)

a. Physical/Chemical Phenomena - Melting of the remaining partially oxidized cladding; reaction of liquid cladding with solid UO_2 to form "liquefied fuel"; flow and refreezing of liquefied fuel to produce "candling" type (cohesive) damage and blockage; continued oxidation of liquefied fuel during flow and after refreezing. The only available data in this regime are those of Hagen at KfK where rod simulators containing a core rod of tungsten (as a heater) surrounded by annular rings of UO_2 were used. More prototypical tests using rods of standard design that are volumetrically heated by either fission or decay heat are required so that the damage and debris formation scenarios for representative fuel rods can be studied and modeled. The effect of high burnup will also be important in this regime and in Regime V below.

b. Safety Issues - The major safety issues for this regime are core coolability (i.e., can the accident be stopped by reflooding) and fission product and hydrogen release from very hot solid fuel rods, liquefied fuel, and fragmented fuel. The fission produce measurements in PBF will complement and verify the out-of-pile experiments on fission products and aerosol release at ORNL.

5. Damage Regime V (T_{max} >4700°F (2870°K)

a. Physical/Chemical Phenomena - Melting of remaining UO_2 and ZrO_2; growth and progression of the melt; foaming of molten UO_2 due to fission product release; interaction of the melt with the pressure vessel and intervals interaction of the melt with water in the vessel and the characteristics of the debris that is formed; hydrogen and fission product release; explosive and nonexplosive steam generation. Except for steam explosion studies currently underway at SANDIA very little information is available on the phenomena mentioned above. New information is definitely required for LWRs in this area.

b. Safety Issues - The safety issues in regieme V are: the coolability of the core debris by reflooding, which is highly dependent on the characteristics of the particulate debris formed by reflood quenching of the melt; rapid steam generation (possibly explosive) and hydrogen generation during reflood quenching of the debris; fission product release during melting and reflood; and the progression of core melt through the reactor internals to attack of the reactor vessel. Melt progression information is important for determining the characteristics of reactor-vessel failure in accidents without recovery and the initial conditions for core-melt entry into the reactor cavity for assessment of the core-melt threat to containment integrity and for PRA. The current treatment of those processes in the MARCH code is unphysical and very conservative.

3.3.2 Damaged Fuel Coolability

A primary goal of the SFD program is to determine, for each state point of severe fuel damage, whether or not the core debris is coolable by simple reflood, and, if not, what the coolant requirements are to achieve coolability. Debris is said to be coolable if a geometry and temperature distribution have been achieved that are stable in time. The coolability approach used in the SFD program is to determine the damage state points for which the core debris is coolable by slow reflood (i.e., stagnant pool) and for those damage state points outside this space, to determine the coolant flow velocity and pressure necessary to achieve coolability.

The most important, most easily defined, and most easily measured coolability limit is the dryout bed specific power at which liquid coolant fails to reach some regions of the debris. It has been shown in fast-reactor safety experiments in the ACRR test reactor with sodium-cooled debris beds that stable temperature distributions and geometries are possible at decay-heat power levels with local dryout in part of the debris bed. However, little is known about the available coolability margins and debris behavior between the point of local dryout and the progression into core melt. Therefore, the well-defined and relatively easy-to-measure dryout limit is the best criterion of coolability to use in reactor safety assessment and research.

A substantial data base and relatively sophisticated analytical models of dryout coolability limits for spatially-uniform packed beds as a function of mean particle size and bed depth have been developed in the fast-reactor safety research program. The experiments have included several coolants, i.e., sodium, water, and organics, and several methods of heating, including fission heating of simulated debris in the ACRR test reactor to simulate fission product decay heating. Lipinski at SANDIA has developed a relatively sophisticated first-principles/model for the dryout limit of a packed unstratified debris bed that agrees well with the world data base for all the liquids tested. This model includes capillary forces and both laminar and turbulent vapor flow. The ACRR experiments with sodium-cooled debris beds have shown that the formation of vapor channels in the bed that can occur at high subcooling can increase the bed dryout limit by about a factor of five. These experiments have also shown that bed stratification with an increase in mean particle size with distance below the bed surface can decrease the bed dryout limit by a factor of five. Verified models of these phenomena and of the onset of channeling in debris beds do not yet exist.

The consensus of experts in this area is that there is a clear need to test and confirm the applicability of current coolability models to important LWR accident conditions , these include very deep beds, high pressure, in let-flow and in particular the expected larger particle sizes expected for LWR severely damaged fuel and core debris, which would substantially increase the coolability of the LWR debris.

3.4 Fission Product Release and Transport (Radiological Source Term)

In response to issues developed by members of the technical community as a result of analyses of the TMI-2 accident, the Nuclear Regulatory Commission requested and the NRC staff prepared NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents." This comprehensive report is the best current summary of capabilities in the technical area. The report was reviewed by internationally known experts and representatives of industry and DOE as well as NRC staff and contractors. The following is an excerpt summary of the state of the art from this report.

1. The current data base suggests that cesium iodide will be the expected predominant iodine chemical form under most postulated light water reactor accident conditions. The formation of some more volatile iodine species (e.g., elemental iodine and organic iodines), however, cannot be precluded under certain accident conditions.

2. The assumed form of iodine (either cesium iodide or elemental iodine) was not predicted to have a major influence on the estimated magnitude of

iodine attenuation in the containment for severe accident sequences with early containment failure in which there is little time for natural fission product retention mechanisms to be effective. However, the assumed chemical form of iodine can influence the predicted attenuation within the reactor coolant system, where, in general, the attenuation factor will be greater for cesium iodide than for elemental iodine (i.e., less iodine will escape into the containment).

3. A number of accident sequences were examined in this report, including several core melt sequences that had been found to be the most important contributors to risk in the Reactor Safety Study (RSS). Reevaluation of fission product release from the fuel indicates that the RSS may have underpredicted the release of certain important radionuclide species during these core melt events. Mechanistic analyses of fission product transport in the containment atmosphere were in reasonable agreement with the empirically based analyses in the RSS. Predictions of the retention of radioactive material within the reactor coolant system (which was not accounted for in the RSS for most accident sequences) range from very little to substantial retention for specific accident sequences involving a water-bounded reactor coolant system (e.g., TMI). In addition, for certain transient initiated core melt sequences where steam flow rates through the reactor coolant system are low and aerosol generation is high, attenuation of fission products within the reactor coolant system could be substantial as a result of agglomeration and fallout of aerosols. Consequently, for certain accident sequences considered in the RSS the release of radionuclides to the environment may have been significantly overpredicted.

However, for other accident sequences (such as large- or medium-sized pipe break accidents) the estimated releases in this report are in approximate agreement with the RSS estimates.

There are very large uncertainties in the release rates for specific radionuclide species. Knowledge of the chemical form of the released radionuclides is, however, quite limited.

The transport behavior of fission products within the reactor coolant system is subject to large uncertainties resulting from limitations in the ability to predict severe accident phenomena, thermal-hydraulic conditions, and the physical and chemical forms of the fission products.

In contrast, the ability to predict the behavior of fission products within the containment structure after release from the primary system is comparatively good for large volume PWR containments. Less well known is the fission product behavior within pressure suppression containments such as in BWRs and in PWR ice condenser plants where the potential attenuation of fission products within the pressure suppression pool and ice beds is subject to large uncertainties. One of the largest uncertainties associated with predicting the amount of radionuclides released to the environment during the most severe accidents (i.e., core melt accidents with containment failure) result from limitations in the ability to predict the timing, mode, and location of containment failure.

The extent to which fission product release to the environment may have been overestimated (or underestimated) in previous studies is difficult to quantify since the range of uncertainty associated with these predictions is very large as a result of the identified limitations in the data base and the early state of development and verification of the predictive methodology.

3.5 Fuel-Melt Interaction (Containment Failure Processes)

The state of the art in studying containment failure processes is defined by two recent studies of severe accidents in the Zion and Indian Point plants. The pertinent reports are NUREG/CR-1409, -1410, and -1411, "Report of the Zion/Indian Point Study," and in NUREG-0850, "Preliminary Assessment of the Core Melt Accidents at the Zion and Indian Point Power Plants and Strategies for Mitigating Their Effects." Both reports have received extensive peer review within the NRC staff and from contractors. Chapter 6 of NUREG/CR-1410 is a summary of the current status of modeling meltdown progression and the resulting threat of the containment. These points are abstracted from that portion of the document cited, and other pertinent comments are interpolated as appropriate. In addition, the Reactor Safety Study and its successor documents, particularly the studies of different containment types have been a major source of information to define the problems and outline the scope of work.

3.5.1 Failure by Hydrogen Burning

The accident at Three Mile Island demonstrated the possibility that hydrogen can be generated in larger quantities than previously considered in NRC regulations. That is, large amounts of hydrogen can be generated in fuel damage accidents, as well as in core melt accidents. In recent licensing actions, NRC has required that licensees install hydrogen control systems in certain small types of containment structures (i.e., ice condenser and BWR pressure suppression models Mark I and Mark II). Rulemaking is also under way to establish new hydrogen control requirements for all construction permit and operating license applicants regardless of containment types.

This failure mode is produced by the pressure loads generated on containment by the combustion of accumulated hydrogen in the containment or possibly by the generation of missiles from the detonation of packets of hydrogen. Also, the burning of hydrogen could affect the operation of safety-related equipment necessary for the safe isolation and shutdown of the plant. This hydrogen would be generated from the reaction of hot steam with zirconium or steel. Additional hydrogen can also be generated later in the accident by the reaction of molten-core material with concrete. Other secondary sources of hydrogen arise from the radiolytic decomposition of water and the corrosion of galvanized materials in the containment and from chemical reactions of sprays with aluminum and other organic/inorganic coatings.

Measures to control and manage accidents involving hydrogen depend on the rate and quantity of the hydrogen released; the distribution of hydrogen, air, and steam; the temperature and pressure in the containment building when the combustion occurs; and the location of the hydrogen release. As noted below, there are uncertainties in a number of these areas.

1. <u>Source of hydrogen</u>: Except in a steam-starved situation, the kinetics for the zirconium-steam reaction is thought to be modeled well by the Baker-Just or Cathcart-Powel models. The steam-steel reaction has a much larger

uncertainty. The importance of this is dependent upon the amount of steel involved when the core slumps and when the vessel fails, releasing molten core and steel to react with water in the reactor cavity. There is a large uncertainty in the relative amounts of hydrogen generated versus steam when the core slumps, depending on the amount of core melt and oxidation prior to slump. Rapid release rates of hydrogen (100-200 lb/min) could cause problems for proposed hydrogen control systems. Additionally, the generation rate of hydrogen and possibly carbon monoxide from molten core/concrete interactions has not been verified, although the rates are calculated to be on the order of 7-10 lb/min for extended periods. Any substantial release of carbon monoxide will generate higher pressures than a corresponding burn of pure hydrogen.

2. Hydrogen release, transport, and mixing: Hydrogen can be released through a relief valve, small or large break or through the high point vent that is now required to be installed on LWRs. The rate of release is dependent on the driving force (system pressure) and the size of the break. At the onset of vessel failure, the remainder of the hydrogen formed but not previously leaked will be released. There is an uncertainty in the release rate and also in the relative amount of steam accompanying the hydrogen release. Once released to the containment, the hydrogen mixes throughout the containment in various compartments and areas, depending on a number of factors (e.g., pressure differences, temperature gradients). Until recently, only limited work had been done on developing analytical models to calculate the transport and mixing of hydrogen in containment, and there is a need for more experimental and analytical work in this area.

3. Combustion of hydrogen: Different igniters or detonators lead to different deflagration and detonation limits. In large volumes with obstructions, and particularly in pipes or ducts, deflagrations may accelerate to detonations, even in mixtures outside the Shapiro-Moffette detonation limits. There is some evidence to indicate that detonations cannot develop in atmospheres containing less than 13 percent hydrogen. A practical lean limit for the ignition of hydrogen with glowplug igniters appear to be somewhat higher than the 4 percent usually determined in laboratory ignition tests using sparks or flames to ignite tubes of gas in upward propagation. Fenwal Laboratories has achieved ignition of mixtures containing 5 percent hydrogen using glowplug igniters. In experiments at Lawrence Livermore Laboratory using a different vessel, attempts to ignite 6 percent mixtures of hydrogen in air failed, but ignition was achieved in 8 percent mixtures.

If hydrogen control by a series of small burns is contemplated, the question of interest is somewhat different from the determination of the minimum combustible limits. Instead, there is a need to know the highest concentration that might not burn under the conditions of the specific ignition system. This level determines the energy loads on the structure. The experiments indicate that the highest concentration that might not burn is about 8 percent (by volume). The use of an average concentration of 10 percent (by volume) for accident computations should provide some margin for variations in hydrogen concentrations throughout the containment building.

Not all compositions within the detonation limits necessarily detonate when ignited. Factors that enhance the probability of detonations are (1) shock waves accompanying ignition, (2) turbulence, and (3) large volumes, especially those with obstructions because they promote turbulence. A recent study of the detonation of hydrogen concludes that the absolute detonation limits (with large explosive detonators) are 13-70 percent by volume in air, rather than the 18-56 percent by volume indicated by Shapiro and Moffette. Uncertainties in the H_2/air/steam deflagration limits, questions on deflagration-to-detonation transitions, and questions on autoignition are being addressed in the NRC program on hydrogen behavior at SANDIA. Also large scale tests are planned by EPRI which will provide insights on the effects of scale on hydrogen combustion.

4. Efficacy of Mitigation Systems: In response to the Commission requirements on the control of hydrogen in LWR accidents, various mitigation systems and schemes have been proposed. The efficacy of this system for various accident sequences, particularly for specific plant designs has not been totally demonstrated. To remove this uncertainty and to investigate potentially better systems, there is work being sponsored by RES on mitigation of hydrogen effects at SANIDA. Also work is being sponsored by EPRI and the industry to assess the efficacy of proposed mitigation system.

5. Equipment Survival: A hydrogen burn can potentially generate temperatures and pressures exceeding the qualification limits used to test safety grade equipment. These higher temperatures and pressures could lead to the failure of a component important in the safe isolation and shutdown of the plant. In order to remove the uncertainty and to develop improved testing methods for safety-grade equipment and in order to meet the test posed by a hydrogen burn, an analytical and experimental program on equipment survival was recently initiated. It is anticipated that questions in this area should be answered within the next 2 years.

3.5.2 Failure by Steam-Spike Overpressurization

This failure mode is induced by the rapid generation of steam when a mass of molten fuel drops into a cavity filled with water or when, as a result of depressurization from vessel breach, the accumulators come on and dump a large mass of water on the very hot and molten fuel. The current state of the art includes a high degree of uncertainty as to how molten fuel and structural material interaction within the pressure vessel actually causes a breach; while a model is available within MARCH, it has significant uncertainties, as described in Chapter 6 of NUREG/CR-1410. The conclusion in that report is that two types of failure are judged most probable: (1) a catastrophic failure of the central portion of the lower head after about 30 minutes of plastic deformation, or (2) a rapid splitting in the form of a small crack at the periphery of the lower hemisphere, depending on whether or not the pressure vessel had been depressurized. Thus, there is a wide range of possible vessel failures, the more probable being at the relative extremes. The mode of failure dictates the extent to which one has to consider catastrophic mixing of a large mass of molten fuel with water in the cavity, or, for that matter, within the vessel during the initial stages of meltdown. In any event, steam will be generated, perhaps at a rapid enough rate to be called a "steam explosion." Such steam, explosion or not, constitutes a significant source of overpressure. In NUREG-0850, the estimate is made (for the Zion and Indian Point plants) that overpressures during this event are large, about 100 psia, with uncertainties of about 20 percent, at least. A chief contributor to the uncertainty is the nature of possible hydrogen burns during this period of the transient. Estimates are that the mixture of steam and air is sufficient to suppress hydrogen burning and hence to suppress a potential source of greater overpressure. On the other hand, experiments with pouring molten fuel simulants into water show a large hydrogen burn coincident with the steam generation, so dynamic effects may be important.

3.5.3 Failure by Steam Explosion

The analysis in NUREG-0850 ascribes a low likelihood to failure by missiles generated by steam explosions. NUREG/CR-1411 had predicted that if a steam explosion causes a failure, it will be at the lower hemisphere and hence unlikely to cause significant missile damage. Continued caution is indicated in this area, however, because of the difficulty in extrapolating from small-scale (few kilograms) to large-scale (several thousand kilograms) mixing experiments with confidence prior to the existence of a good model of the explosion-detonation process. Such models should be available in the near future.

3.5.4 Failure by Slow Overpressurization

A simple heat balance, even taking into account the slow transfer of heat through the containment, leads one to predict that unless long-term cooling is restored, the containment will eventually fail by slow overpressurization. The time of failure depends on the containment failure characteristics, the rate of gas generation during the core-concrete-coolant interaction, and the potential for hydrogen burning during the period of slow overpressurization. In the cases considered in NUREG-0850, the key uncertainty for the case of a flooded cavity was the failure characteristics of the containment that led to an estimated uncertainty of about 5 hours in the failure time. In the case of a dry cavity, the principal uncertainty is the rate of gas formation, but the failure time is in any case predicted to be late. For smaller containments or containments of lower pressure capabilities, other sources of uncertainty might also become important.

3.5.5 Failure by Basemat Melt-Through

There is some contention whether this mode of failure is inevitable, as assumed in WASH-1400, or indeed in some circumstances whether it will occur at all. In any event, the failure time is quite late. The conclusion in the Kemeny Report was that such failure would not have occurred if the TMI accident had gone to gross meltdown, and in NUREG-0850 the conclusion is that, with a flooded cavity, basemat penetration may not occur. For conservatism, a time of failure of 3 days was assumed for this mode. Two major sources of uncertainty are the extent to which the interacting mass of core and concrete can be cooled by the overlying water and the rate at which hot but solidified fuel melts through concrete. In some experiments, it has been observed that water was held away from the interacting masses by a crust of material, so that establishing a coolable debris bed prior to significant attack on the concrete basemat may be prerequisite for preventing this mode of failure.

3.6 Accident Mitigation

The state of the art in mitigation systems is discussed in the Zion/Indian Point studies referenced above. Other studies in more detail have focused on one or more specific systems. Core retention systems are reviewed in "A Review of Core Retention Concepts to Light Water Reactor Containments," NUREG/CR-2155, where it is concluded that core retention systems can only reduce risk significantly if above-grade containment ruptures are prevented by another system, such as a filtered vent. However, analyses and proposed designs that show the potential effectiveness of a retainer do not adequately address all the possible thermal-hydraulic and materials uncertainties associated with the problem. Even though basemat melt-through may not be a significant source of risk, a core retention system could be designed to limit the gas generation by fuel-melt interaction, thereby lessening the load on the above-grade containment. This benefit was not explored in depth in NUREG/CR-2155. Experimental work is necessary to provide information on how well the proposed concepts actually work. Analyses and experiments must be integrated to investigate the following detailed heat transfer and materials concerns before an effective core retainer can be identified:

- Distribution of core debris in the reactor cavity,
- Differentiation of melt into immiscible metallic and oxidic phases,
- Complex nature of the oxidic phase,
- Correct partitioning of heat sources,
- Radiative heat transfer to upper containment,
- Scaling of laboratory experiments to full-core situations,
- Chemical reactions,
- Formation of eutectics,
- Effects of water,
- Actual mechanisms of melt penetration, and
- Formation of gases and liberation of fission products.

Vented filtered containment (VFC) was one of nine alternatives considered in "A Value-Impact Assessment of Alternative Containment Concepts," NUREG/CR-0165. VFC was judged to offer the greatest potential for reducing public risk for the least impact.

Core melt mitigation systems for the ice condenser plant were studied at INEL and reported in "Phase-2 Status Report - Core Melt Mitigation System Design for an Ice Condenser Plant" (to be published). This study concludes that hydrogen presents the dominant threat to containment amd that postaccident inerting offers an attractive way to deal with this threat and presents several conceptual designs to achieve this goal.

In all the studies done to data, the unifying fact found in them all is the recognition that more research and detailed designs are needed to support any judgment whether or not to require the addition of further mitigation systems.

4. PROGRAM LOGIC, SCHEDULE, INTERFACES, AND COSTS

The logic, overall schedule including major milestones and key interfaces for the Severe Accident Research Program are shown in Figure 4-1 and Table 4-1. The program structure is derived from six key Decision Units or Subelements in the RES Long-Range Research Plan (NUREG-0740), namely Reliability and Risk Assessment, Severe Accident Sequence Analysis, Behavior of Damaged Fuel, Fission Product Release and Transport, Fuel Melt, and Accident Mitigation. These Decision Units or Subelements comprise the 1# program elements that yield the three major product catagories shown in Figure 1. These major product categories provide the following types of information and technical bases for policy decision and regulatory products (regulations, regulatory guides, standards, and standard review plan revisions) for severe accidents in nuclear power plants:

1. Data and guidelines for refinements to plant systems design and operating practices,

2. Verified methodology for accident loadings and system responses, and

3. Information and methodology for decision on potential risk-reduction add-ons.

Figure 4-1 shows the summary schedule in terms of major milestones (listed in Table 4-1) for each of the 13 program elements. The logic for the program is

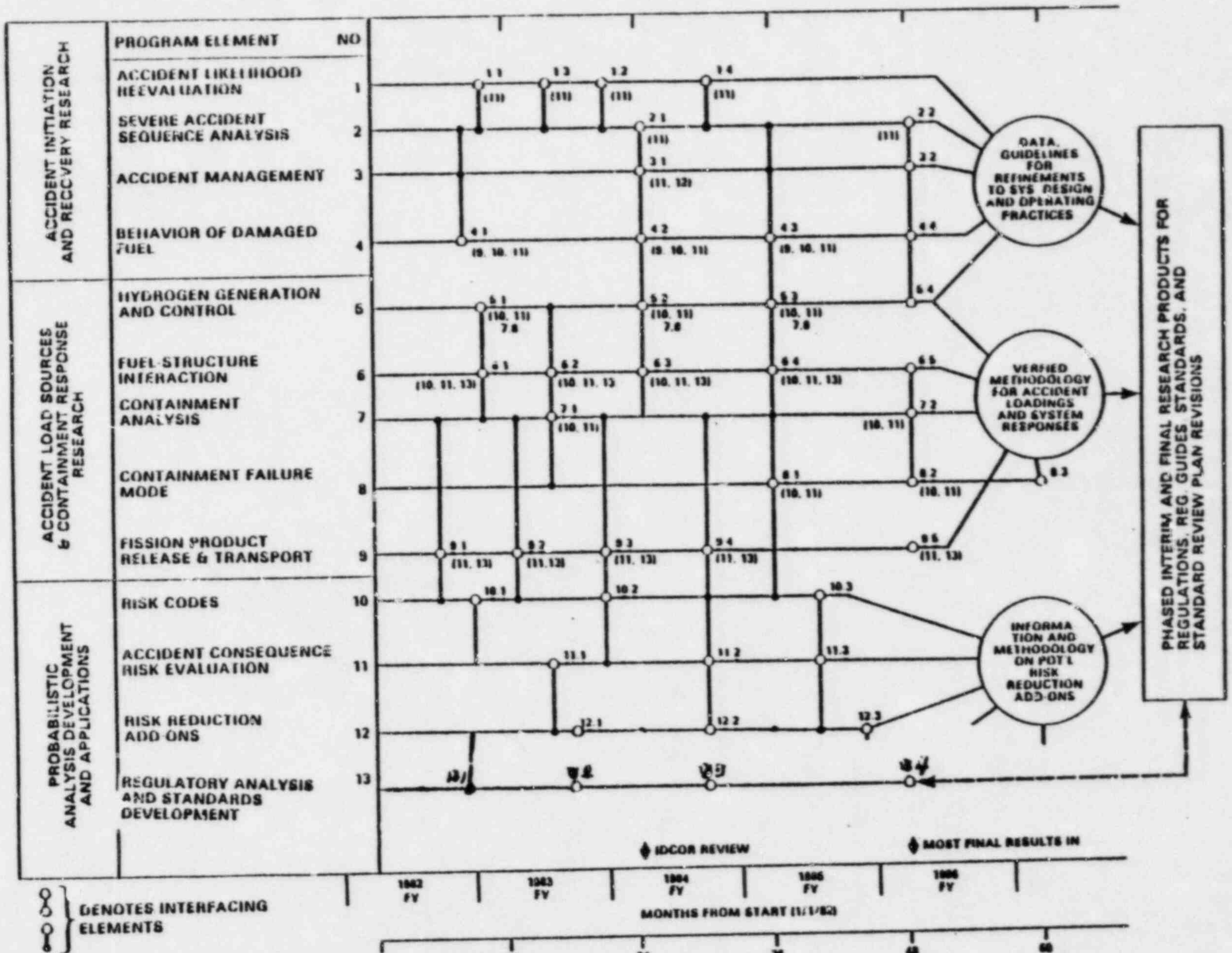


Figure 4-1 Severe Accident Research Program Plan

Table 4-1

SEVERE ACCIDENT RESEARCH PROGRAM
MAJOR MILESTONES

MAJOR MILESTONES		DATE START = 1/1/82
1.	Accident Likelihood Reevaluation	Mo. from Start
	1.1 RSSMAP/IREP Final Report	9
	1.2 Precursor Studies ... Phase 1	9
	... Phase 2	9
	... Final	21
	1.3 Station Blackout Studies Final Report	9
	USI Resolution	15
	1.4 Accident Sequence Reevaluation ... Phase 1	6
	... Phase 2	18
	... Phase 3	30
2.	Severe Accident Sequence Analysis	
	2.1 Assessment of Operator Guidelines	24
	2.2 Management Strategies for Severe Accidents	48
3.	Accident Management	
	3.1 Operating Procedure Guidelines for Recovery from Core Damage Event	24
	3.2 Refinements to System Design and Operating Procedures	48
4.	Behavior of Damaged Fuel	
	4.1 SCDAP MOD 0 Available, First PBF Phase I Test and First ACRR Coolability Experiment	8
	4.2 Complete Phase I PBF Tests and Initial ACRR Separate Effects for Damage State Coolability Criteria with SCDAP MOD 1. TMI-2 RPV Head-lift	24
	4.3 SCDAP MOD 2 with Improved ACRR Phenomenological Model. Initial NRU Full-length Verification. Whole-core Analysis Available	36
	4.4 Phase II PBF, ACRR, and NRU results for SCDAP Coolability and TMI-2 Data for SCDAP/Whole Core Benchmark	48
5.	Hydrogen Generation and Control	
	5.1 Improved Combustion Models, Preliminary Thermal Models for Equipment Survivability, Analysis of H_2 Control for Two Containment Types	12

Table 4-1 (Cont.)

