NUREG–1365

Revised Severe Accident Research Program Plan

FY 1990 – 1992

U.S. Nuclear Regulatory Commission

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Abstract

For the past 10 years, since the Three Mile Island accident, the NRC has sponsored an active research program on light-water-reactor severe accidents as part of a multifaceted approach to