MAJOR MILESTONES		Mo. from Start
	5.2 Assessment of Alternative Control Methods, Improved Equipment Response Model and Improved Transport Code	24
	5.3 Preliminary Assessment Flame Acceleration, other Plant-Specific Analysis	36
	5.4 Large-Scale Proof Tests	48
6.	Fuel-Structure Interaction	
	6.1 Large-Scale Fuel-Melt Interaction Transient tests	12
	6.2 CORCON MOD 2	18
	6.3 Large-Scale Melt Interaction Sustained Tests, Retrofit Retention Concepts, Melt/Concrete Aerosol Source	24
	6.4 CORCON Verification Tests, Castable Concrete Tests	36
	6.5 Fuel Debris-Coolant-Concrete Interaction Tests	48
7.	Containment Analysis	
	7.1 Improved Version of CONTAIN	18
	7.2 CONTAIN Verification	48
8.	Containment Failure Mode	
	8.1 Static Pressure Loads - Steel and Concrete Containment	36
	8.2 Static Loads Reinforced Concrete, Dynamic Loads - Steel and Concrete	48
	8.3 Dynamic Loads - Reinforced Concrete	60
9.	Fission Product Release and Transport	
	9.1 TRAP-MELT MOD 2 for RSC Transport	6
	9.2 NUREG-0772 Follow-On - Reassessment of Source Term	15
	9.3 NSPP Aerosol-Steam Tests, Release Rates Irradiated-Fuel to 2000°C and Melt Aerosols Source, Chemical Species - Vapor and Aqueous, TRAP-MELT MOD 3	21
	9.4 ESF Severe Accident Performance, TRAP-MELT Verification for Volatile F.P. Transport	30

Table 4-1 (Cont.)

MAJOR MILESTONES	Mo. from Start
9.5 TRAP-MELT Aerosol Transport, Release Rates Irradiated and Simulated fuel to 2800°C	48
10. Risk Codes	
10.1 MARCH-2/MATADOR	9
10.2 Preliminary Version of MELCOR	21
10.3 Final Version of MELCOR	40
11. Accident Consequence and Risk Evaluation	
11.1 Evaluations with MARCH-2/MATADOR	15
11.2 Evaluations with Preliminary Version MELCOR	27
11.3 Evaluations with Final Version MELCOR	40
12. Risk Reduction and Add-On Cost Benefit	
12.1 Integrated Risk-Cost (RSSMAP BASIS)	7
12.2 Integrated Risk-Cost (MARCH-2 Basis)	18
12.3 Integrated Risk-Cost (Early MELCOR)	30
12.4 Integrated Risk-Cost (Final MELCOR)	43
13. Regulatory Analysis and Standards Development	
13.1 Regulatory Options for Severe Accident Risk Limitation (preliminary findings)	8
13.2 Regulatory Options for Severe Accident Risk Limitation (interim findings)	20
13.3 Regulatory Options for Severe Accident Risk Limitation (revised findings)	32
13.4 Regulatory Options for Severe Accident Risk Limitation (revised findings)	48

indicated by the vertical tielines between program elements in parthentheses. This logic is based on the need to transfer results among program elements for timely accomplishment of element objectives, as required for each of the three research product categories and the regulatory end-products. The program logic also provides a consistent basis for dealing with initiatives such as IDCOR. Application and integration of the research products into regulatory end-products is accomplished in Program Element 13, Regulatory Analysis and Standards Development.

The timing of the program is consistent with staff proposals made in support of FY 1983 budget submittals.

Schedule of Results

Currently we are in the process of developing a detailed network for the flow of information between programs and schedule of milestones. Illustrated in Figures 4-2, -3 and -5 are overview networks for the 13 program elements. The three figures provide a listing of some of the main products that will be generated from the carious program elements. The overviews are set-up to provide the products available in 2 and 4 year time period. The list is not complete and in most cases the products listed actually represent several products which were condensed to simplify the figures. A more detailed network will be available in 4-6 weeks, which will give a better indication of the timing of the results and interfaces betweeen program elements.

While it is anticipated that major outputs covering most outstanding problems will be available in 4 years, there are significant intermediate results that have a high degree of usefulness and that can be used to draw together interim assessments across the board to define some issues more narrowly, resolve others, and provide interim bases for policy considerations on severe accident regulatory requirements.

A body of knowledge exists now that constitutes the current state of the art, and well before the program described here reaches maturity and begins to wind down, there will be major additions to that body. Some major interim stages are:

End of first year: The first results of the Severe Core Damage Analysis Package (SCDAP) should be available together with initial data from tests in PBF and ACRR to improve our knowledge of how to cope with fuel damage in the presence of degraded core cooling (in particular what core coolant level will, if maintained, limit further damages); an improved version of the MARCH code for risk analysis (MARCH 2); and improved hydrogen combustion models should be available. This should permit us to better define the problems in analyzing small-break accidents with respect to the role of hydrogen, whose generation will be predicted using SCDAP and whose combustion will be modeled by advanced methods. By using improved versions of MARCH, we will be able to factor these new models into an iteration of the Severe Accident Sequence Analysis as well as the risk analyses proper.

End of second year: The first integrated reappraisal of all three products should be available at this time, including the first output of the value impact of possible risk-reduction add-ons; guidelines for operating procedures for recovering from core damage events; integrated assessment of containment loads; hydrogen generation and control analyses; and better data on containment loads from core-concrete interaction. Key input will come from a best-estimate reassessment of the release-from-plant radiological source term, available early in this period. This integrated reappraisal will incorporate the assessments provided by the industry group, IDCOR, whose final report is due at this time.

A reappraisal will again be possible, during the third year of the program, including risk reduction reevaluation via the MELCOR code that allows for the systematic introduction of new models and data into the risk analyses; the results of industry assessments, including early results of the NREP program, and a much-improved SCDAP model.

The interim and final research products of the program are phased to provide information and technical bases for draft policy or regulatory options developed in Program Element 13 within 18, 30, and 48 months after the start of the Severe Accident Research Program.

Projected costs by program element are given in Table 4-2.

TABLE 4-2

SEVERE ACCIDENT RESEARCH PROGRAM

PROJECTED COSTS

(MILLIONS OF DOLLARS)

PROGRAM ELEMENTS	FY 82	FY 83	FY 84	FY 85
1. ACCIDENT LIKELIHOOD REEVALUATION	2.5	2.5	1.0	0
2. SEVERE ACCIDENT SEQUENCE ANALYSIS	2.3	3.4	3.6	3.6
3. ACCIDENT MANAGEMENT	0	0.8	1.0	1.3
4. BEHAVIOR OF DAMAGED FUEL	18.0	25.0	29.0	25.0
5. HYDROGEN GENERATION AND CONTROL	3.2	4.3	4.5	3.8
6. FUEL-STRUCTURE INTERACTION	4.5	4.8	5.0	4.7
7. CONTAINMENT ANALYSIS	1.0	1.3	1.0	0.7
8. CONTAINMENT FAILURE MODE	1.0	2.0	2.5	2.7
9. FISSION PRODUCT RELEASE & TRANSPORT	4.3	6.7	8.5	5.5
10. RISK CODES	2.4	2.1	1.9	0.9
11. ACCIDENT CONSEQUENCE RISK EVALUATION	0	0.3	0.4	0.3
12. RISK REDUCTION ADD-ONS	1.3	2.4	2.3	1.1
13. REGULATORY ANALYSIS	2.3	3.0	4.0	3.4
TOTALS	42.8	58.6	64.7	53.0

TABLE 4-2 (CONTINUED)

SEVERE ACCIDENT RESEARCH PROGRAM

TOTAL COSTS

(MILLIONS OF DOLLARS)

		Σ FY 82-85
1.	ACCIDENT LIKELIHOOD REEVALUATION	6.0
2.	SEVERE ACCIDENT SEQUENCE ANALYSIS	12.9
3.	ACCIDENT MANAGEMENT	3.1
4.	BEHAVIOR OF DAMAGED FUEL	97.0
5.	HYDROGEN GENERATION AND CONTROL	15.8
6.	FUEL-STRUCTURE INTERACTION	19.0
7.	CONTAINMENT ANALYSIS	4.0
8.	CONTAINMENT FAILURE MODE	8.2
9.	FISSION PRODUCT RELEASE AND TRANSPORT	25.0
10.	RISK CODES	7.3
11.	ACCIDENT CONSEQUENCE RISK EVALUATION	1.0
12.	RISK REDUCTION ADD-ONS	7.1
13.	REGULATORY ANALYSIS AND STANDARDS DEVELOPMENT	12.7
	TOTALS	219.1

Figure 4-2

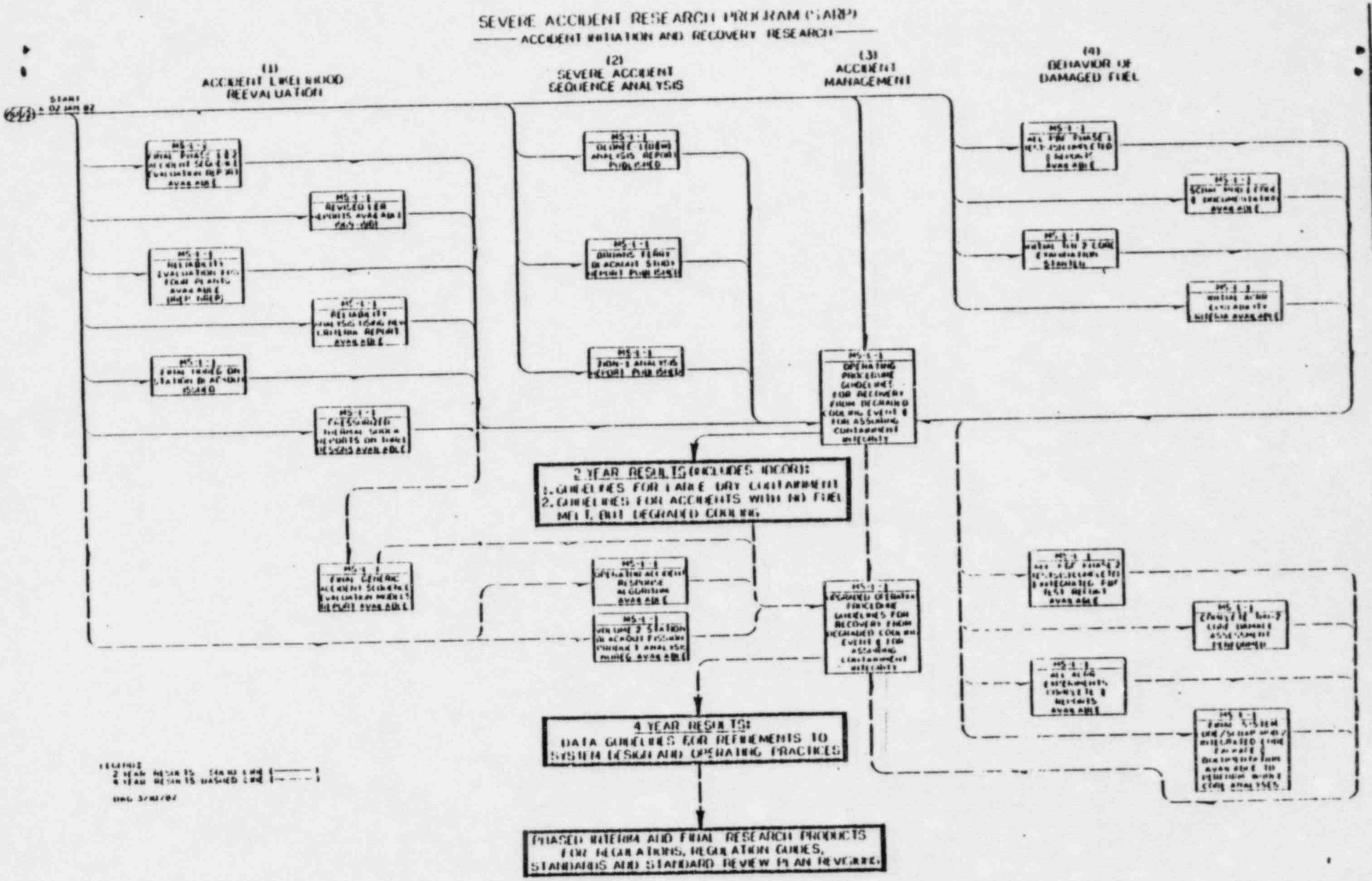

SEVERE ACCIDENT RESEARCH PROGRAM (SARP)
ACCIDENT INITIATION AND RECOVERY RESEARCH
(1) ACCIDENT LIKELIHOOD REEVALUATION
(2) SEVERE ACCIDENT SEQUENCE ANALYSIS
(3) ACCIDENT MANAGEMENT
(4) BEHAVIOR OF DAMAGED FUEL
START
OPERATING PROCEDURE GUIDELINES FOR RECOVERY FROM DEGRADED COOLING EVENT & FOR ASSURING CONTAINMENT INTEGRITY
2 YEAR RESULTS (INCLUDES IDCOR):
1. GUIDELINES FOR LARGE DRY CONTAINMENT
2. GUIDELINES FOR ACCIDENTS WITH NO FUEL MELT, BUT DEGRADED COOLING
4 YEAR RESULTS:
DATA GUIDELINES FOR REFINEMENTS TO SYSTEM DESIGN AND OPERATING PRACTICES
PHASED INTERIM AND FINAL RESEARCH PRODUCTS FOR REGULATIONS, REGULATION GUIDES, STANDARDS AND STANDARD REVIEW PLAN REVISIONS

Figure 4-3

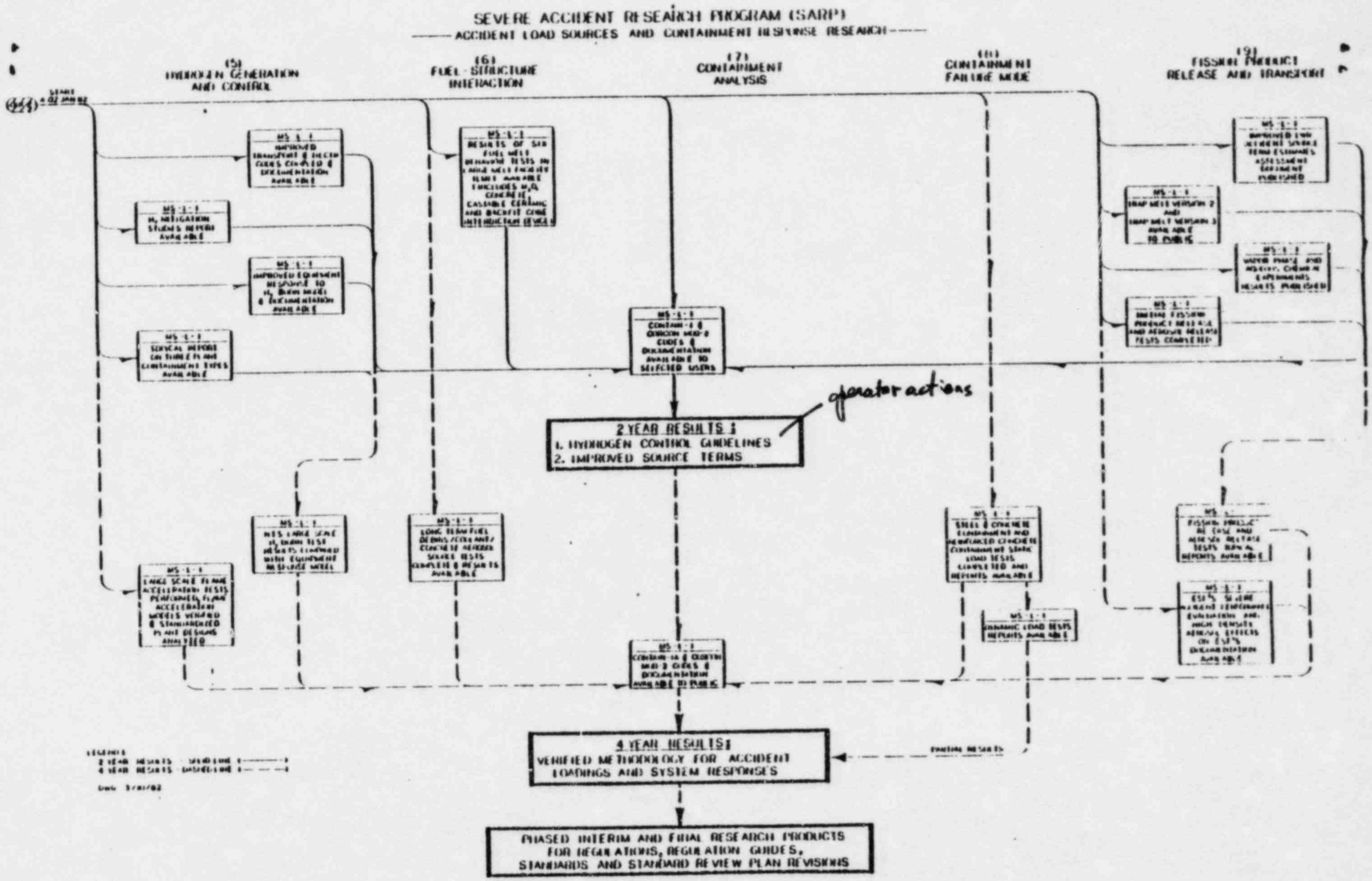
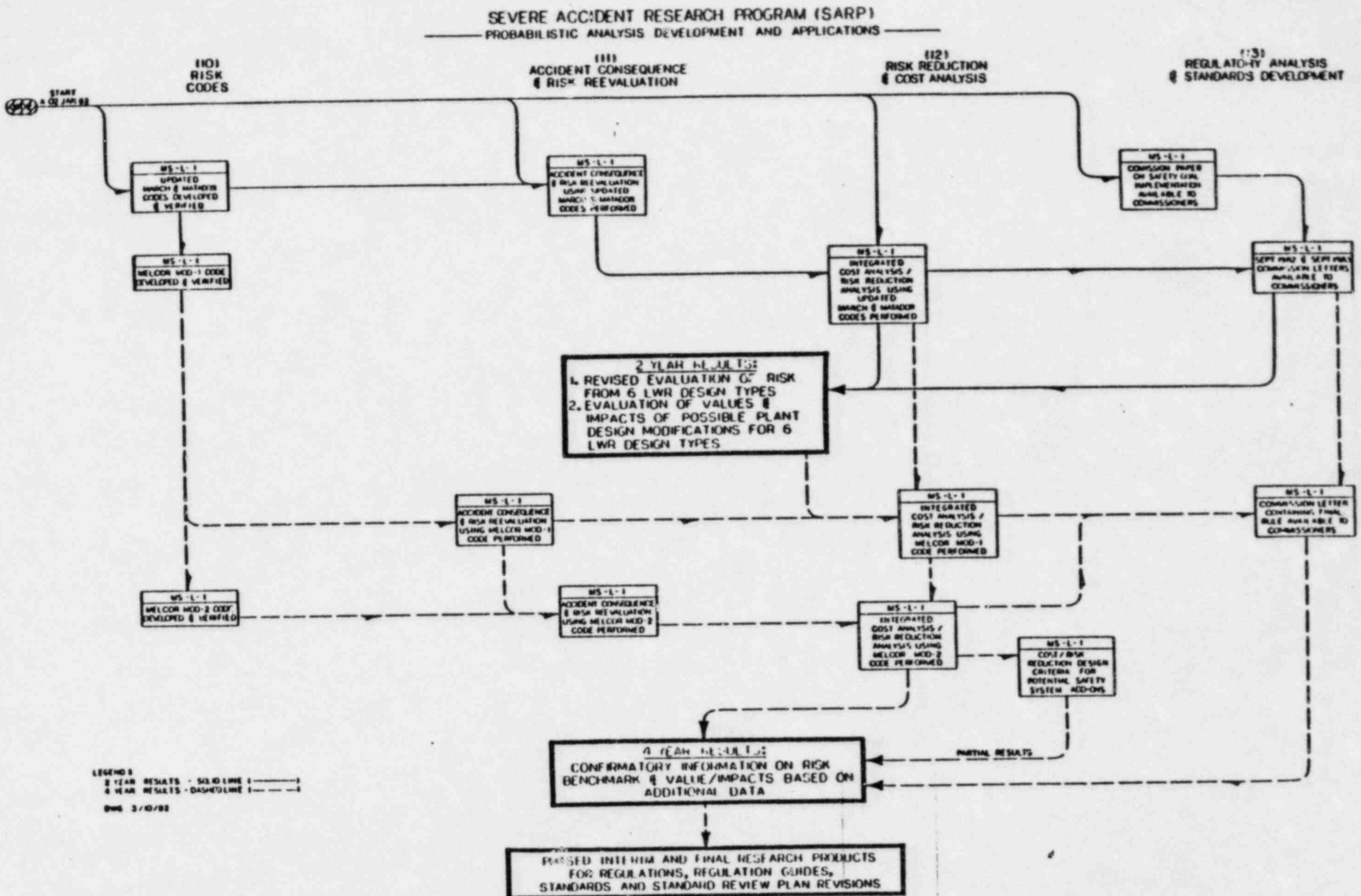

SEVERE ACCIDENT RESEARCH PROGRAM (SARP)
ACCIDENT LOAD SOURCES AND CONTAINMENT RESPONSE RESEARCH
(5) HYDROGEN GENERATION AND CONTROL
(6) FUEL-STRUCTURE INTERACTION
(7) CONTAINMENT ANALYSIS
(8) CONTAINMENT FAILURE MODE
(9) FISSION PRODUCT RELEASE AND TRANSPORT
2 YEAR RESULTS:
1. HYDROGEN CONTROL GUIDELINES
2. IMPROVED SOURCE TERMS
operator actions
4 YEAR RESULTS:
VERIFIED METHODOLOGY FOR ACCIDENT LOADINGS AND SYSTEM RESPONSES
PHASED INTERIM AND FINAL RESEARCH PRODUCTS FOR REGULATIONS, REGULATION GUIDES, STANDARDS AND STANDARD REVIEW PLAN REVISIONS

Figure 4-4


SEVERE ACCIDENT RESEARCH PROGRAM (SARP)
PROBABILISTIC ANALYSIS DEVELOPMENT AND APPLICATIONS
(10) RISK CODES
(11) ACCIDENT CONSEQUENCE & RISK REEVALUATION
(12) RISK REDUCTION & COST ANALYSIS
(13) REGULATORY ANALYSIS & STANDARDS DEVELOPMENT
START
MS-L-1
UPDATED MARCH & MATADOR CODES DEVELOPED & VERIFIED
MS-L-1
ACCIDENT CONSEQUENCE & RISK REEVALUATION USING UPDATED MARCH & MATADOR CODES PERFORMED
MS-L-1
COMMISSION PAPER ON SAFETY GOAL IMPLEMENTATION AVAILABLE TO COMMISSIONERS
MS-L-1
MELCOR MOD-1 CODE DEVELOPED & VERIFIED
MS-L-1
INTEGRATED COST ANALYSIS / RISK REDUCTION ANALYSIS USING UPDATED MARCH & MATADOR CODES PERFORMED
MS-L-1
SEPT 1982 & SEPT 1983 COMMISSION LETTERS AVAILABLE TO COMMISSIONERS
2 YEAR RESULTS:
1. REVISED EVALUATION OF RISK FROM 6 LWR DESIGN TYPES
2. EVALUATION OF VALUES & IMPACTS OF POSSIBLE PLANT DESIGN MODIFICATIONS FOR 6 LWR DESIGN TYPES
MS-L-1
ACCIDENT CONSEQUENCE & RISK REEVALUATION USING MELCOR MOD-1 CODE PERFORMED
MS-L-1
INTEGRATED COST ANALYSIS / RISK REDUCTION ANALYSIS USING MELCOR MOD-1 CODE PERFORMED
MS-L-1
COMMISSION LETTER CONTAINING FINAL RULE AVAILABLE TO COMMISSIONERS
MS-L-1
MELCOR MOD-2 CODE DEVELOPED & VERIFIED
MS-L-1
ACCIDENT CONSEQUENCE & RISK REEVALUATION USING MELCOR MOD-2 CODE PERFORMED
MS-L-1
INTEGRATED COST ANALYSIS / RISK REDUCTION ANALYSIS USING MELCOR MOD-2 CODE PERFORMED
MS-L-1
COST / RISK REDUCTION DESIGN CRITERIA FOR POTENTIAL SAFETY SYSTEM ADD-ONS
PARTIAL RESULTS
4 YEAR RESULTS:
CONFIRMATORY INFORMATION ON RISK BENCHMARK & VALUE/IMPACTS BASED ON ADDITIONAL DATA
PROPOSED INTERIM AND FINAL RESEARCH PRODUCTS FOR REGULATIONS, REGULATION GUIDES, STANDARDS AND STANDARD REVIEW PLAN REVISIONS
LEGEND:
2 YEAR RESULTS - SOLID LINE
4 YEAR RESULTS - DASHED LINE
DWG 3/10/82

5. PROGRAM ELEMENTS

This chapter contains a description of each of thirteen program elements of this plan. Each element is described in detail using the following format:

1. Element Description
2. Technical Issues Resolved by This Element
3. Key Interfaces with Other Elements
4. Background and Status
5. Plan of Work as a Function of Time

5.1 Accident Likelihood Analysis

5.1.1 Element Description

In this element a number of studies will be performed to reassess the predictions of severe accident sequences and likelihoods made in PRAs. This reassessment 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, as distinguished from full core meltdown accidents. Thus in this work explicit consideration is being given to issues such as the impact of risk of:

- external events (e.g., earthquakes, floods) and other similar common-cause events (e.g., fires);
- internally and externally-initiated sabotage; and
- operator intervention to arrest core damage prior to large-scale fuel melting.

The programs of this element are one part of a package of programs related to probabilistic risk assessment (PRA). This work package (Elements 5.1, 5.10, 5.11, and 5.12) has as its overall objective the analysis and periodic updating of the characterization of:

- The risk associated with current LWRs; and
- The risk reduction benefit and costs of possible plants modifications.

As the periodic updates are completed, the technical work of these elements will then be used in concert with Element 5.13 (Regulatory Analysis and Standards Development) to address issues relating to:

- safety goals; and
- cost-effective means for reducing LWR risk.

5.1.2 Technical Issues Resolved by This Element

Technical issues to be addressed in this element include:

1. Development of more robust predictions of the variety, qualitative character, and likelihood of severe accidents in a variety of LWR design types;

2. Estimation of the contribution to risk originating in external events and sabotage;

3. Relative likelihood of "TMI-like" accidents as opposed to full core-melt accidents; and

4. Identification of important precursors to severe accidents from actual LWR operating experience.

5.1.3 Key Interfaces with Other Elements

This element has key interfaces with four elements:

1. To provide additional information to the Severe Accident Sequence Analysis element relating to the relative importance of different accident sequences;

2. To provide accident sequence likelihood information to the Accident Consequence and Risk Reevaluation element for combination with consequence analyses into risk predictions;

3. To provide accident sequence likelihood information to the Risk Reduction and Cost Analysis element for use in evaluating the risk reduction "benefit" of possible plant modifications, and

4. To provide information on the potentially serious vulnerabilities of reactor plants to severe accidents for use in the Regulatory Analysis element.

5.1.4 Background and Status

Individual programs in this element are as follows:

1. Accident Sequence Evaluation Program

   In this program, reviews are to be made of the accident sequence (event tree) evaluations in plant-specific risk assessments such as the Reactor Safety Study (RSS) and the RSS Methodology Applications Program (RSSMAP) (see below). These risk assessments, the reevaluated event trees, and the event likelihood assessments from this program will be used as the foundation from which the second and subsequent risk analyses of plant modifications (discussed below) will depart. The objective of the event tree reevaluation will be to consider the need for and to 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, operator error, etc.; and to attempt to make the event trees more generic than originally established. Studies of sabotage, earthquakes, etc. are being developed to draw upon ongoing research programs such as the Seismic Safety Margins Research Program.

This program is now in the middle of its first iteration of sequence likelihood updating; completion of this phase is planned for mid-1982.

2. 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 are made. The screening of events has now been under way for more than a year. A report will be published in the spring of 1982 covering precursor events from 1969 to 1979. Subsequent reports will carry the analysis up to 1981 and will address problems of completeness and likelihood estimation in the current (i.e., spring 1982) report.

3. Reactor Safety Study Methodology Applications Program (RSSMAP)

The RSSMAP program is intended to apply the methods and insights of the Reactor Safety Study (RSS, WASH-1400) to a somewhat broader spectrum of LWR designs. Event tree and limited-scope system reliability analysis has been performed on four designs: a B&W 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. In addition, the program has recently been amended to include the analysis of a GE plant with a Mark II containment. The final product of each plant study is a discussion of the likelihood of experiencing serious core damage, of having particular magnitudes of releases of radioactive material from the plant (i.e., the RSS "release categories") and an explanation of what types of accidents (e.g., station blackout, ATWS) contribute importantly to these likelihoods of releases.

Reports on three of the four original RSSMAP plant studies are now published (NUREG/CR-1659, Volumes 1, 2, and 4). The remaining study on a Combustion Engineering plant will be published in late FY 1982 as NUREG/CR-1659, Volume 3. It is planned that the Mark II analysis will be completed by late FY 1982 or early FY 1983.

4. Interim/National Reliability Evaluation Program (IREP/NREP)

In the IREP program, PRAs are being performed on a set of plants, using current 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 the 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 latter, larger-scale effort.

The five plant analyses now under way in IREP are scheduled for completion in FY 1982; NREP studies have not yet been initiated.

5. Industry PRA Reviews

In addition to the plant PRAs being performed under RSSMAP, IREP, and NREP, licensees have initiated (for varying reasons) PRAs on specific plants. It is planned that, as such PRAs become available, reviews will be undertaken and the results incorporated into the overall accident sequence likelihood reassessments being performed in this element.

6. 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 (in part) of its 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.

5.1.5 Plan of Work As a Function of Time

It is planned to update the element's accident sequence likelihood evaluations on roughly an annual basis (i.e., the program will be performed iteratively). The timing on the completion of these iteractions and interrelationships among the described programs are shown in Figure 5.1.

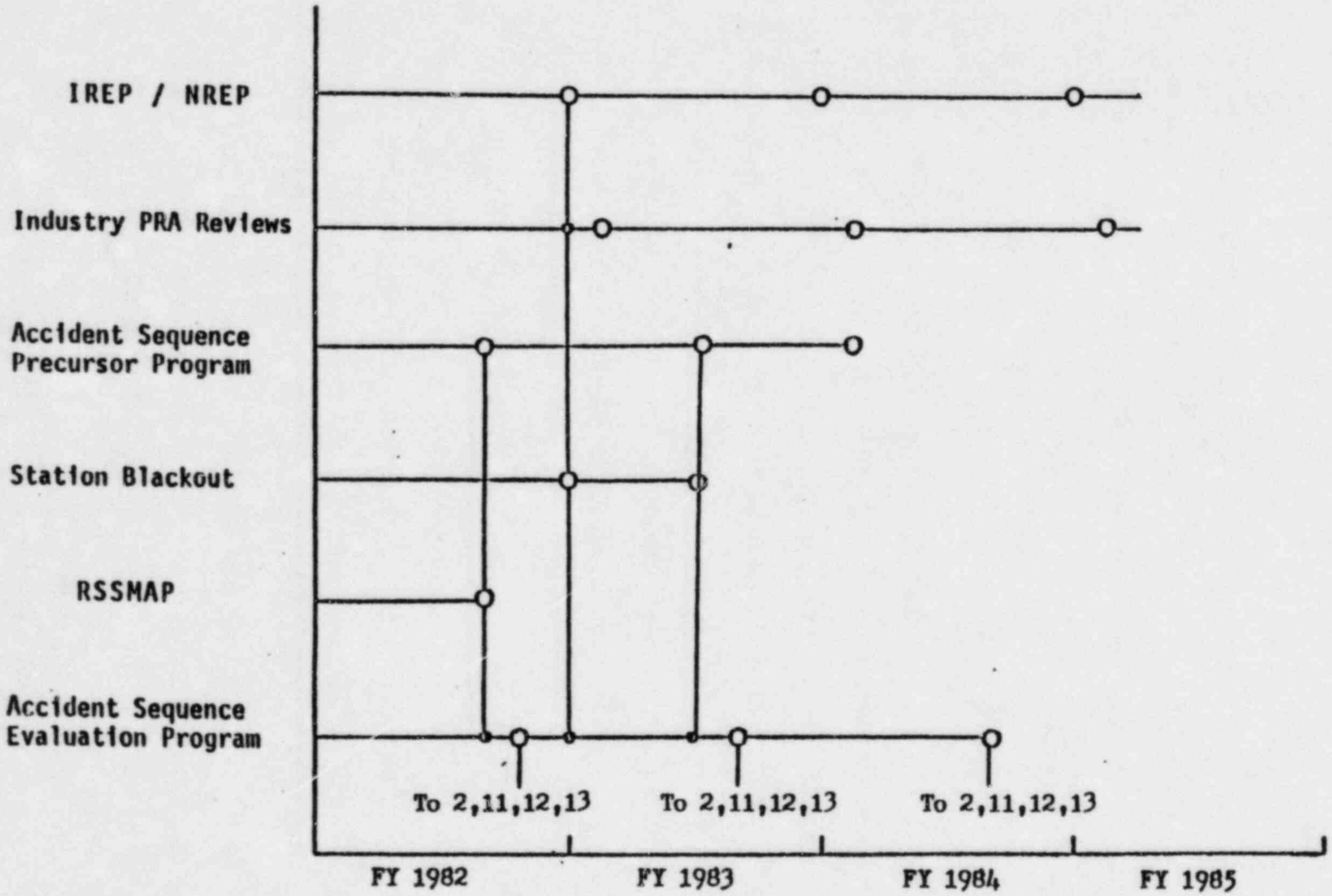


Figure 5.1 Accident Likelihood Reevaluation

5.2 Severe Accident Sequence Analysis

5.2.1 Element Description

This element addresses the problem of improving the understanding of reactor accidents both within and beyond the design basis in order to develop better strategies to prevent, manage, and mitigate severe accidents. Insights into issues will be gained by applying best-estimate state-of-the-art codes (e.g., RELAP, TRAC, MARCH/CORRAL, CRAC), risk assessment methodologies, and plant operation procedures to several specific plants.

One limitation is that SASA is an analytical program. SASA has the capability to give firm recommendations on when a safety action should be automated in performance to operator action or when strict procedures should be provided in place of operator guidelines based on symptoms. SASA will be looking into these issues.

5.2.2 Technical Issues Resolved by This Element

The issues being addressed by this element include severe accident analysis for specific plant design, licensing and safety concerns generated by NRR Abnormal Transient Operator Guidelines (ATOGs), operator instrumentation information needs, fission product release and transport, NRC unresolved safety issues such as:

1. Station blackout,

2. Shutdown decay heat removal requirements,

3. Anticipated transient without scram (ATWS)

4. Safety implications of control systems and systems interactions in nuclear power plants, and

5. Hydrogen control measures and effects of hydrogen burn on safety equipment,

and an increased capability in the NRC emergency response area.

The objective of the proposed Severe Accident Sequence Analysis (SASA) program is to improve understanding of reactor accidents and of the human-machine interface during a broadened spectrum of accident sequences, including those within and beyond design basis limits. 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 plant to a safe shutdown condition, and the environment in which safety systems will be required to survive. Emphasis will also be placed on the recommendation for operator guidelines for severe accidents, minimum instrumentation to follow an accident, and assessment of the effects of the availability of equipment.

Extensive human factors engineering studies are underway related to accident prevention in the Division of Facility Operation (DFO) and in the industry programs. The Augmented Operator Capability (AOC) program personnel at LOFT have

have developed and installed the computer equipment and software necessary for displaying plant data using color CRT. AOC program results may be used by NRC in developing standards and eequirements for computer generated displays in nuclear power plants.

5.2.3 Interfaces With Other Elements

SASA will characterize sequences defined by risk assessment and provide a data base for assessment of IREP-developed methodologies. Rulemaking on plant-specific designs will define reactor accident sequences for analysis by SASA. Licensing and safety concerns generated by NRR Abnormal Transient Operating Guidelines (ATOGs) and other licensing reviews will serve to define SASA issues resulting from identified deficiencies in the symptom-oriented review. The SASA program also has interfaces with the following elements: (1) fission product release and transport, (2) behavior of damaged fuel, and (3) accident management.

In the interface with fission product release and transport, analysis is in progress on the sequence describing station blackout at Brown's Ferry Unit 1. The station blackout is assumed to persist beyond the point of battery exhaustion. Without DC power, cooling water can no longer be injected, potentially leading to core meltdown and containment failure. During this sequence, fission product transport paths exist, as identified by SASA, that by-pass the suppression pool. In the interface with damaged fuel, analysis of a hypothetical core meltdown accident initiated by loss of offsite power for the Zion 1 PWR has been recently completed. In the interface with accident management, SASA is involved in the development of a diagnostic algorithm. This process consists of sequence definition and the development of sequence signatures, accident/transient diagnostic methods, and diagnostic software.

5.2.4 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 Unresolved Safety Issue A-44.

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 useful in monitoring 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 in order to assess the adequacy of these procedures.

The completion of the sequence analysis to date has developed an 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 establish appropriate operator action for a variety of scenarios. The data base can also be used to evaluate the accuracy of PRA methodology which will likely play a role in the future process of plant licensing.

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 arising from thermal shock with cold repressurization, in the presence of a sensitized flaw. The analyses will assess mitigative actions undertaken 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. These include:

1. Depressurization capability in CE plants without Pilot Operated Relief Valves (PORVs),
2. Multiple steam generator tube ruptures,
3. B&W NSSS design features,
4. Emergency guideline development for ATWS events, and
5. High point vents.

The results and conclusions of SASA can be validated by the following events:

1. Advanced or full-scope training simulator.
2. Feeding back results into PRA
3. Licensee Event Report (LER).

The validation of the best-estimate computer codes describing accident behavior is currently outside the scope of SASA.

5.2.5 Plan of Work As a Function of Time

Issues in FY 1983 and beyond addressed by SASA include:

1. Support the development of a severe accident policy for nuclear reactors by addressing system functional requirements to assess prevention and mitigation of a core melt accident in severe accidents involving multiple

system failures; evaluate equipment and system survivability in severe accident environments; and evaluate the impact of proposed plant features on severe accident sequences in which they are not primarily intended to function.

2. Support the resolution of Unresolved Safety Issue A-9, Anticipated Transient Without Scram (ATWS). (This will be accomplished by assessing the effectiveness of potential procedural or plant design changes in ensuring the acceptability of the consequences of an ATWS.)

3. Provide analyses addressing the adequacy of assumptions concerning radionuclide transport and source term models.

4. Provide development and evaluations of current or future guidelines for plant emergency operating procedures to assess their usefulness for mitigating and preventing severe accidents.

5. Address plant behavior under complex transients coupled with multiple failures to define the plant behavior and to define the human machine interface required to prevent and mitigate severe accidents.

6. Provide analyses addressing general licensing issues such as depressurization of PWRs without power-operated relief valves, structural integrity of reactor containments, and cold repressurization of aging plants. These analyses will address the consequences of such events.

The schedule is shown in Figure 5.2.

	FY 1982	FY 1983	FY 1984	FY 1985
Evaluation of Procedural and Plant Configuration Changes Supporting Severe Accident Rule Making	Define operator actions in preventing or mitigating the accidents		Documentation	Complete Rule Formulation 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
Definition of Radionuclide Transport characteristics			Evaluate rate of FP movement in the plant. Estimate FP inventories and transport rates	Completion Date Dependent Upon Resolution of Degree of Conservatism of Current Models
Evaluation of Plant Abnormal and Emergency Operating Procedures	* Examples listed In Section III D.. Work to address analyses needs in support of operator guidelines i) as requested by NRR and ii) as required within SASA			
Resolution of Licensing Concerns		Analyze plant grouping for e.g. ANO-1	Documentation	
Identification and Implementation of Operator Information Needs		Identify key parameters that characterize a sequence. Relate symptoms of accident sequences to emergency procedures	Develop Complete Accident signatures	Complete developing an algorithm Documentation
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 plant enviornments, evaluate procedure and plant design effectiveness	Documentation

FP = Fission product

Figure 5.2 Severe Accident Sequence Analysis

5.3 Accident Management

5.3.1 Element Description

The objective of this program is to develop integrated strategies that combine elements of plant design and operating configuration with operator guidelines and procedures to optimize the capabilities to prevent, to arrest the progress of, or to mitigate the consequences of potentially severe accidents. To achieve this goal, we need to improve our understanding of reactor accidents and of the man-machine interface (MMI) during a broad spectrum of accident sequences both within and beyond the design basis. Data from the human factors research program will be used to establish MMI requirements for accident management and human performance research products will provide the technical inputs to this program. This task will also define technical constraints for human factors research by addressing the need for new plant features to be automatically or manually actuated. Identified equipment and operator guidelines for handling severe accidents will be evaluated for their risk reduction benefit using improved PRA methods.

This element will integrate analysis and experiments to provide the technical basis for decisions on changes in regulatory requirements on system design criteria and operating guidelines or to confirm the adequacy of established requirements. These regulatory requirements could relate to safety features, instrumentation, or administrative controls on operation.

5.3.2 Technical Issues Resolved by This Element

Technical issues to be addressed in these elements relate to instrumentation, plant design, operator guidelines and training, and the physical phenomena that must be understood to make the decisions necessary for accident management.

Accident management will be considered in a broad context that includes preventing accidents, arresting the course of an accident, and mitigating the consequences of an accident. Some specific issues include:

1. The identification of design or configuration changes that would improve the capability of preventing initial failures from leading to serious accidents, or improve the ability to limit the progression of damage, or mitigate the consequences of severe damage.

2. The improvement of man-machine interfaces to bring the operator's perception of the plant state closer to reality.

3. The identification of instrumentation to improve the information available to the operator for making decisions.

4. The improved guidelines for operator response during severe accidents.

5. To identify criteria, practices and data base needed to minimize the effects of human error in design, operations and maintenance.

6. To identify the means to assure containment integrity after the loss of the core and pressure vessel and thus limit the consequences of the public.

7. To establish the nature and extent of regulatory involvement in the development, review, approval and implementation of plant procedures, and review regulations concerning emergency response.

5.3.3 Key Interfaces With Other Elements

This element is closely tied to Elements 5.1, 5.2, 5.4, 5.5, 5.10, 5.11 and 5.13 (See Figure 4-1). The prevention aspects of accident management depend upon a combination of accident likelihood reevaluation and severe accident sequence analysis to identify design changes such as pump cooling circuits, auxilliary power circuity, etc., plant configurations, or administration controls that can reduce the likelihood of an accident and hence reduce the need for recovery procedures or mitigation. In the event of system failures, the SASA program provides the basis to evaluate recovery procedures and guidelines. However, experimental research on the behavior of damaged fuel and hydrogen generation and control are essential parts of formulating recovery and control guidelines; for without a firm phenomenological base, the conclusions drawn from SASA studies will have an unacceptable degree of uncertainty in assessing just how to bring the plant to a safe shutdown from a severely damaged state. In the area of human engineering there are two principal items: i) the human factors engineering program and ii) the human reliability program. Both support MMI. The former identifies design criteria and performance effects, operator role definition, automated systems interaction and personnel training requirements. The latter provides human error data, data banks and assessment methods to support probablistic risk assessment.

5.3.4 Background and Status

Probabilistic risk assessments have continued to evolve since the two reactors were analyzed in the Reactor Safety Study (WASH-1400). Since then, risk assessments have been done under the Reactor Safety Study Methodology Applications Program (RSSMAP) and Interim Reliability Evaluation Program (IREP) as well as through industry studies. These studies have made a significant contribution to identifying the dominant contributors to risk. The SASA studies have built upon these studies to examine in detail dominant accident sequences such as small-break LOCA, large-break LOCA, interfacing systems LOCA, feedwater transients and loss of AC power. These studies have provided valuable insight into the response of specific plants to these scenarios and how operator actions and/or the restorations of systems during the course of an accident can change the scenario and its consequences. The systematic use of these methods to develop guidelines is beginning and will be further developed under this element.

Human performance is recognized to be a significant and 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. Human Factors Engineering Research will generate information, data, methods and standards relevant to evaluating the MMI of nuclear facilities. Within two years, 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.

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.

Human Reliability research will develop and verify models of human performance, and quantitatively assess the human contribution to risk. Within two years this program will generate methodologies, guidelines, and a data base suitable for constructing models of human performance to assess human reliability.

5.3.5 Plan of Work As A Function of Time

This element ties together several other research elements. Within 2 years, there will be sufficient input from the other elements to provide a Phase 1 Accident Management Report. This includes:

i) Identification of operator error rates.

ii) Establishment of MMI requirements for accident management.

iii) Development of operator procedures for recovery from potentially degraded cooling accidents.

iv) Assessment of response of containment and ESF's to severe accident environment.

v) Development of operating procedures for assuring containment integrity during severe accidents.

Within 4 years, the supporting analysis and experiments as indicated above will have developed substantial additional data allowing the Phase 1 report to be updated and published in final form in order to support regulatory requirements relating to safety features, instrumentation, and operating guidelines.

Figure 5.3 shows this integration of elements in the final product.

ACCIDENT MANAGEMENT

	FY 82	FY 83	FY 84
Analysis of MMI	Identify Operator Error rates, perform evaluation of human reliability assessment	Develop Reg. Guidelines operating Procedures, for Safety Equipment, Validate Human Performance Models	Update work and models. Validation of MMI Requirements
		Establish MMI Requirements For Accident Management	
Accident Management Prevention	Assess Results of PBF Phase I, ACRR Coolability SCDAP MOD1 Analysis	Develop Technical Basis for guidelines for potential recovery from hypothetical degrade cooling accidents	Assess results for further testing in PBF and ACRR. Exercise improved SCDAP in reference to degraded cooling recovering procedures
Accident Management for Mitigation	Assess Response of Containment and ESF's to Severe Accident Environments.	Develop proposed procedures for assuring containment integrity during Severe Accidents. Assess need for additional equipment for operator to manage or mitigate accident.	Complete assessment of procedures and additional equipment requirements.

Figure 5.3 Accident Management

5.4 Behavior of Damaged Fuel

5.4.1 Element Description

The accident at TMI-2 raised many questions concerning the behavior of severely damaged LWR reactor cores. Many of these questions were formally addressed by such groups as the President's (Kemeny) Commission, the Rogovin Special Inquiry Group, the NRC Task Force on TMI-2, the Nuclear Safety Oversight Committee (NSOC), and the Advisory Committee on Reactor Safeguards (ACRS). A few of the more important questions indicated implicitly or explicitly that research should be started to:

1. Determine the general behavior of severely damaged fuel by studying its behavior in the 2200°F to 4000°F temperature range that appears to have been imposed on the TMI-2 reactor core,

2. Determine the actual hydrogen release and transport kinetics and, therefore, appropriate hydrogen mitigation features to be required,

3. Determine kinetics of fission product release and transport and their consequences,

4. Determine the consequences of hot core interactions with cooling water and their effect on reactor vessel and containment integrity,

5. Determine the coolability limits and cooling requirements of such cores at various stages of degradation,

6. Apply the newly acquired knowledge to significantly improve the calculation of perceived risk using the methodology developed in the Reactor Safety Study (WASH-1400), and

7. Apply the new information to support and confirm the NRC policy on regulatory requirements for severe accidents.

Accordingly, a four-part, integrated research program was initiated under this element that included (1) integral in-pile tests in the Power Burst Facility (PBF) and the NRU Reactor at Chalk River, Canada, (2) participation in the TMI-2 core examination, (3) separate effects experiments both in and out of reactor, and (4) the development of an integrated Severe Core Damage Analysis Package (SCDAP) to integrate and make usable the results of the program.

The results of this research program will be incorporated into accident-analysis systems codes and be used to provide guidelines for refinements to system design, operating procedures for improved accident management, and technical bases for improved PRA.

5.4.2 Technical Issues Resolved by This Element

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 radiogically 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.

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.

The technical objective of the Severe Fuel Damage program is to develop a data base and models for the range conditions covered in severe accidents predicting the following: the coolability (dryout limits) of the damaged fuel by reflooding; the rate of generation of hydrogen from the fuel cladding and reactor structure as they interact with water and steam; and the magnitude and rate of release of fission products. In addition, we seek 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 seek to determine a correlation, if any, between reflood rate and core uncovery time that minimizes further fuel damage from quenching.

The four-part integrated program of research (conducted in parallel) is planned to provide the needed information base of data and verified models. The first part consists of integral, multi-effects, in-pile tests in the PBF to provide early scoping data on governing phenomena, and later, in PBF and NRU, 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 ACRR and in the laboratory, to furnish a more specific data base for model development. An analysis package is the third part of the integrated program, including development of severe fuel damage models from the experimental data base and their integration into the severe fuel damage code, SCDAP. The fourth part is the benchmark data base to be obtained from the TMI-2 core examination.

The ultimate benefits of this program to the NRC will include:

1. An understanding of SFD phenomena that are important in determining the performance requirements for engineered safety features for in-vessel termination of the accident,

2. A base of technical support for the policymaking process for severe accidents, and

3. Guidelines for the acceptance criteria used by the staff in reviewing documentation submitted by licensees.

An additional, more general benefit from the program will be to provide the necessary SFD data base and models for more accurate calculations of the true risk to the public from a given accident scenario. Detailed knowledge of degraded core behavior will allow risk analysts to include risk-reduction computations resulting from recovery of previously failed safety systems and/or operator actions during a severe accident sequence. Also, it may be determined that the state of the core after a given sequence may be considerably more benign than that currently assumed because of a lack of available data on core degradation processes. It is expected that the results of this program may increase public confidence in the safety of nuclear reactors.

These benefits accrue from a program of limited scope, since the key data are the correlations between damage state, coolant flow characteristics, hydrogen generation rate, and fission product release. The basic assumption is that a categorization of damage among five or so states is sufficient for this purpose. Should that assumption turn out to be basically untenable, then the scope and objective of the program would need to be reviewed in depth.

5.4.3 Key Interfaces with Other Elements

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

1. Hydrogen Generation and Control Program - Data from the PBF test program and the resulting SCDAP assessment will provide this program with an accurate time-dependent hydrogen release rate from the core.

2. Severe Accident Sequence Analysis (SASA) - It is clear that for the SASA effort 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 probabilistic risk assessment 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.

   Although it can be argued that most plant responses are either well known or calculable from first principles, the responses of the core and fuel behavior as well as fission product release and transport in severe accident scenarios constitute notable and crucial exceptions. There are at present no data which can credibly be used to predict the extent of severe core degradation, fission product behavior and responses, hydrogen generation, core support structure attack, and large melt behavior. If it is remembered that the purpose of nuclear reactor safety is to minimize exposure of the public to fission product radiation, it is clear that core degradation and meltdown accidents that inherently pose the greatest ranges of releases must be better understood for the SASA program to be complete. The Behavior of Damaged Fuel program element is an essential part 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 plant to a safe shutdown from a severely damaged state.

3. Risk Codes - As the SASA program document explains, insights into issues will be gained "by applying best-estimate state-of-the-art codes such as RELAP, TRAC, MARCH-CORRAL, and CRAC...." In addition to the MARCH code, a new risk code (MELCOR) is being developed by the Office of Nuclear Regulatory Research. These latter codes are the only codes that address the physical behavior of the degraded core. Yet, the MARCH code was never intended as a device to predict details of degraded core behavior. It is being used for this purpose since no other codes exist and since a reliable data base is not available as an alternative.

To test MARCH code capabilities and limitations in the context of these applications, NRC recently completed an assessment of MARCH. Results of this assessment were that the code was limited or deficient due to fundamental phenomenological uncertainties in accident modeling and to modeling simplifications. The report notes significantly that there is difficulty in validating such models due to an inadequate phenomenological data base. The new code, MELCOR, will suffer from the same lack of sufficient verification information until core behavior data are supplied by new research efforts described herein.

4. Fission Product Release and Transport Program - The PBF test program will provide important integral data on the release of fission products and aerosols during severe core damage events and under incipient core melt accident conditions. The data will be principally of two kinds. Direct measurements of the amounts, types, and timing of the release of fission products and non-radioactive aerosols will be made during the tests as a function of fuel rod temperature, fuel rod oxidation state, and amount of fuel/cladding melting. 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 ones that are very difficult to investigate and quantify in out-of-pile separate effects tests.

Because of limitations involved in heating larger bundles of fuel out-of-pile, these larger scale (multi-rod) tests must be conducted in-pile. As a result of the larger bundle sizes, temperature gradients will exist across the test section. These temperature gradients will assure that a prototypic mixture of relatively volatile fission products and less volatile fission products and structural material will be coincidentally released into the steam/H_2 atmosphere of the bundle and hence the important chemical and physical interactions between all released species will be present. These include deposition of more volatile fission products on less volatile and previously condensed aerosol materials and chemical reactions between fission products and the various constituents of their environment (i.e., steam, H_2, other fission products, structural materials, and surfaces).

5. Accident Management Program - The data and analyses developed in the SDF program will be applicable to accident management planning and execution. This program will furnish the information required to assess the state of the core and its coolability limit and, therefore, the proper management of such an accident, including guidelines for refinements in system design, operating procedures, and possible additional instrumentation requirements.

6. Accident Consequence and Risk Evaluation Program - Since consequence and risk calculations depend on assumptions of the core state under various conditions, the data and analyses developed in the SFD program are necessary for accurate risk and accident consequence calculations. Data and analyses will be applied directly to further development of the MARCH code and to development of the new risk code, MELCOR.

5.4.4 Background and Status

Following a review of the events surrounding the accident at TMI-2, the NRC Special Inquiry Group (SIG), headed by Rogovin, and the President's Commission on TMI, headed by Kemeny, published findings and recommendations regarding severe accidents. It is clear from these two independent evaluations of the TMI-2 accident that the bases for licensing nuclear power plants can no longer be restricted to the design basis accidents used in the past, but must be expanded to include consideration of more severe accidents. According to the SIG report: "Modification is definitely needed in the current philosophy that there are some accidents so unlikely that reactor designs need not provide for mitigating their consequences." Further, "Reconsideration of the required 'design basis' for nuclear power plants should be initiated immediately." In addition, the President's Commission stated, "We urge strongly that research be carried out promptly to identify and analyze the possible consequences of accidents leading to severe core damage. Such knowledge is essential for coping with results of future accidents. It may also indicate weaknesses in present designs, whose correction would be important for the prevention of severe accidents." In addition to this need for research to better understand and predict the course and consequences of a degraded core event, several specific issues were raised by the SIG and the President's Commission; among them being: (1) instrumentation to diagnose plant conditions and (2) accident management and consequence mitigation. In the following paragraphs, these issues are discussed in the context of their relationship to information needs of the program on the Behavior of Damaged Fuel.

With regard to instrumentation, the SIG reports noted that "critical information was lost" because of inadequate instrumentation at Three Mile Island. Both the SIG and the President's Commission recommended that monitoring instruments and recording equipment should be qualified for the full range of accident conditions so that critical plant information would be available to the operators. Several of these recommendations were addressed by Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and by ANSI/ANS-4.5, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors." Although some instrumentation requirements are specified by the guide and standard, there are presently no bases for interpreting the state of a reactor core from the information available from the instrumentation.

Accident management and consequence mitigation were identified in Recommendations D.1 and D.2, respectively, of the President's Commission. These are: (1) "equipment should be reviewed...to help them (operators) prevent accidents and cope with accidents when they occur;" and (2) "Equipment design... should be reviewed from the point of view of mitigating the consequences of accidents." The equipment available to an operator would include engineered safety features (ESFs). To respond to these recommendations, it is necessary to review and establish the need for possible refinements in system design requirements. Such refinements cannot be established without experimental research to define the magnitudes of fission product and hydrogen releases and the conditions under which a degraded core is coolable.

Upon consideration of the above issues, it should be clear that research which quantitatively addresses the consequences of severe accidents is needed. Both the President's Commission and the SIG made recommendations to this effect. Specifically, the President's Commission recommended "that continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of melt-down," and further, "that as a part of the formal safety assurance program, every accident or every new abnormal event be carefully screened, and where appropriate be rigorously investigated, to assess its implications for the existing system design, computer models of the system...management, and regulatory requirements." The SIG recommended that the design basis for nuclear power plants should be reconsidered and that one area of review must be "the magnitude of the accident, including but not limited to the severity of fuel damage and core disruption, the magnitude of release of radioactive material, and the magnitude of hydrogen generation."

5.4.5 Plan of Work As a Function of Time

1. Integral In-Pile Tests in PBF - The major part of the program of integral in-pile tests is the Severe Fuel Damage (SFD) series in PBF. Phase 1 of the program, which is now under way, will provide integral scoping data in the temperature range 2200-2400K. 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 post-test 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.

The consequences of a severe accident are dependent upon the sequence of damage in the accident, and there are many paths that degraded core accidents can take. The PBF severe fuel damage test program is designed to map the response surface defined by the damage phenomena produced by varying certain key experimental parameters, i.e., heating rate, cooling rate, steam flow, peak temperature, fuel rod burnup, bundle size, and low melting control and structural materials. This mapping of the damage phenomena will be accomplished by attempts to bound the range of the effects of these various parameters. Depending on the parameter, the damage produced in these tests can range from fuel rods with cladding totally oxidized to ZrO_2 and geometry altered only by localized 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. The amount and timing of the release of fission products and the generation and transport of hydrogen are also expected to vary with the parameters mentioned above. The data from these highly instrumented and well-controlled PBF severe fuel damage tests will be combined with the data from special separate effects experiments and the examination of the TMI-2 core to form a complete picture of the behavior of a large LWR during a degraded core cooling event.

Current Phase 1 plans call for five 32-rod tests to be conducted: two at slow heating rates less than 0.5°C/second (to fully oxidize the cladding and therefore preclude the formation of liquefied fuel), two at faster heating rates of about 4°C/second, and one approximating the estimated TMI-2 conditions. One of each of the slow- and fast-rate heating experiments will be cooled slowly from maximum temperatures of about 2175K (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 with the resultant production of core debris. The detailed experimental conditions of the fifth test have not yet been specified. These tests will also verify the adequacy of the designs of the test train and the shroud that is required to contain the liquefied fuel. Moreover, the tests will produce the debris to be used for determining the expected size-ranges, compositions, permeability, and coolability of severely damaged fuel. These characteristics will be used in guiding and planning separate-effects debris coolability experiments in the ACRR. Finally, fission product and hydrogen release and transport data will be obtained from all tests and used to verify and assess SCDAP models as well as current source term analysis methodology.

There has been preliminary planning for a second phase of integral SFD testing in the PBF to explore the effects on core behavior of high-burnup fuel, control rod materials, and high temperature accident conditions. This series may also include experiments at higher temperatures and larger bundle size to explore the effects of using a decay-heat source built up by a 1-week irradiation of previously irradiated fuel rather than fission-simulation of decay heat. The larger test-bundle size would also be used to determine the effects of large arrays on blockage distribution and the effects of prototypic amounts of absorber materials. These latter tests will require a modification of the PBF to incorporate a larger test loop that would accommodate up to full-diameter 17 x 17 PWR fuel bundles.

2. Integral In-Pile Tests in the NRU Test Reactor - Subsequently, integral SFD data will be available from tests in the NRU reactor at Chalk River, Canada for 21-rod, full-length fuel bundles. The data will supplement the 3-foot PBF results and permit determination of the scaling effect of a 12-foot axial length. These tests are planned to cover a wider range of accident conditions than the PBF Phase 1 tests, including higher pressure and simulation of both PWR and BWR conditions.

3. Separate-Effects Experiments on Coolability of Debris in the ACRR - The second major part of the research on severe fuel damage is a program of supplementary 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 conditions (if any) for which simple quench is not sufficient to cool the debris and terminate the accident, and to determine the cooling (pressure and flow rate) necessary for coolability under these conditions. The dryout coolability limit is reached when liquid cannot penetrate to all points in the debris bed because of the outflowing vapor. Considerable data and rather sophisticated analytical models of the quasi-static dryout coolability limits of debris beds of decay-heated particulate fuel debris under liquid pools have been developed in fast-reactor safety research. Beginning in late FY 1982, a series of seven LWR-specific core-debris coolability experiments will be started in 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.

4. Separate-Effects Experiments on Fuel Debris Formation and Relocation in the ACRR - A program of separate-effects phenomenological experiments has also been started in 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 diagnostic data continuously in time for high-probability unprotected 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 SCDAP. These separate-effects experiments effectively supplement the larger-scale integral Phase 1 SFD tests in PBF, and they will substantially broaden the data base for model development.

Laboratory separate-effects experiments are planned (depending on the scope of German research) to determine the thermodynamics and kinetics of the reactions between UO_2, Zircaloy, and steam. 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.

5. Melt Progression to Reactor-Vessel Failure - Substantive analysis will be performed on the progression of core melt for the high-probability unrecovered severe accident sequences up to the point of reactor-vessel failure. The results of this analysis will guide planning for future experiments in this area, and will provide input to improvement of the MARCH code and development of the new MELCOR code. Results of this melt-progression analysis should significaltly reduce the very large uncertainties regarding the initial conditions used in assessing the core-melt threat to the containment and uncertainties in PRA.

6. Development of SCDAP - The Severe Core Damage Analysis Package (SCDAP) computer code will predict the physical state of a light water reactor core as a function of time during various degraded core cooling accidents significantly more severe than the present design basis accidents.

   The SCDAP computer code will provide a capability for analyzing fuel and core component behavior for severe accidents in an LWR core including UO_2 melt progression and ultimately reactor vessel failure. When completed, SCDAP will calculate component temperatures as a function of time and axial position; fuel rod deformation, including clad ballooning and collapse; the amount and chemical forms of released fission products; oxidation of core components and the amount and axial distribution of the hydrogen generated and released; the amount and location of liquefied and resolidified material; the mass of rubble debris and the characteristics and spatial distribution of this debris; and an estimate of coolant flow blockage.

The SCDAP code will be a key product of the NRC's integrated experimental and analytical SFD research program. SCDAP will encompass much that is known and understood about the physical and chemical states of a reactor core at various points in time during a severe accident. LWR reactor utilities may use SCDAP for analysis of proposed accident management procedures and refinements in the design of engineered safety features, core instrumentation, and information display systems. The NRC and its contractors will use SCDAP to evaluate licensing analyses and probabilistic risk assessment.

The modeling approach being used for SCDAP makes maximum use of existing computer models developed and assessed for LWR design-base events and of LMFBR safety on core-debris coolability limits.

The initial version of SCDAP will be developed without any data from the planned SFD experiments. However, many of the models will be preliminary in nature and must be assessed through comparisons with experimental data. The Phase 1 PBF Severe Fuel Damage Test Series and the initial ACRR separate-effects experiments will provide the primary data base for this purpose. The SCDAP development plan is, therefore, closely coupled with the Phase 1 testing program.

The first two Phase 1 tests will provide data for assessment of the initial version of SCDAP. The first test, SFD-ST, will provide information on several unresolved questions about in-reactor chemistry at high temperatures. (Chemistry is important because it provides an important heat source and affects subsequent fuel behavior.) Out-of-pile correlations for Zircaloy oxidation kinetics above 1850K that are based on the data of Urbanic and Hagen will be checked. Predicted oxidation kinetics in steam or steam-starved environments will be compared with measured oxide thicknesses and hydrogen release rates to see if the oxidation heating and hydrogen release rates are correctly modeled by SCDAP. The second test, SFD-1, will be used to clarify the kinetics of formation of liquefied fuel under realistic temperature distributions and its interaction with reactor materials such as Inconel grid spacers which may join liquefied U-O-Zr and affect the melt's properties significantly. Post-irradiation examination (PIE) of the fuel rods will provide compositions and configurations of material that [illegible] distributed and solidified. This information will be compared with SCDAP predictions to check if the limited out-of-pile ternary phase diagrams now available are sufficiently accurate for modeling in-pile liquefaction and solidification.

The third and fourth Phase 1 tests, SFD-2 and SFD-3, will be used to assess the second version of SCDAP. The initial versions of SCDAP will use correlations based on PBF Power-Cooling-Mismatch and Reactivity Insertion Accident tests to determine debris size distribution. SFD-2 will provide initial data for size distribution under conditions that more closely match severe accident conditions. The test will also provide needed information about the composition of debris of different sizes, size distribution, packing densities, fragmentation criteria, and size and location of flow channels. The fourth test, SFD-3, will include both liquefaction and fragmentation, and it will confirm the models of these processes developed and assessed using the two previous tests where the processes were studied separately.

Even though the Phase 1 tests will not include fuel with extensive burnup, PIE will provide preliminary information regarding the release, transport, and relocation of fission products. PIE of the test fuel will determine the redistribution of fission products within the test assembly. As data on fission product behavior are gathered from the Phase 1 tests, the data will continually be compared with SCDAP predictions to confirm and refine the models.

The separate-effects phenomenological experiments in ACRR will also provide data and models for the development of SCDAP. Information will be obtained on the process of fuel debris formation and relocation in flowing steam, debris formation by quench at different points in severe-accident sequences, and on debris coolability limits.

Development and assessment of the final version of SCDAP will depend on the last PBF Phase 1 test, the PBF Phase 2 tests, the ACRR separate-effects experiments, and on the TMI-2 core examination program. SFD-4 will provide a data base on fuel behavior for typical (TMI-2) heating rates for comparison with predictions by SCDAP. The Phase 2 tests will include fuel with significant prior irradiation, control rods, and tests to higher temperatures. The irradiated fuel tests are needed to confirm and extend the preliminary fission product chemistry data obtained during the Phase 1 tests. These tests are also needed to improve SCDAP modeling of fission product release rates for elements like cesium, iodine, tellurium, and tin whose release rates are not well characterized, and for in-pile fuel stresses and cracking patterns typical of severe accident conditions. Finally, use of irradiated rods allows direct comparisons of temperature and pressure as a function of time for a bundle powered by decay heat. This will be the first time such a comparison is possible with a well-instrumented bundle and will be an excellent assessment for SCDAP.

Some of the Phase 2 tests will include control rod materials in the test bundles which, because of their low melting and boiling points, may have significant effects upon damaged fuel behavior. The Phase 2 tests will provide significant data relevant to control rod behavior, and they will serve as scoping tests to indicate any major phenomena that may have been missed in SCDAP modeling. Additional data on the effects of control rod materials will come from the ACRR separate-effects experiments.

7. TMI Core Examination - The TMI-2 core examination constitutes a unique and important resource on the characteristics of severely damaged fuel. Early recovery and adequate analysis are highly desirable to provide a benchmark for research on and understanding of the behavior of severely damaged fuel, including development of the SCDAP code. The current program of the Fuel Behavior Branch includes modest support for analysis of TMI-2 core debris, but does not, of course, address the cost of the TMI-2 recovery operation. It is a matter of some concern that because of uncertainties in the schedule for recovery of the TMI-2 core that these valuable data on the fuel behavior of reactor fuel in a severe accident may not be available until very late in the period covered by this plan.

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 tests, the early ACRR phenomenological separate-effects experiments, the TMI-2 core examination, and with models developed from these data, some of the additional programs will prove unnecessary. In any case, the SFD program requirements and program plans should be and will be reexamined periodically.

A summary matrix of questions to be answered by the SFD program and the sources of information needed to provide answers is shown in Table 5-1.

The following paragraphs provide a year-by-year description of the expected program results:

<u>In FY 1982</u> - The first version of the SCDAP code will be completed. The first PBF scoping test will be performed and analyzed. The first LWR core-debris coolability experiment will be conducted in the ACRR.

<u>In FY 1983</u> - Two additional Phase 1 PBF tests will be performed and analyzed. The second version of the SCDAP code will be issued containing updates resulting from the first three PBF tests. Three experiments on debris

formation in flowing steam and two debris coolability experiments will be conducted in ACRR. An assessment of the use of LMFBR debris-coolability models for LWR conditions will be made. Analyses of results of the NRU clad ballooning program will be completed and planning for follow-on tests will begin. Ex-pile laboratory tests on the thermodynamic and kinetic relationships between uranium, Zircaloy, and oxygen will be started.

In FY 1984 - Analyses and experiments will be completed on all Phase 1 SFD tests conducted in the PBF. Fission-product distributions, detailed analysis, and accident signatures will be reported. One or two Phase 2 PBF tests will be conducted if sufficient funding is available for planning in FY 1982 and FY 1983. The final advanced version of the SCDAP severe fuel damage code will be completed. The initial series of experiments on debris formation and relocation in flowing steam in the ACRR will be completed, and a new series on debris formation under reflood quenching will begin. Two debris-coolability experiments will be performed in ACRR. Two initial NRU follow-on tests will be completed to bridge the gap between clad ballooning data for large- and small-break LOCAs. Initial information will be available from the TMI-2 core examination.

In FY 1985 - Three Phase 2 SFD tests will be conducted in the PBF and the preliminary data analyses will be reported. Final data analyses on the Phase 1 PBF SFD tests will be completed and reported. Three SFD tests will be performed in the NRU reactor; data and analysis reports will be issued. The experiments in ACRR on debris formation and debris coolability will be completed. The SCDAP code will be assessed and kept on maintenance. Additional information from the TMI-2 core examination will be available for benchmarking SCDAP and its extensions to whole core analysis with other codes.

In FY 1986 - All SFD testing will be completed. Development of the phenomenological models from the results and of the integral SCDAP code will be completed. An assessment of the needs for higher temperature and larger-scale testing in PBF or NRU will be made at this time.

Input-Output Matrix for the Severe Fuel Damage Experiment Program*

To Answer These Questions: ⟶

We Need: ↓

Information, From ⟶	Experiments: Multi.-Ef.	Experiments: Sep. Ef.	1) For what range of severe accident conditions is the fuel debris coolable by slow reflood for in-vessel accident termination?	2) Beyond that range, what reflood velocity and pressure are necessary to cool the debris?	3) Can too rapid reflood fail the reactor vessel?	4) What are the amounts and rates of hydrogen generation?	5) What is the radiological source term out of the reactor vessel?	6) If debris cooling fails, what are the conditions of reactor vessel failure and debris entry into the containment?
1. Clad Ballooning, Burst, and Blockage	PBF-1, 2 NRU	ACRR Lab	X	X		X	X	
2. Oxidation (Hydrogen)	PBF-1, 2 NRU	ACRR Lab	X	X		X	X	
3. Fission Product Release and Attenuation	PBF-1, 2	Lab					X	
4. Fuel Debris Characterization	PBF-1, 2	ACRR	X	X	X			X
5. Fuel Debris Relocation, Blockage	PBF-1, 2 NRU	ACRR	X	X	X			X
6. Reflood Debris Characterization	PBF-1, 2 NRU	ACRR Lab	X	X	X	X		X
7. Rapid Steam Generation and Explosion	PBF-2 LMF	ACRR Lab			X	X		X
8. Damaged Bundle Coolability	PBF-2	TBD	X	X				
9. Fuel Debris Coolability	PBF-2	ACRR Lab	X	X				X
10. Post-Dry-Out Behavior		ACRR Lab						X
11. Melt Progression	TBD							X
12. Debris Characterization at Vessel Failure	TBD	TBD						X

* The data and models from these experiments are integrated into the SCDAP severe core damage code

Table 5-1

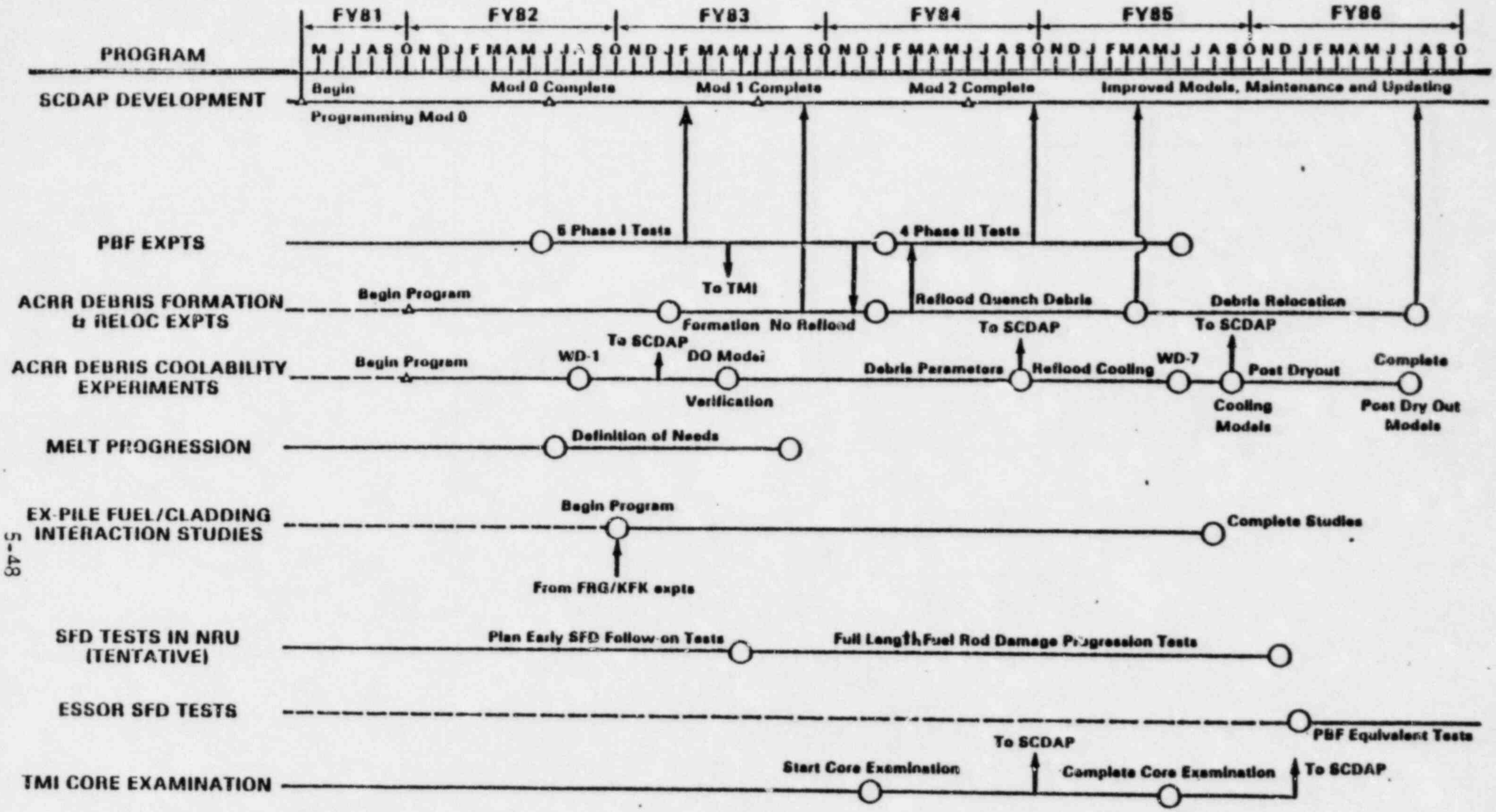


Figure 5.4 Behavior of Damaged Fuel

5.5 Hydrogen Generation and Control

5.5.1 Element Description

During an accident, or as the consequences of an accident, significant quantities of hydrogen can be generated in the reactor vessel from steam-zirconium and steam-steel reactions and in the containment building from molten core-concrete interactions. The burning of this hydrogen could (1) produce loads on containment that could exceed the ultimate strength of the building or (2) cause the failure of safety-related equipment that would affect the safe operation of the plant. This program is currently providing information and analytical models to quantify this threat and to assess the efficiency of mitigation systems proposed by near-term operating license applicants and of other possibly more efficient systems. This work also includes the development of an analytical model that will permit better understanding of hydrogen transport, mixing, and combustion phenomena.

5.5.2 Technical Issues Resolved by This Element

As a result of Commission requirements that plants be able to handle hydrogen releases, a number of prevention methods and mitigation schemes for hydrogen control have been proposed by utilities and vendors. An objective of the RES hydrogen program is to provide technical information on the adequacy and efficacy of these systems and also to assess the possible alternatives to these systems. In the review of accident scenarios where hydrogen is released, key technical questions arise as to the timing of the release, the amount of transport or mixing of the gases in containment, the potential for the occurrence of transition from deflagration to detonation, the pressures and temperatures from hydrogen combustion, and their effects on equipment survivability.

Major technical objectives are listed below:

1. Accident analysis calculations will be made with the MARCH code for 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 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 Fully Instrumented Test Series (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 at Sandia 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, VEGS 16-foot tank, and radiant heat facility to characterize the response of this equipment to hydrogen deflagration in the presence and absence of steam. The results of these tests will then be extrapolated to full size containment, using the methodology developed in the analytical part of the program currently being sponsored by NRR.

5.5.3 Key Interfaces With Other Elements*

1. Accident Management Program - Information developed from this program will be issued to assess and develop guidelines for handling hydrogen emergencies.

2. Containment Analysis - This program will provide improved burning models to be incorporated in CONTAIN and other containment analysis models.

3. Containment Failure Models - This program will provide the pressure and temperature loads placed on a containment building during a hydrogen burn. This information will be used in assessing the response of the containment to these loads.

4. Risk Code - Models developed in this program element will be incorporated (possibly as modified versions) into risk codes such as MARCH or MELCOR.

*The Electric Power Research Institute (EPRI) has a large research program on hydrogen that is complementary to the NRC effort in many areas and is providing direct information in some areas (e.g., mixing, large-scale effects) to the NRC program.

5.5.4 Background and Status

1. Hydrogen Behavior Program

In this program deflagration and detonation models (named HECTR and DETON respectively) are being developed to more accurately predict temperature and pressure histories in containment during and after a hydrogen combustion. The work on deflagrations will include the effects of CO_2, CO, and water fog evaporation. The detonation model will be able to predict Chapman-Jouguet pressures and temperatures characteristic of the transition through a detonation front, including increases after normal reflection. The current effort includes heat transfer mechanisms such as radiation, convection with or without condensation, conduction into surfaces, and the evaporation of sprays in the model. A first version of these codes should be available in FY 1982. The models will be improved and validated in time to provide the NRC with a technical assessment of interim deliberate ignition systems currently being proposed by several utilities.

A number of detonation calculations have been performed using the CSQ code for Sequoyah analysis (NUREG/CR-1762) and for the Zion study. The CSQ code will be used in conjunction with DETON to calculate the hydrogen combustion loads on various containments.

MARCH calculations have been performed for a number of scenarios in which hydrogen is released from the primary system. These analyses have been done for Zion (large dry PWR) and Sequoyah (ice condenser) and are currently planned for Grand Gulf (BWR MARK III). They have provided useful information to NRR in their assessment of the Sequoyah interim distributed ignition system and will likewise be used in the evaluation of the hydrogen control system for Grand Gulf.

A difficult area of analysis now being addressed deals with hydrogen transport. An attempt is being made to modify or develop a code to predict the concentrations of hydrogen, air, and steam in containment as functions of position and time for hypothetical LWR accidents. The German code RALOC is being assessed to determine its present and potential ability to handle transport and mixing analysis. Currently it is planned to test the code against a series of hydrogen mixing tests in the Containment Safety Test Facility at Hanford Engineering Development Laboratory (HEDL) and sponsored by EPRI. Additionally, two other codes, COBRA and SOLA-D, will be modified and tested against the HEDL experiments. The staff is currently considering the need for combining the combustion and transport models together in a single hydrogen code.

The hydrogen program also includes experimental projects directed at determining hydrogen deflagration and detonation limits in air and steam and noting how the location and strength of the ignition source affects those limits. The experiments will determine temperature and pressure profiles as a function of time, thus providing information needed to develop and validate analytical models. Additionally, tests are being planned to understand autoignition of hydrogen, particularly as it relates to hydrogen and steam jet releases, similar to what might occur from a pipe break.

A key area of study is the work on the transition from deflagration to detonation that can occur as a flame propagates from one chamber to another, through a concentration gradient, or accelerates around structures. Flame acceleration in the upper structure of an ice condenser containment was a concern that was raised relative to the Sequoyah distributed ignition system and subsequently in the McGuire hearings. (Flame acceleration occurs when a flame front bends around a structure and begins to break up, inducing turbulence and developing more surface area and causing the flame to burn faster with higher temperatures and pressures. These phenomena can cause lean mixtures to reach temperatures and pressures exceeding the theoretical adiabatic-isochoric limits.) Some engineering scale experiments are currently being planned to mock up the upper structure of Sequoyah to assess this effect. Additionally, contractors at McGill University are performing laboratory-scale tests on flame acceleration and the transition from deflagration to detonation. Analytical work in this area is somewhat behind the experimental effort. However, it is expected that as a better understanding of the phenomena involved is developed, model development will accelerate. It may be necessary to perform some large-scale testing on hydrogen combustion, particularly in the area of deflagration-to-detonation transition. An assessment of this need will be performed in FY 1982.

2. <u>Hydrogen Combustion, Mitigation, and Prevention Program</u>

Work in this area is directed towards an understanding and assessment of methods to control hydrogen combustion or at least to dampen its effects. The hydrogen combustion, mitigation, and prevention program was originally a part of the hydrogen behavior and control program but was separated in order to allow more emphasis to be placed in studying the design criteria and feasibility of proposed prevention and mitigation schemes.

Experimental facilities and tests being run under the hydrogen behavior and control program to characterize hydrogen deflagration and detonation also include the effects of water fogs and foams as a mitigative approach to controlling temperatures and pressures. A .05 percent volume fraction of suspended water droplets can reduce the temperature from a stoichiometric mixture of hydrogen and air from approximately 2700°K to less than 1100°K with a proportional reduction in pressure. The combination of hydrogen igniters and a water fog system appears to present a much higher level of protection in handling potential hydrogen combustion accidents for certain containments than either system alone.

Mitigation experiments on the effectiveness and feasibility of oxygen depletion, pre-inerting and postaccident inerting with carbon dioxide and halons are also planned. Other prevention and mitigation schemes, such as gas turbines and high capacity recombiners, will also be assessed. Although some of this work may continue into FY 1984, most should be completed by FY 1983.

An experimental investigation of the feasibility of a deliberate flaring technique in conjunction with a high point vent as a method for controlling hydrogen release during an LWR accident will also be performed. This assessment should be completed in FY 1983 with the need for further work to be determined at that time.

3. Hydrogen Burn Equipment Survival Program

The combustion of hydrogen in containment can lead to the failure of important safety equipment or equipment necessary for plant isolation during an accident. The H_2 burn equipment survival program was initiated in order to experimentally assess the effects of hydrogen combustion on equipment. This program also provides a data base in order to develop analytical models to assess equipment survivability. NRR is sponsoring the other half of this program, i.e., to develop analytical methods to calculate the effects of hydrogen combustion on equipment. The experimental facilities used in the hydrogen behavior and control program will be used to actually test equipment in hydrogen burning environments; this test will begin in FY 1982. The program is phased in two parts: (1) to provide information to NRR in the short term on the effects of hydrogen combustion equipment for near-term licensing decisions and (2) to develop reliable methods to predict the response and survivability of equipment. In addition to the current facilities available for testing, the possibility of using Sandia's Radiant Heat Transfer Facility to test larger components, if necessary, is being assessed. The test plan now covers multiple burns under deflagration conditions, and the need to perform tests under detonation conditions is also being considered.

5.5.5 Plan of Work As a Function of Time

The analyses of hydrogen accidents for the three specific plants (Zion, Sequoyah, and Grand Gulf) should be completed in FY 1982. Analysis for other specific plants or standardized plant designs will be performed as requested using the deflagration code HECTR and a validated hydrogen transport code. An assessment of H_2/air/steam deflagration limits will be completed in early FY 1983. An experimental assessment of flame acceleration effects (except for any large-proof test whose need remains to be assessed) and preliminary analytical models should be available by the end of FY 1983. An assessment of the effects of aerosols on hydrogen control systems will be performed in FY 1983. An experimental assessment of the effects of fogs and foams on hydrogen burns will be completed by the end of FY 1982 and an initial assessment of pre-inerting, O_2 depletion, and postaccident CO_2 inerting by the end of FY 1982 with a final evaluation in FY 1983.

The hydrogen burn equipment survival program will have completed a number of experiments on safety-related equipment during FY 1982 and during that same period will have developed some preliminary thermal models for assessing the effects of hydrogen burns on equipment. Experiments on equipment survival should be completed in FY 1984 and should result in standard methods for predicting equipment res½pmse tp hydrogen burn conditions. Figure 5.5 lists additional milestones for this program element.

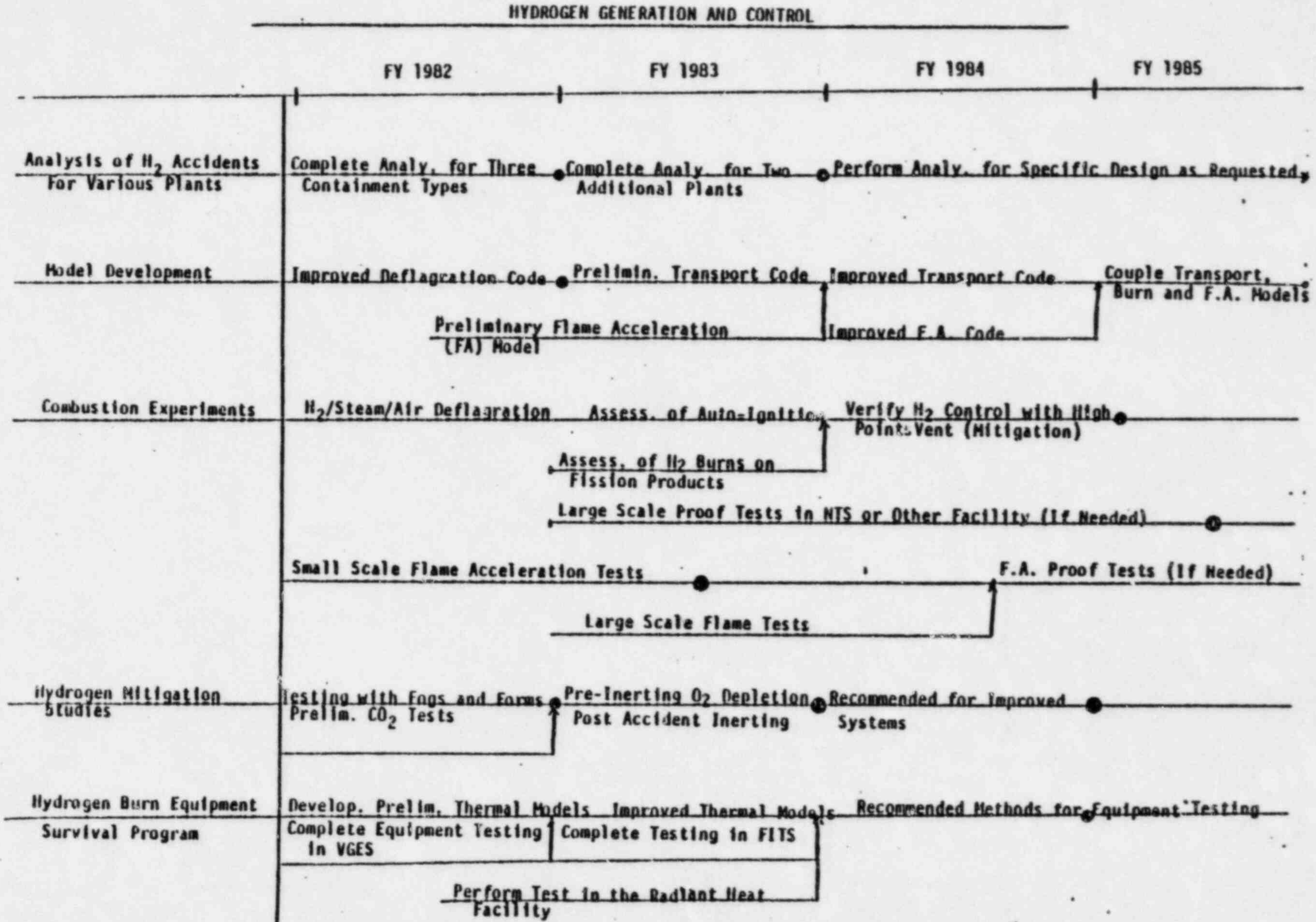


Figure 5.5 Hydrogen Generation and Control

5.6 Fuel-Structure Interaction

5.6.1 Element Description

This element addresses the interaction of fuel and other material from the primary system with the receiving component below the pressure vessel. After the fuel has escaped from the pressure vessel, the consideration of consequences of this release requires knowledge of interactions of the fuel with other material, i.e., with the basemat concrete, with water that is either present below the pressure vessel or introduced later, with concrete that is present with coolant, and with mitigation structures or devices.

Studies and experiments are conducted to evaluate the interactions with respect to penetration rates, heat generation and release, gas and aerosol release, and the rapid generation of steam with the possible formation of missiles. In turn, these quantities establish the loads on the containment for risk assessment.

5.6.2 Technical Issues Resolved by This Element

The technical issues include the interactions of core material or hot severely damaged fuel products 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 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, most probably by failure of the lower head, and also to generate missiles that would threaten the containment if there were no missile shield. Nonexplosive rapid steam generation during both in-vessel and ex-vessel melt/water interaction, the "steam-spike" problem, also has the potential to fail the reactor vessel and the containment directly. The characteristics of the products as formed by water reflood of very high temperature fuel are key elements in assessing the coolability of that mixture.

5.6.3 Key Interfaces With Other Elements

The key interfaces are with Containment Analysis, Fission Product Release and Transport, Containment Failure Mode, and Accident Management. The core-melt/concrete interaction experiments establish source term values for the Fission Product Release and Transport element. In addition, these experiments establish the parameters for the core/concrete interaction model of the CORCON code, which is a module of the CONTAIN code. Concrete penetration rates, heat generation and partition, and the steam generation and work conversion efficiencies are also included. CORCON also computes the generation rate of H_2, CO and aerosols as well as radiative heat loss from the pool surface. The project includes eventual treatment of overlying water, crust formation and hot solid debris on concrete. The CONTAIN code is a major tool for analytical investigation in the Accident Management element.

5.6.4 Background Status

The scope of this task includes small-scale scoping and phenomenological experiments of thermal, mechanical, and chemical interactions of fuel above the solidus temperature, and of high temperature core debris simulants with concrete, refactory, and sacrificial materials; large-scale scoping or model-verification tests; 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 fuel-mass-concrete interaction.

A large facility to study the interaction of molten fuel and structural material has been completed and the first large-scale test crucible prepared for an initial test in the second quarter of FY 1982. A second large-scale test is being assembled. A smaller facility for sustaining the heating of hot structural material and/or some material on basemat materials has been completed and the initial test has also been completed. The development of most instrumentation for the scope of testing noted above is nearing completion.

An extensive data base on the conversion of core-mass 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 in a range necessitating further research. This estimate may be revised in the light of larger-scale test results. Medium-scale FITS tests have been completed with thermite-generated corium melts dropping into water. 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 yet exist to construct a mechanistic model of the thermal detonation process that would have predictive capability.

5.6.5 Plan of Work As a Function of Time

A systematic study with large-scale fuel-mass interactions with concrete and retention materials has been initiated. The first test will be conducted in the second quarter of FY 1982. A second large-scale test will be conducted in FY 1982 and orders for additional test crucibles will be placed upon completion of a satisfactory test. The intermediate-scale testing of sustained heating on basemat concrete and core interdictive materials will proceed throughout FY 1983 and 1984. These tests will be instrumented to provide quantitative aerosol and gas emission (H_2, CO, CO_2) information as well as upward and downward heat flux and erosion rates. The first independent results are expected in FY 1985 from the KfK Beta facility using large thermite melts.

The interdictive materials to be considered are MgO bricks and castable ceramics (notably the high alumina cements). Core-coolant effects will be introduced in FY 1983-1984. Both the introduction of hot materials into coolant pools and the introduction of water onto the hot materials will be included. In the FY 1984-1985 period, tests for floating nuclear plants are expected to be included.

In addition, back-fit considerations of loose gravel beds (thoria) will be studied for sweepout and coolability. Thermite melts into wet and dry beds indicate negligible effect from the presence of water. Cornium simulants and pressurized discharges onto gravel beds will be evaluated.

Experiments will be initiated on nonexplosive rapid steam generation conditions (steam spikes) that might threaten the integrity of both the reactor vessel and the containment. Experiments in these areas will continue through FY 1985 and will then terminate unless important problems requiring further work become apparent.

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

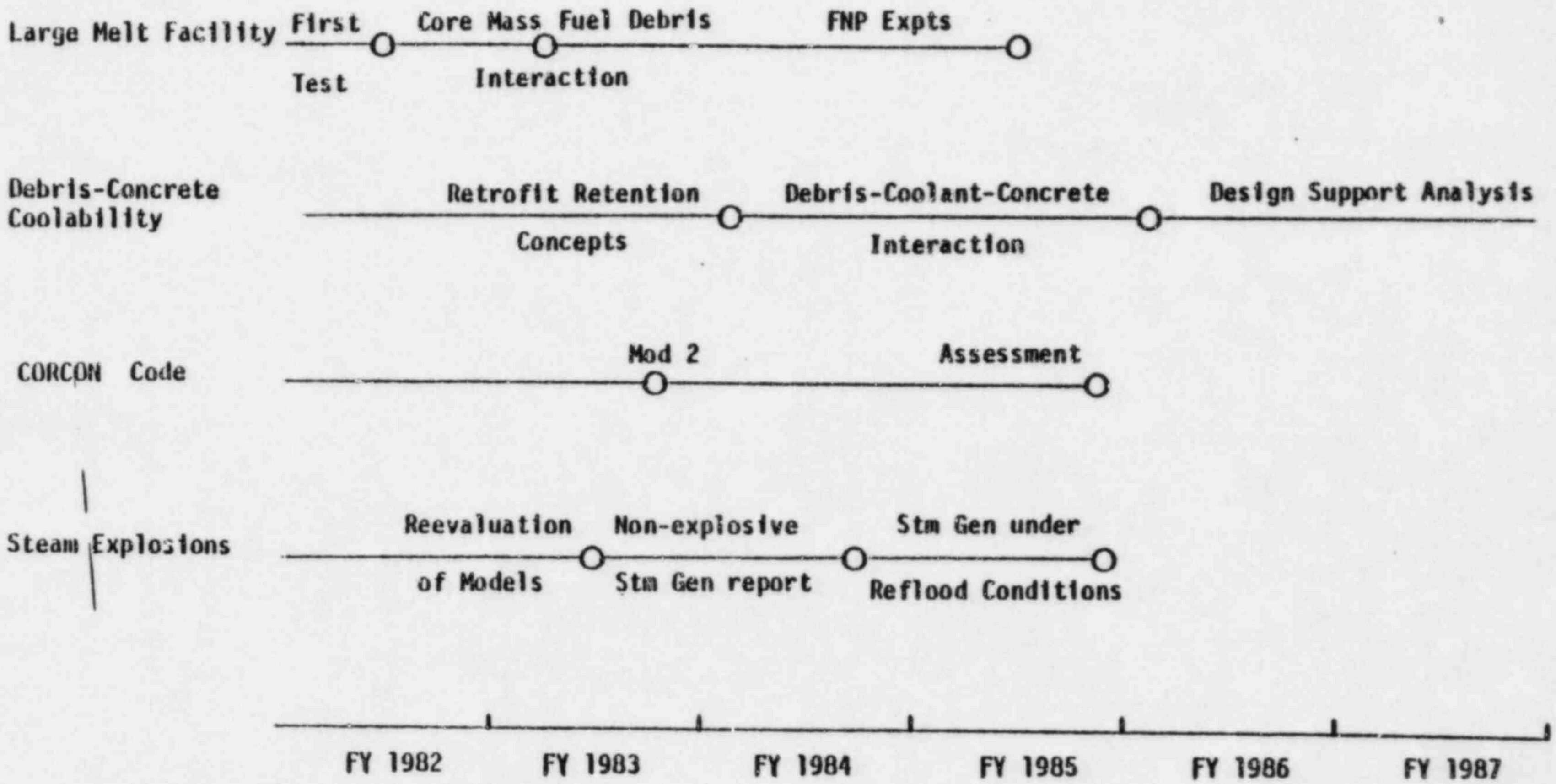


Figure 5.6 Fuel-Structure Interaction

5.7 Containment Analyses

5.7.1 Element Description

This program element is intended to provide analytical tools and phenomenological models to assess the loads that threaten the containment during a severe accident. If the primary system has failed and extensive fuel damage has occurred, the containment will be threatened by steam pressure, fission product and aerosol formation and potentially hydrogen burning, and, in extreme accidents, a threat to the basemat for molten fuel.

To do these analyses, a generalized systems code, CONTAIN, together with the necessary submodules, is being developed that is sufficiently generic and highly flexible so as to be capable of handling both BWRs and PWRs with the variety of different containment designs that exist.

5.7.2 Technical Issues Resolved by This Element

The major technical issues to be addressed by the element are the quantification of loads that threaten the containment and characterization of the radiological source term that would threaten the public should the containment leak or fail. Some issues related to containment threat are:

1. Overpressurization due to steam production,
2. Overpressure due to hydrogen combustion,
3. Potential for missile production,
4. Gas production from molten-fuel/concrete reactions,
5. Basemat penetration from molten fuel,

6. Coolability of core debris in reactor cavity, and

7. Performance of containment engineered safety features.

The foregoing list is not intended to be comprehensive, the program includes an ongoing task to ensure the completeness of the potential threats to be considered.

Technical issues related to source terms are:

1. Fission product chemical behavior, and

2. Characterization of aerosols and gases that can transport fission products.

5

5.7.3 Key Interfaces with Other Elements

Because of the wide spectrum of phenomena involved, the CONTAIN program is related to a large number of other ex-vessel accident research issues, i.e., experimental projects aimed at the empirical determination of core/coolant interactions with concrete, the generation and transport of aerosols, the behavior of high-temperature debris pools, and the detailed nuclear decay properties of fission products. The containment-analysis program is thus closely interactive with many of the efforts being conducted within program elements 5.5, 5.6, 5.8, and 5.9. Also, the computational models in CONTAIN involve a variety of other analytical research projects, the purpose of which is to provide reliable predictive calculations of the various processes.

5.7.4 Background and Status

The CONTAIN code project was initiated under LMFBR-research auspices, but the generic nature of the problem was early recognized, so that the basic code structure was designed to be independent of reactor type. One of the first problems to which CONTAIN was applied related to the Zion-Indian Point study. The importance of considering water-vapor condensation on aerosols was studied. At present, the CONTAIN code is operational on the Sandia and NRC computer systems, and a draft report of the project has been issued. The first version of the CORCON code has been documented as a stand-alone tool and is currently being interfaced with CONTAIN. The MAEROS code is a primary submodule of CONTAIN and lacks only final refinement; a draft report has been given limited distribution. The MAEROS aerosol model is the state-of-the-art tool for computing fission product transport via aerosol migration and will be tested against a number of other similar codes such as HAARM, QUICK, and NAUA.

5.7.5 Plan of Work As a Function of Time

The CONTAIN-CORCON interface will be operational in FY 1982. Verification of the aerosol treatment will be completed as final experimental data become available. The LWR-model specifications will be completed in FY 1982 along with completion of the LWR-debris/coolant interaction model. The first complete LWR version of CONTAIN will be operational by the first part of FY 1984, and verification will be continued throughout FY 1984 as results of the core-concrete experiments become available.

The CORCON code will be extended to treat the conditions expected later in the accident scenario, i.e., attack on concrete by slurred pools and solid crusts of high-temperature core materials. This will be completed by the first part of FY 1984.

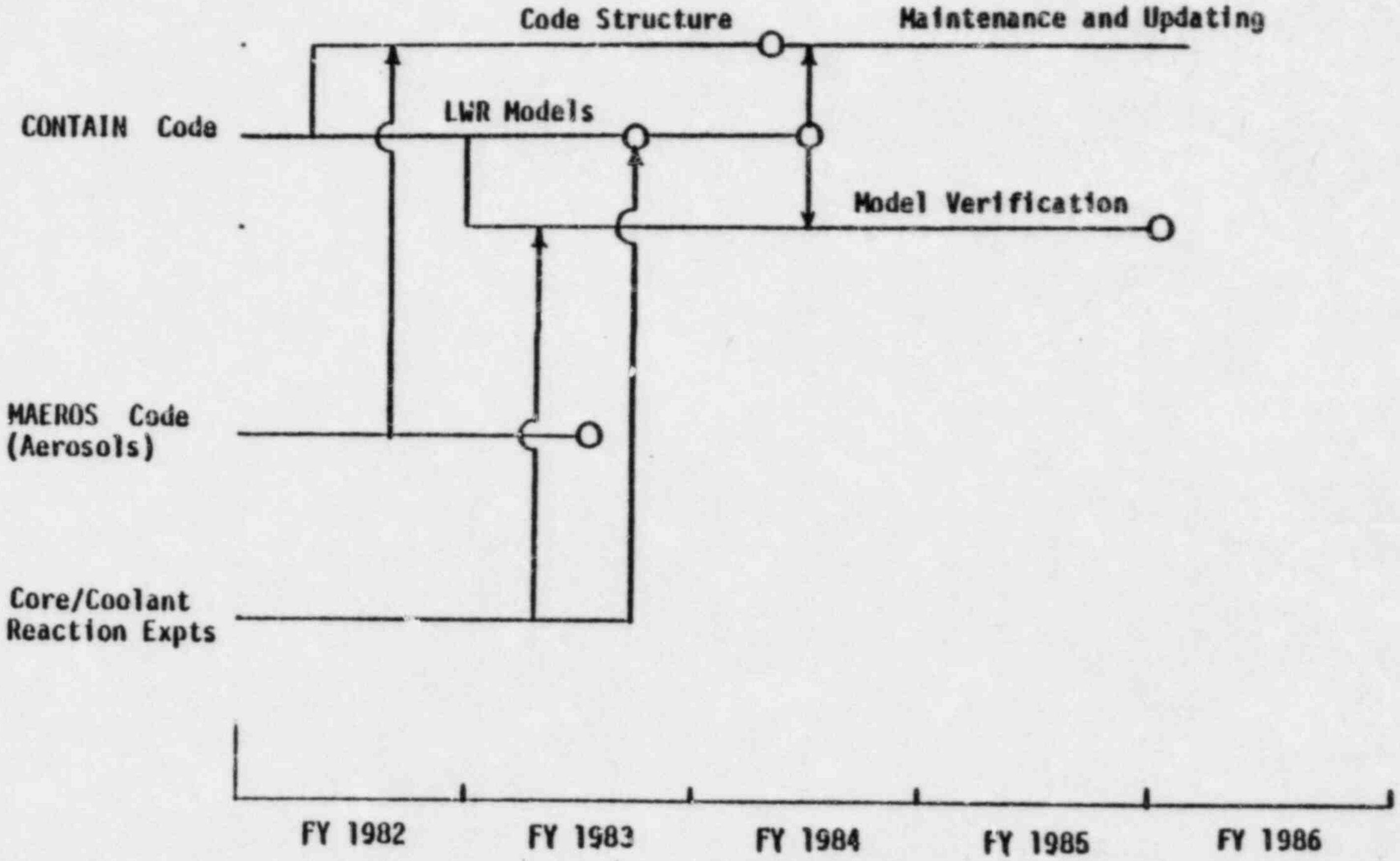


Figure 5.7 Containment Analysis

5.8 Containment Failure Mode

5.8.1 Element Description

The major source of risk to the public from the operation of nuclear power plants stems from accident scenarios that lead to a containment failure. The regulatory concern in this element 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. This is especially so if the contemplated failure mode is localized leakage. Both assessments of the risk posed by loads outside the design basis, such as hydrogen burns, and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment will fail.

This element does not address, however, the failure mode arising from the failure to isolate the containment due to improper valve positioning. Both utilities and the NRC address this part of the problem through quality assurance practices, inspection and enforcement and other administrative and management techniques. Scenarios that bypass containment via penetrations are also treated in the SASA and in the Accident Consequence and Risk Evaluation elements. This element treats three possible failure modes: faulty valve operation, materials failure in electrical penetrations due to high temperature, and mechanical failure of the containment due to either excessive local deformation at major penetrations or structural failure.

The main safety question relates to the ability to predict, with high confidence, the amount of load that can be sustained by a containment structure before the rate of leakage becomes unacceptable. State-of-the-art methods can not reliably predict whether leakage will begin around penetrations or in the membrane region of the shell. If leakage at penetrations is critical, the effects of aging on gasket performance will be of significance. The technical problems involve developing an ability to predict deformations for the wide array of containment types, relating deformations of containment structures to leak behavior, and determining the sensitivity of predictions to uncertainties in actual containment structures and the loads associated with accident scenarios.

The objective of this element is to develop and verify methodologies that are capable of reliably predicting the capacity of containment structures under accidental and severe environmental loadings. The reliability of any predictive method must be verified through experiments. This project contains a combined analytical and experimental effort towards the establishment of reliable predictive methods for the safety margins of containment structures under accident and severe environmental conditions. Steel and concrete containments, both prestressed and reinforced, will be studied for pressure and seismic loadings.

Failure resulting from both material rupture of the containment shell and excessive leakage due to deformations at major penetrations, exclusive of electric penetration aperature seals, will be studied. The temperature and pressure histories necessary to cause excessive leakage due to degradation of elastomeric materials in electrical penetration conductor seals will be established. The potential for containment failure due to faulty operation of containment isolation valves and the significance of possible containment fan cooler failure will be examined as part of a research program on equipment qualifications.

5.8.3 Key Interfaces With Other Elements

There is, and will continue to be, significant interaction with other NRC-sponsored programs related to the severe accident research program. Particularly close coordination will be maintained with the programs on Fuel-Structure Interaction, Hydrogen Generation and Control, and Containment Analysis. In addition, there will be interactions with the Risk Code Development element. There will also be interaction with other U. S. programs. Contributions to the Containment Failure Mode program element are anticipated by the provision of analytical predictions of capacity to be compared against test results from the Electric Power Research Institute. There will be coordination with the containment capability program being contemplated by the Department of Energy and the IDCOR program on containment overpressure response.

Two foreign programs have been identified as sources of information. One is the proposed test-to-failure of a scaled model prestressed concrete containment to be conducted in Great Britain. The other is the planned testing, on a shake table in Japan, of containment models to simulate seismic response.

5.8.4 Background and Status

Effort in FY 1982-1983 will be limited to determining containment capacity under static overpressure. The principal FY 1982 activity is experimentation involving the examination of six prototypical steel models about 1/32 the size of a containment and the design of the large prototypical steel model about 1/10 actual size. This large steel model will be used in the FY 1983 experimental effort.

Other FY 1982 activity is directed toward the understanding of behavior of steel containment structures including deformation-leakage relationships. This effort will be used for analytical support of the experiments, the selection of analytical methods for assessment of predictive capabilities, and the comparison of the analytical predictions with the experimental results. Also included in FY 1982 is the investigation of concrete containment behavior beyond the elastic limit of its steel reinforcement and of its steel prestressing tendons. This effort will be used for the design of reinforced concrete prototypical models that will be fabricated in FY 1983, and for supporting analytical investigations. A small part of the FY 1982 activity is directed toward obtaining data from the Canadian experiment on a prototypical model of the CANDU containment and interfacing with the British on their proposed experiments on a model of the SNUPPs containment. It is anticipated that the British tests will be conducted in FY 1983.

5.8.5 Plan of Work As a Function of Time

Effort in FY 1984 will concentrate on three items: (1) an evaluation of analytical predictions of steel containment capacity in light of the experimental results, (2) the conduct of tests-to-failure, under static overpressure, of models of reinforced concrete containments, and (3) completion of the comparison of predictive methods for prestressed concrete containment behavior against the British data.

Six tests-to-failure are anticipated for concrete models. All models will be approximately 1/10 the size of a typical containment. 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 seismic reinforcement and penetrations and will provide data against which three-dimensional predictions can be compared.

The planning of dynamic, unsymmetric pressure tests will begin in FY 1984. Based on results from the hydrogen combustion 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. Experiments related to seismic effects are scheduled to begin in FY 1988.

The following results are anticipated during FY 1984-1987 and are shown graphically in Figure 5.9.

FY 1984 Comparisons of estimates of steel containment capacities with experiments under static pressure; acceptance criteria for seismic peripheral shear values under biaxial tension.

FY 1985 Comparisons of predicted capacities for prestressed and reinforced concrete containments with experiments under static pressure.

FY 1986 Comparisons of predictions of steel containment capacity under dynamic pressure loads with experimental results.

FY 1987 Comparisons of predictions of capacity for reinforced and prestressed concrete containments under dynamic pressure loads with experimental results.

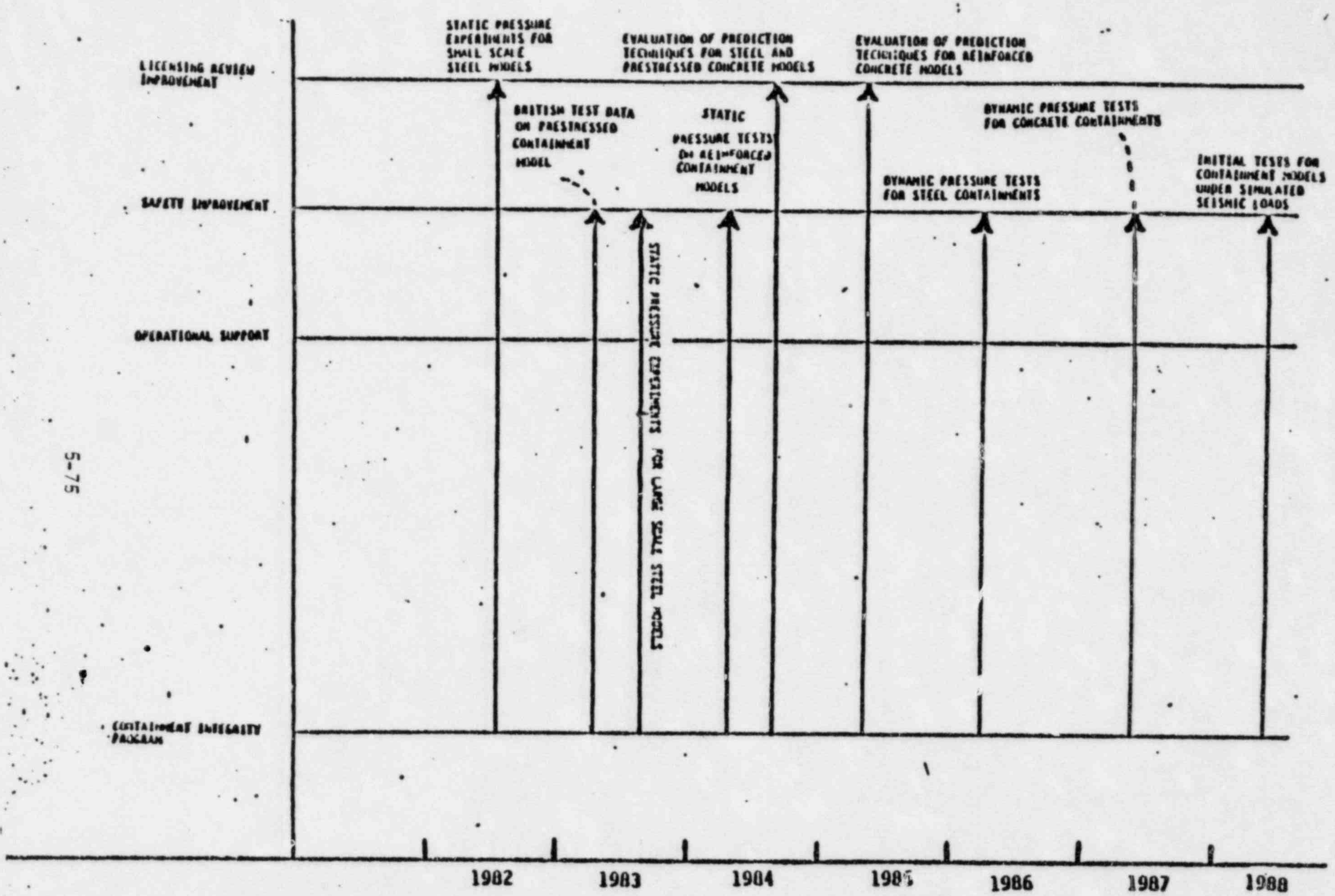


Figure 5.8 Containment Failure Mode

5.9 Fission Product Release and Transport

5.9.1 Element Description

Fission product release and transport research is directed at developing an experimental data base and models to predict the radiological source term for accident consequence assessment. This information is needed for emergency preparedness, risk assessment studies, siting rulemaking actions, and for equipment qualification analysis. While a significant amount is known about fission product release and transport under controlled LOCA conditions, there are gaps in the data base relative to fission product release and transport behavior under severe core damage and core melt accident conditions.

Nuclear power reactor safety studies consistently indicate that the uncertainties associated with estimating fission product release and transport behavior are among the largest contributors to uncertainties in the risk to the public from severe accidents at nuclear power plants. This result is not surprising for two reasons: (1) offsite consequences are directly affected by the magnitude, timing, and makeup of the source term released from containment, and (2) there are large uncertainties regarding the actual potential source term. The ultimate objective of this research program is to improve the quality of predictions of the potential fission product radiological source term released from containment under accident conditions.

5.9.2 Technical Issues Resolved by This Element

NUREG-0772 identified a number of key uncertainties related to estimating fission product source terms. The most important of these are:

1. Reactor coolant system (RCS) aerosol and fission product behavior (experimental data for model verification),
2. RCS thermal/hydraulic conditions 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 hydrogen combustion on fission product physical and chemical forms (experimental data), and

9. Coupled models of containment fission product vapor transport, aerosol behavior, steam effects, and effects of ESFs.

The objective of these research programs is to develop a data base for assessing fission product release from the fuel and fission product transport behavior 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 reactor coolant system and in the containment, and the effectiveness of engineered systems in mitigating fission product release under severe accident conditions.

5.9.3 Key Interfaces with Other Elements

Radiological source term and radiological source term analysis require definition of accident sequence characteristics. This need is supplied by probabilistic risk assessment studies (elements 5.1, 5.2, 5.10, 5.11) that identify overall system performance and the dominant accident sequences.

Fission product release and transport analysis also requires detailed information on the physical processes that occur during severe accidents. Among the most important are:

1. RCS thermal-hydraulic behavior,
2. Fuel heatup, melting, movement, etc. (element 5.4),
3. Molten fuel/concrete interactions (element 5.6),
4. Molten fuel/coolant interactions (element 5.6),
5. Containment response to severe accident loads (i.e., failure time and mode) (element 5.8), and
6. Hydrogen combustion (element 5.5).

Major uses of the results of the fission product release and transport source term research are equipment qualification, probabilistic risk assessment, definition of siting requirements, and emergency planning.

5.9.4 Background and Status

An intensive program to evaluate realistic source terms for severe LWR accident sequences was conducted during the Reactor Safety Study, WASH-1400. 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 systems. 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 LWR primary coolant systems 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, MAEROS.

5.9.5 Plan of Work As a Function of Time

The following three sections describe specific research projects and near-term results expected during FY 1982 and FY 1983. Figure 5.9 presents a detailed milestone schedule for these programs.

1. Fission Product Release. Research programs to investigate and quantify the release of fission products and aerosols from the fuel include:

   a. 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 early FY 1982 with the higher-temperature tests (to 2600°C) to begin in early FY 1984.

   b. Experiments to investigate the release of fission products and structural material aerosols from larger bundles of fuel (.5 to 10kg) using simulated irradiated fuel (fissium) and out-of-pile heating technique (FY 1982-1983).

   c. A program to investigate the release of aerosols from molten pools of core materials interacting with reactor cavity concrete with core retention materials, and with residual coolant (ending in FY 1984).

d. Examination and analysis of samples of the TMI-2 core (schedule depending on the TMI-2 cleanup schedule).

e. Development and improvement of mechanistic models (FASTGRASS AND START) to predict the release of fission products during interactions of the damaged and molten fuel with residual coolant and plant structures.

f. Measurements of fission product release during Phase 1 severe fuel damage testing in the PBF reactor (FY 1982 and FY 1983).

2. <u>Fission Product Transport</u>. Research programs in the areas of fission product vapor and aerosol transport and deposition include:

a. 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 behavior and models for fission product (and aerosol) release from the core (ongoing, to be completed in FY 1984). Results from this program will be factored into the CONTAIN code.

b. An experimental and analytical program to provide model development data for the TRAP-MELT code in the area 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).

c. 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 the CONTAIN code to predict overall fission product transport behavior. These improved mechanistic codes will be used to benchmark simpler models in MELCORR. (To be completed in FY 1983.)

d. A series of small scall experiments will be initiated to provide data for interim verification of 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 83.

e. Modification and operation of a 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.

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

3. <u>Fission Product Control</u>. 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 area are programs to:

a. Investigate and quantify the radioiodine retention performance of impregnated activated charcoal adsorbers under accident conditions (completed in FY 1982).

b. Conduct research on the fission product mitigation performance of engineered safety features (e.g., containment spray systems, suppression pools, ice condenser beds) under the radiological and environmental conditions predicted for severe core damage and core melt accidents.

c. Study the effects of ' rge aerosol sources (predicted for the most severe accidents) on engineered safety feature performance.

4. <u>NUREG-0772 Follow-On Research</u>

a. Updated, severe accident, release-from-plant, fission product source terms to supplement WASH-1400 estimates will be developed. (Completed in FY 1983).

b. Quantitative estimates of the uncertainties associated with these source term predictions and identification of the major sources of the uncertainty will also be provided by this study. (Completed in FY 1983.)

c. Analysis of past reactor accidents and core destructive tests for insights into fission product release and transport behavior and to compare current assumptions and models with measured releases are underway (completed in FY 1982).

5. Longer-Term Research Program Plan (FY 1984-1988)

a. Fission Product Release From Overheated Fuel - Beginning in FY 1984, 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. The test apparatus will include techniques (laser Raman spectroscopy) for direct in situ determination of fission product chemical form. Two test series will be conducted in FY 1984, three in FY 1985, three in FY 1986, and two in FY 1987.

b. Reactor Coolant System (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.

c. 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 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 analysis will be performed using the extended TRAP-MELT code, the CONTAIN code, and/or the MELCORR/MATADOR code on large-scale containment aerosol tests. (To be conducted in the Federal Republic of Germany). Again, these analyses should be completed by FY 1986.

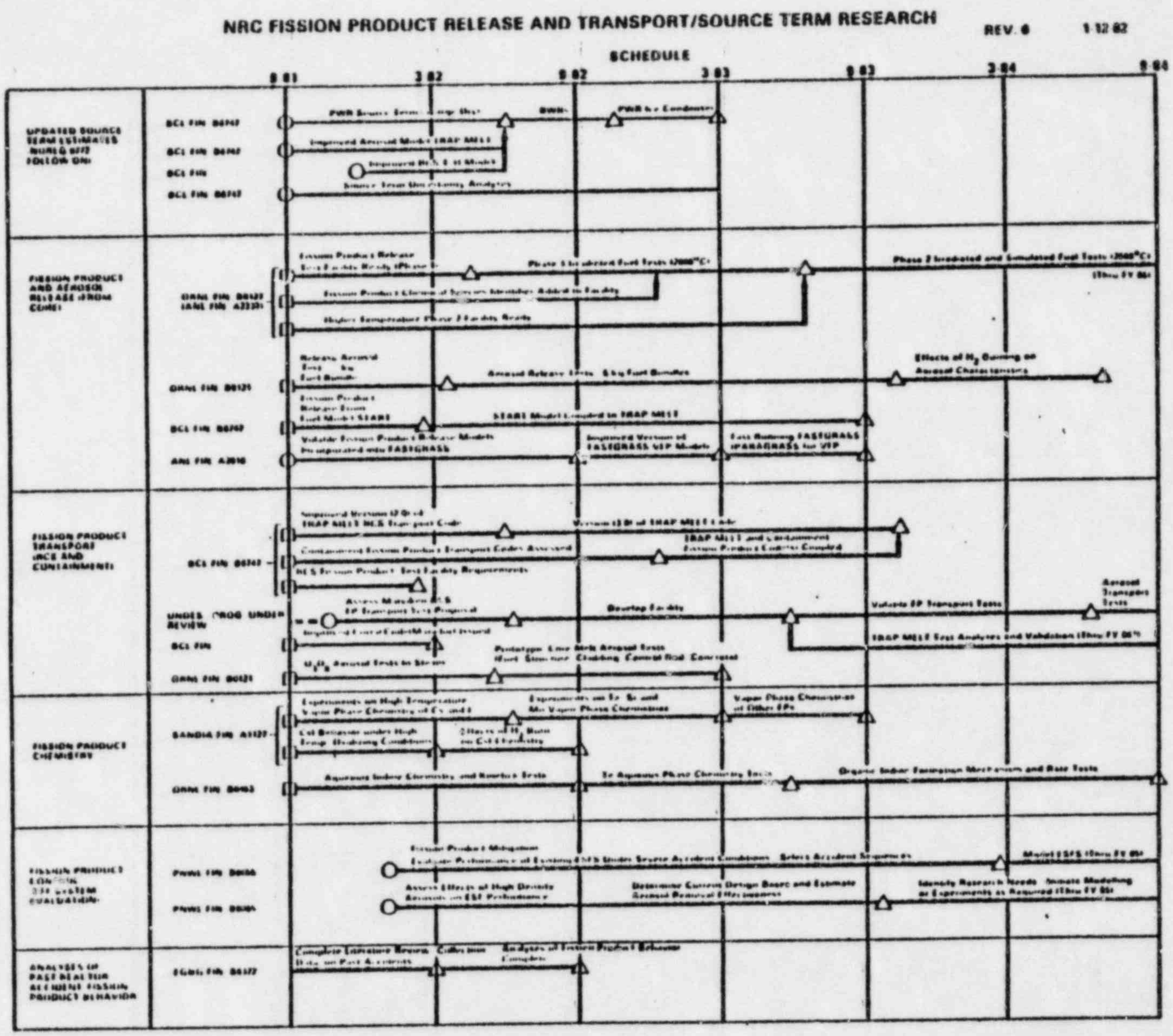


Figure 5.9 Fission Product Release and Transport

5.10 Risk Code Development

5.10.1 Element Description

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

The code development work in this element is being undertaken in order to correct identified deficiencies in existing risk codes (MARCH, CORRAL/MATADOR, and CRAC) (See Section 5.10.4). These deficiencies relate both to the modeling of physical processes within the codes and to the actual structure of the code. The nature of the deficiencies is such that some are amenable to short-term upgrading while others require longer time or supporting phenomenological research. For this reason, the code development in this element is to be performed in two parallel paths, one relatively short term, the other long term. These two paths are described briefly below.

1. MARCH-2/MATADOR Development Program

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

2. <u>MELCOR Development Program</u>

The MELCOR development program is intended to produce the longer-term replacement computer code for use in risk studies. One fundamental characteristic of this code is that it is to be developed using a "data management system" and a modular structure. This then permits the following:

- Ease of model replacement as new experimental data and analytical models become available;

- Direct and completely compatible linkage between in-plant thermal-hydraulic and radionuclide transport, and ex-plant consequence models, permitting advances in both "best-estimate" calculations and uncertainty propagation; and

- Code and model scrutibility and maintainability.

As noted just above, unification of the subject areas of the present three codes under MELCOR is being undertaken to permit direct assessment of the entire course of a severe accident, a feature particularly important to uncertainty analyses. It should be noted, however, that it is intended to develop the code in such a way that major portions of it (e.g., that portion equivalent to the present MARCH code) can be run independently, if so desired.

5.10.2 Technical Issues Resolved by This Element

After undergoing extensive reviews, both the MARCH and MATADOR codes have revealed a number of deficiencies. Such deficiencies have resulted, in some instances, in the need for numerous additional supporting calculations, sensitivity studies, etc. In the MARCH-2/MATADOR program, the more important and more readily resolvable deficiencies will be accounted for and thus will result in risk codes having additional credibility and requiring relatively fewer supplemental analyses.

The longer-term MELCOR program will be used to develop a risk code structure that is much more readily understandable and amenable to modification, through the use of structured computer programming and the data management system noted above. In effect, this code development thus provides a scrutible means by which new data can be incorporated in physical process analyses for PRA and uncertainty analyses can be performed.

5.10.3 Key Interfaces with Other Elements

This code development will use results from various experimental programs in elements 5.4 through 5.9, during and for MELCOR development, and will provide improved capability for consequence prediction in the elements "Accident Consequence and Risk Reevaluation," and "Risk Reduction and Cost Analysis." as well as for other PRA studies not specifically included in this plan.

5.10.4 Background and Status

The risk codes now in use (MARCH 1.1, CORRAL 2, CRAC 2) had their origins in the analyses performed for the Reactor Safety Study (RSS). Following the release of the RSS, work was initiated to improve upon the initial code versions; CORRAL-2 thus became available in 1977 and MARCH 1.1 and CRAC 2 in 1981. Since 1977, work has also been under way to upgrade CORRAL-2 to account for new supporting data, to add models of certain phenomena not previously included, and to make its structure more amenable to modification. This work has led to a preliminary version of the code, now renamed "MATADOR."

Each of these risk codes has been the subject of critical review. Assessments of MARCH 1.1 were performed as part of specific operating license reviews (NUREG/CR-2228, NUREG-0850) and in a more general way (NUREG/CR-2285). The MATADOR code has been subject to an independent peer review as part of its development and completion. CRAC 2 has been used in an international comparison and validation exercise. As noted above, these reviews have resulted in the identification of important deficiencies. The MARCH-2/MATADOR development program will provide a short-term means to correct the most important identified deficiencies in these codes. The MELCOR development is intended to provide the longer-term means for the correction of such deficiencies, especially through the incorporation of models based on the ongoing experimental programs described in other elements.

5.10.5 Plan of Work As a Function of Time

Working versions of MARCH-2 and MATADOR are scheduled to be completed in July 1982 with public release by the end of FY 1982. A working version of MELCOR is planned to be available late in FY 1983, with a final version available in mid-FY 1985. Figure 5.10 shows this schedule and its relationship to other research elements.

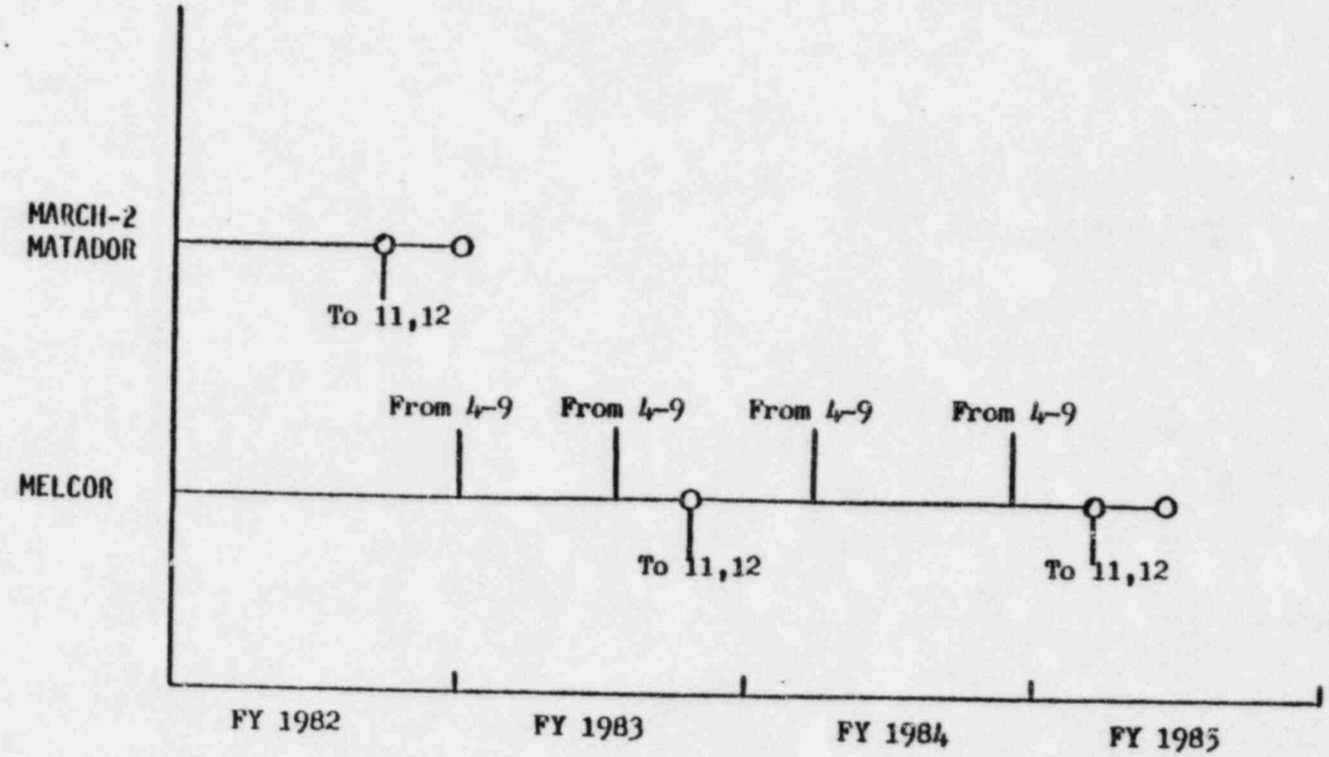


Figure 5.10 Risk Codes

5.11 Accident Consequence And Risk Reevaluation

5.11.1 Element Description

This element relates to the application of advanced versions of risk codes for the re-analysis of the consequences of important accident sequences. That is, as the severe accident physical process risk codes are developed (as discussed in the preceding element), they will be put to use in this element to reanalyze the consequences of accident sequences determined to be important in previous risk studies and in the "Accident Likelihood Analysis" element. Further, as these consequence analyses are completed, they are to be combined in this element with the sequence likelihood results, resulting in the redefinition of the risk of studied plants. In this way, previously completed risk studies can be periodically updated to reflect the latest advances in accident likelihood and consequence analysis.

5.11.2 Technical Issues Resolved by This Element

The principal technical issue being resolved in this element is providing a periodically updated set of consequence and risk analyses through which best estimates of actual levels of plant risk can be determined. Such determinations of plant risk levels are important both for consideration with respect to possible safety goals and as a basis for assessing the risk reduction benefit of possible plant modifications.

5.11.3 Key Interfaces With Other Elements

This element has important interfaces with the "Accident Likelihood Analysis" and "Risk Code Development" elements, drawing from them information on accident sequence likelihoods and improved analytical capabilities, respectively. In addition, the results of this element are an important input to the "Risk Reduction and Cost Analysis" element, providing the periodically updated risk reevaluations described above. It also provides input to the "Regulatory Analysis and Standards Development" element by providing further evidence on the effectiveness of current requirements at limiting risk.

5.11.4 Background and Status

Present-day use of PRA in regulatory decisionmaking relies heavily on risk studies such as the Reactor Safety Study and RSSMAP. Such PRAs do not fully represent and account for many issues that have arisen since their analyses were performed. Examples of such issues include the possible reductions in fission product source terms in some accident sequences; potentially conservative treatment of certain containment threats (i.e., the "steam spike"); and the potentially important likelihood of reactor vessel failure due to pressurized thermal shock. Up until the present time, no formal mechanism for incorporating such matters into previously completed and much-used PRAs has been available. With the recent initiation of accident likelihood analysis and risk code updating (both described above), the work of this element is intended to become such a mechanism.

5.11.5 Plan of Work As a Function of Time

It is planned that the performance of these consequence and risk reevaluations will be performed iteratively and at roughly 1-year intervals. This schedule and its interrelationships with other elements is shown in Figure 5.11.

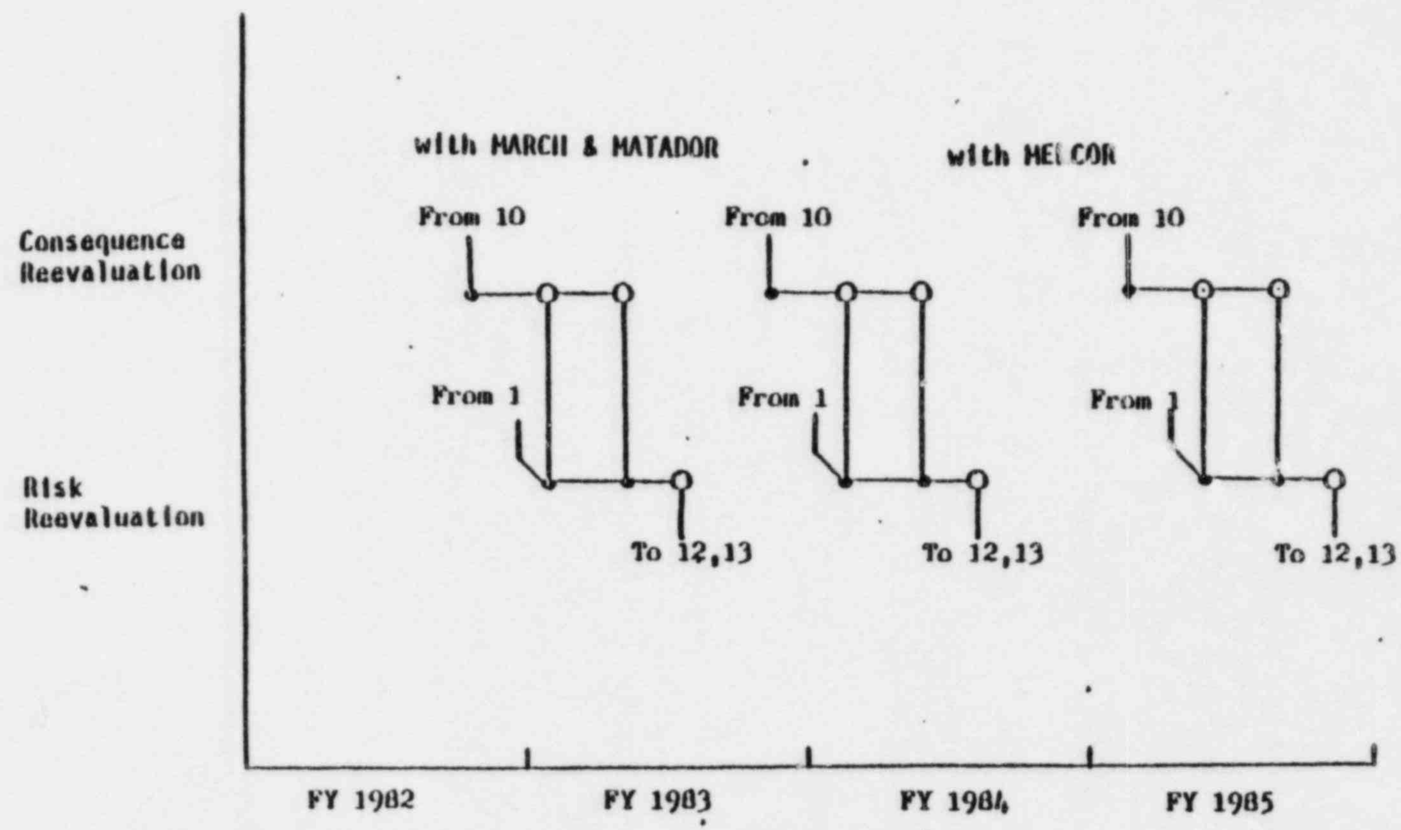


Figure 5.11 Accident Consequence and Risk Evaluation

5.12 Risk Reduction and Cost Analysis

5.12.1 Element Description

In this element, analyses of the risk reduction potential and costs associated with a spectrum of possible plant modification are to be performed. Included in these possible modifications are, for example, filtered-vent containment systems, alternate shutdown heat removal systems, and stronger containments. The objective of such analyses is to identify those modifications that appear to present the most cost-effective risk reduction. Since such results will vary with the specific plant design being considered, analyses are to be performed for all major design types (PWR large dry and ice condenser containments and BWR Mark I, II, and III designs). This work is being performed in a series of several phases, each roughly one year long. The first phase uses existing data to define a relatively small set of promising modifications from a more broad set (see Table 5-2). This work is to be completed in the summer of 1982. This narrowed list will then be the subject of more detailed study using improved codes (from Element 5.10), accident likelihood estimates (Element 5.1), and accident consequence and risk estimates (Element 5.11). Using this work, a comprehensive definition of the cost effectiveness of possible plant modifications is to be obtained late in the summer of 1983.

An example of the technique used for the determination of possible plant modifications (and combinations of modifications) to be analyzed in the first phase is shown in Table 5-3. For a particular plant, existing PRA results are studied and the functional characteristics of the more important accident sequences identified. Thus the first column in Table 5-3 shows the types of functional failures for each sequence and the relationships among these failures (e.g., ECC failure results from containment failure in the first sequence). Using this information for each sequence, the applicability of possible plant modification (see Table 5-2) to cope with this sequence and its functional failures is determined. Examination of the matrix of applicable modifications for all important sequences provides the means for groups of modifications which qualitatively appear to be most promising. This screening technique thus provides a method for developing and analyzing in more detail the risk reduction benefit and cost of a finite number of modifications and combinations.

5.12.2 Technical Issues Being Resolved by This Element

The principal technical issue being resolved in this element is the identification of those possible plant modifications that offer the most cost-effective means of reducing risk for the set of major LWR design types.

5.12.3 Key Interfaces With Other Elements

Important information for this element is to be obtained from the "Risk Code Development" and "Accident Consequence and Risk Reevaluation" elements. The first element will provide the analytical models for use in the risk-reduction studies, while the second element will provide updated levels of plant risk to be used as benchmarks for the risk-reduction studies.

5.12.4 Background and Status

In 1978, the NRC performed studies at the request of Congress to identify those areas of LWR design that appeared to offer the greatest potential for improving the safety of these plants. The results of these studies are reported in NUREG-0438, "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants." In this report, two possible plant modifications were identified as having the most promise for improving safety, these modifications being filtered-vent containment systems and alternate shutdown heat removal systems. As a result, programs to investigate this potential in more detail were initiated in 1979 and have continued up to the present time. In 1981, it was decided to link these programs with a new program that would study these modifications in concert with studies of a somewhat broader spectrum of possible changes (shown in Table 5-2).

This program has two general objectives: (1) to develop conceptual designs of a set of possible plant modifications for preventing or mitigating severe accidents, and (2) to perform analysis of the risk-reduction value and cost of these possible additional plant features for comparison with developed reevaluations of LWR risk.

This program will be performed in an iterative manner. In the program's first phase, a broad spectrum of possible severe accident prevention and mitigation features will be screened using semiquantitative techniques to develop a small set of the apparently most promising features. The initial scoping study, based on the existing RSS and RSSMAP risk assessments, will be published in the summer of 1982. It will serve to narrow the range of design variants warranting further study with improved PRA models in the 1983 and later iteractions. Following this, conceptual designs will be developed for these possible plant modifications. These designs will be developed only to the degree necessary to permit rough evaluations of their attendant risks, reliabilities, and costs.

To fulfill the second objective, the risk-reduction value of these designs will then be evaluated, using MARCH-2 initially, to be followed later by the use of MELCOR. For each design, this evaluation will include estimates of its capability to reduce either the likelihood or consequences of a severe accident and the "competing risk," i.e., the potential risk increase associated with such matters as spurious operation. This assessment of the net risk-reduction value will be combined with cost estimates to determine the values and impacts of the various features.

5.12.5 Plan of Work As a Function of Time

The details of the schedules of these programs and their relationships to other elements are shown in Figure 5.12. As noted above, the first major milestones involve:

- the completion of effort to narrow a range of possible risk-reduction modifications to the most promising ones (in the summer of 1982); and

- the completion of a more detailed study of the narrowed set for their cost-effectivenesses in reducing risk (in the summer of 1983).

Dominant LOCA Functional Accident Sequences					Applicable Candidate Degraded Core Safety Approaches									
RS	ECC	COE	COL	RR	Additional Containment Heat Removal Systems(1)	Addition of Hydrogen Control Devices(3)	Addition of Core Retention Devices(4)	Addition of Containment Venting System	Addition of Particulate Removal System	Additional Coolant Injection Systems	Increase RPS Reliability	Increase Containment Design Pressure(6)	Addition of Missile Shields(7)	Other Accident Specific Fixes(8)
	PCF		X	PCM	X			X				X	X	X
	X		PCM	PCF		X	X			X		X	X	X
Dominant Transient Functional Accident Sequnces														
RS	ECC	RCSOP	CO	RR										
	PCF		X	PCM	X			X				X	X	X
	.X		PCM	PCF		X	X			X		X	X	X
X	PCF		X	PCM	X See Note 2.			X See Note 5.			X		X	X

PCF - Safety function fails post containment failure.

PCM - Safety function fails post core melt.

Notes

1) Prevents containment overpressure from gas generation.
2) CHR system capacity must equal 20-30% of operating power.
3) Prevents containment overpressure from hydrogen burning.
4) May prevent containment failure if device reduces noncondensible gas generation.
5) Vent must be of high flow design for ATWS.
6) Affects probability of containment overpressure from hydrogen burning or gas generation.
7) Prevents containment failure from steam explosions.
8) These include increasing the reliability of emergency AC power systems, safety/relief valve reclosure, and Automatic Depressurization System actuation.

Table 5-3 Results of Initial Safety Option Characterization

Table 5.2

Severe Accident Safety Features Under Consideration

CANDIDATE IMPROVEMENTS	OPTIONS
1. ADDITIONAL CONTAINMENT HEAT REMOVAL	ACTIVE VERSUS PASSIVE
2. CONTAINMENT ATMOSPHERE PARTICULATE CAPTURE	
3. CONTAINMENT ATMOSPHERE MASS REMOVAL	FILTERED VERSUS UNFILTERED LOW FLOW VERSUS HIGH FLOW
4. INCREASED CONTAINMENT MARGINS	INCREASED VOLUME INCREASED PRESSURE CAPABILITY PRESSURE SUPPRESSION FEATURES
5. COMBUSTIBLE GAS CONTROL	DELIBERATE IGNITION INERTING (PRIOR/POST ACCIDENT) FIRE SUPPRESSION (HALON/WATER FOGS)
6. CORE RETENTION DEVICES	DRY VERSUS WET ACTIVE VERSUS PASSIVE COOLING OR NO COOLING
7. MISSILE SHIELDS	IN-VESSEL STEAM EXPLOSIONS VESSEL THERMAL SHOCK VESSEL MELT-THROUGH AT HIGH PRESSURE EX-VESSEL STEAM EXPLOSIONS COMBUSTIBLE GAS EXPLOSIONS
8. BWR CONTAINMENT SPRAY SYSTEM	
9. PWR PRIMARY SYSTEM DEPRESSURIZATION	AUTOMATIC VERSUS MANUAL ADDITIONAL RELIEF CAPACITY PRESSURE SUPPRESSION FEATURES RADIOACTIVITY REMOVAL SYSTEMS
10. ADD-ON DECAY HEAT REMOVAL SYSTEMS	HIGH PRESSURE VERSUS LOW PRESSURE OPEN LOOP VERSUS CLOSED LOOP PRIMARY SYSTEM VERSUS SECONDARY SYSTEM
11. SPECIFIC PREVENTION CONCEPTS	IMPROVED DRAIN OR VALVE DESIGN IMPROVED MAINTENANCE PROCEDURES IMPROVED CONTROL LOGIC REDUCTION OF COMMON MODE DEPENDENCIES

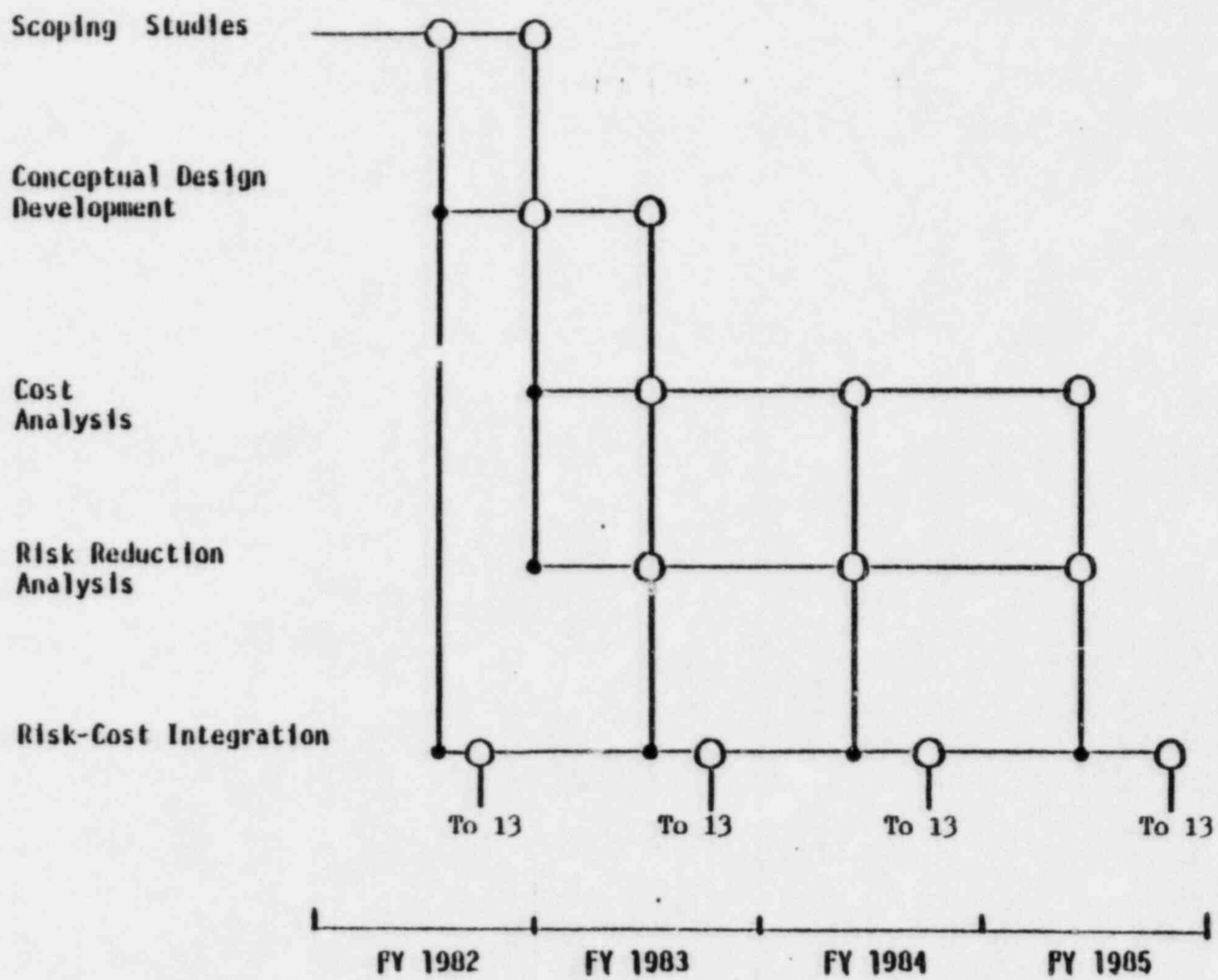


Figure 5.12 Risk Reduction and Add-On Cost Benefit

5.13 Regulatory Analysis and Standards Development

5.13.1 Element Description

A variety of projects are under way or planned that directly support reactor safety standards development and the development of staff aids and guides for use in regulating severe accident risks. These include:

1. Development of value-impact or cost benefit guides and improved techniques for use by the staff in evaluating new requirements,

2. Risk assessment sensitivity analyses to catalog the ways in which nuclear power plants might pose severe accident risks well in excess of those in the Commission's safety goals or suggested by published reactor PRAs,

3. Studies of the risk-limitation effectiveness of the general design criteria, the regulatory guides, and the standard review plan,

4. Studies of the reliability with which high risk designs or procedures are identified and corrected by safety evaluation, quality assurance (QA), inspection and enforcement, industry safety practices, and experience feedback systems,

5. Studies of the limitation, implications, and ways of implementing the Commission safety or safety goals policy,

6. Analysis of reliability assurance practices in nonnuclear industries and adaptation of the more promising methods to the nuclear regulatory arena,

7. Systematic review of the technical content, interdependence, and institutional forces acting on the current body of reactor safety standards, and

8. Development of draft rules, regulatory guides, review plans, and aids to regulatory decisionmaking.

5.13.2 Technical Issues Resolved by This Element

The current body of reactor safety standards limits the risk posed by severe accidents through the mechanism of conservative reactor design, through the postulation of design basis accidents (from which most safety systems derive their principal safety design bases), and through a variety of regulations intended to ensure safety system reliability (the single failure criterion, QA, technical specifications, conservative deterministic codes and standards, etc.). It is now widely recognized that accidents different in character or far more severe than the design basis accidents are credible. The more probable causes of such accidents are believed to originate in multiple failures or human errors outside the domain of the single failure criterion or of those common-cause failure mechanisms currently addressed in the regulations (seismic qualification, safeguards, fire protection, etc.).

Most of the purely technical issues involved in resolving these regulatory issues are dealt with in the many other research elements described in this plan. However, the regulatory issues and the interface with the technical issues embracing the adequacy of the design basis accidents and the adequacy of the several ways of assuring safety system reliability are the subject of this element.

5.13.3 Key Interfaces with Other Elements

The work of this element is closely coupled (both input and output) with policy standards development initiatives at the Commission and NRR, including: (1) safety goals and NRR implementation strategies; (2) plans for the resolution of Unresolved Safety Issues; (3) requirements for plants at high-population density sites such as Indian Point and Zion; (4) requirements for near-term CPs; and (5) requirements for standard plants or licenses to manufacture.

The element will draw from three other elements of the plan: (1) Accident Likelihood Analysis; (2) Accident Consequence and Risk Reevaluation; and (3) Risk Reduction and Cost Analysis. The draft standards emerging from the element will be the principal product of the accident research program apart from the codes, data, insights, and reports issued by the many other elements.

5.13.4 Background and Status

A number of severe-accident-related rule initiatives have already been undertaken. These include the final and proposed rules for the control of hydrogen during severe accidents and the 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.

Research to address the DC power Unresolved Safety Issue has been completed and has resulted in the formulation of new proposed regulations. Comparable research on Station Blackout, Alternate Decay Heat Removal Systems, Alternate Containment Concepts, and Core Catchers is also dealing with issues involving design criteria and formulation of regulations.

Research into the consequences of severe reactor accidents is currently being used in the development of a guide and a handbook for value-impact assessment that would enable the placing of a value on regulatory initiatives to reduce the frequency or severity of severe accidents. Tables of the present worth of projected losses for accidents characterized by user-input frequency and severity classification will be published in the late spring of 1982. A first, partial edition of the value-impact handbook will appear near the end of FY 1982. More comprehensive editions will follow at intervals thereafter.

A "reverse" risk assessment, a sensitivity analysis for light water reactors, will be published in FY 1983. It will employ current PRA results, together with state-of-the-art analysis of the limitations of PRA, to give a systematic answer to the question, "If a nuclear plant were to pose severe accident risks greatly in excess of the Commission's safety goals, how might this come about?"

It will attempt a comprehensive catalog of the uncertainties in accident likelihood and consequence estimation through which very high risks might have escaped discovery.

A study of the risk relevance and effort of implementing the pre-TMI standard review plan has been completed in draft form. It demonstrates that studies of the risk-limitation effectiveness of regulations are feasible. A new effort to examine the risk-limitation effectiveness of the general design criteria (GDC) is scheduled for FY 1982. It will be followed by comparable studies of other parts of 10 CFR Part 50, the regulatory guides, and the current standard review plan at 6-month intervals thereafter. These studies will be similar to that done previously to rank the 133 generic safety issues in 1978. The basis for these studies will be the many reactor PRAs and safety system reliability assessments that have been and are currently being published. These PRAs contain many findings about the comparative importance to safety of system design features and operations practices. These studies, (i.e., of GDC and other sections of Part 50) are expected to identify and document the evidence supporting inferences of unnecessary overregulation as well as cases of safety-significant loopholes in the regulations.

The studies of the risk-limitation effectiveness of the regulations will be followed by a study of the ways in which industry compliance practices and NRC inspection and enforcement, and the experience feedback mechanisms of the industry and of the NRC, detect and correct safety-significant defects in design and plant operations. The risk-based studies of nuclear plants, particularly the reverse risk assessment, will provide many clues about the ways safety might be compromised by poor design choices, provisions for test and maintenance, and the conduct of operations. These will be assembled into an analysis of the reliability with which such flaws are detected and corrected. The resulting findings will be classified and proposals made for regulatory initiatives to deal with any prominant weak spots in the fabric of safety assurance practices. This work is scheduled for FY 1984.

For some time, there has been research into the technical bases of quantitative safety goals. The Commission will soon publish a safety policy statement, including qualitative and quantitative safety goals. A technical analysis of the implications, strengths, and weaknesses of the Commission goals will be prepared in the spring of 1982. Additional studies of implementation strategies willl be performed in late FY 1982. The impact of the safety policy on severe accident regulatory policy will be studied in detail.

In FY 1982 a program was begun 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 ensure the reliability of complex systems. Probabilistic system reliability analysis was originally invented in these industries. Many of their techniques have never been tried in the area of reactor safety, however. The FAA has adapted many of these techniques in a regulatory arena. These approaches to safety assurance or reliability assurance will be studied to identify, if possible, ways to sharpen the focus of reactor safety assurance requirements to establish a higher level of risk-limitation effectiveness while avoiding unnecessary overregulation.

As the foregoing research programs develop a picture of the needed or desirable changes in the reactor safety regulations, an effort will be made to propose optional rules, regulatory guides, and review practices. The advantages and disadvantages, values and impacts, and cost and benefit will be documented for each formulation. Supportive analyses of the interdependence of the regulations and of the impact of implementing changes will be performed to provide a comprehensive basis for agency and public consideration of the pros and cons of each approach.

5.13.5 Plan of Work As a Function of Time

The schedule for this element can be seen in Figure 5.13.

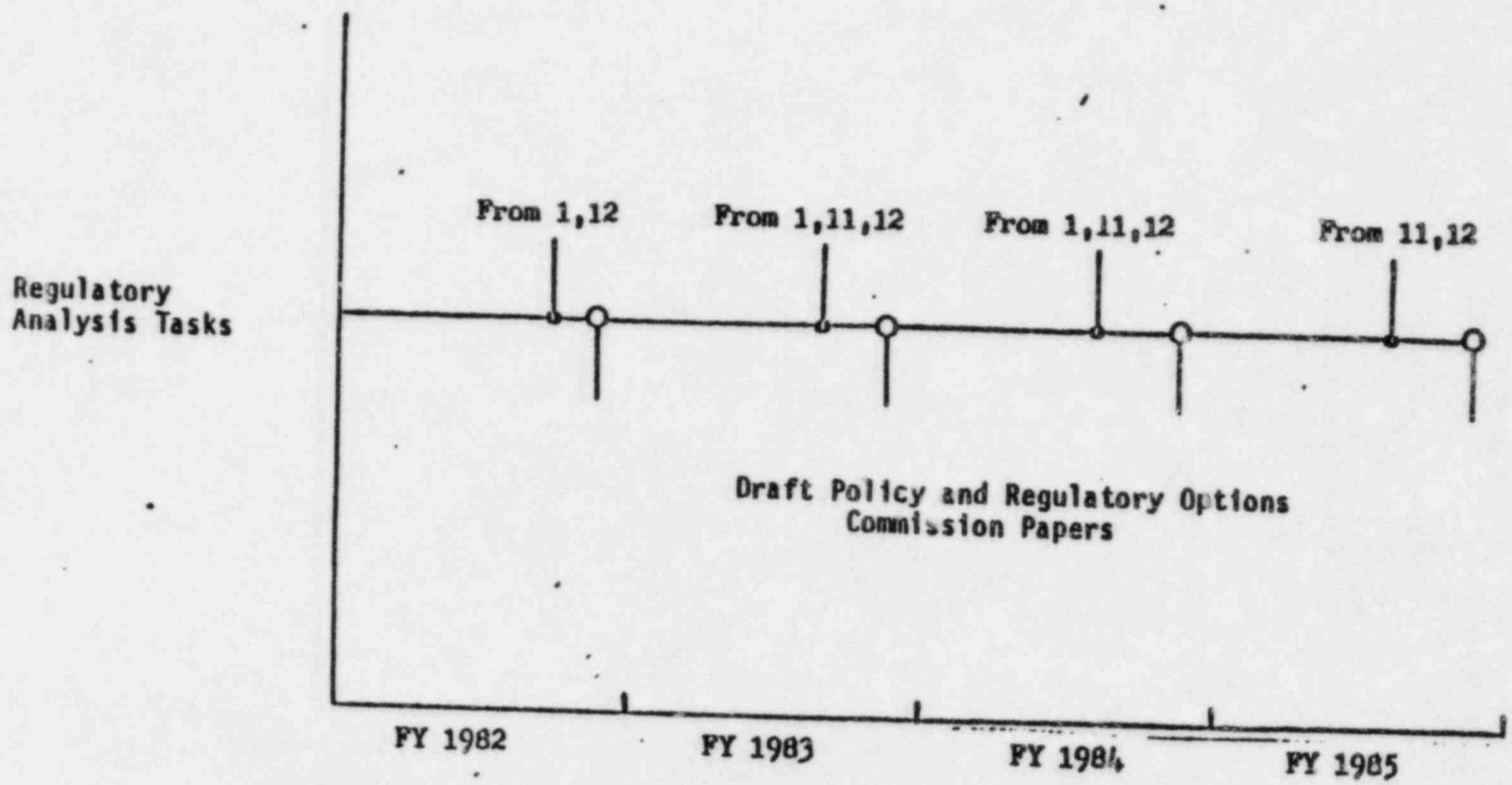


Figure 5.13 Regulatory Analysis and Standards Development

6. CONCLUSIONS

Presented in this report is a plan for an integrated program of research to establish a data base for policy and regulatory decisions regarding the treatment of accidents in nuclear power plants. The plan treats the aspects of Accident Prevention, Accident Management, and Accident Mitigation. In covering these aspects, the plan has sought to be consistent with the Commission's long-established policy of defense in depth. The need for a tightly integrated research program grows out of the wide range of uncertainties in the phenomena and methodology, as revealed by casework, most specifically casework on the Zion and Indian Point plants. That is, the integrated nature of the plan is as much a consequence of logical necessity as it is of policy. This integration poses some pitfalls as well as advantages.

The advantages stem from the finding that, if an even pace is maintained among the various program elements, interim results can be used to guide the development of regulatory decisions and to focus the program more narrowly on the needed technical information. Thus, some of the fruits of the program will be available in about 18 months so that the program will have the feature of being self-correcting and contracting in scope rather than expanding. Significant results are forecast by the end of 1983 with completion of most major tasks by 1985.

The pitfalls stem from the fact that some of the program elements are technically very advanced; thus success may be limited in some cases by lack of technological capability. In the experimental areas, this is most likely to be true of tests at the very high temperature characteristic of molten fuel. In the analytic areas, the picture is somewhat more sanguine except that the extension of PRA methodology into detailed design assessments related to severe accidents is an untried exercise. Similarly, the success of the Severe Core Damage Analysis Package (SCDAP) depends on the assumption that damage to fuel can be characterized by a small number of regimes denoted by just a few parameters.

Nevertheless, for the program to succeed, it is not essential that there be a uniform degree of precision in all data elements. Rather, the uncertainties among the competing aspects of value-impact analysis: Between Prevention and Management, on the one hand, and Mitigation, on the other, should be of about the same order. Naturally, the greater the uncertainty in the technical elements, the greater the uncertainty in the conclusion of the value-impact analysis. It is misleading to do an analysis of that which is precise versus that which is uncertain in order to determine the relative advantage. Trends that indicate less chance of precision in one area as compared with another can be examined early in the program and the scope of work re-adjusted to ensure that meaningful comparisons are achieved.

ENCLOSURE B

Supplemental

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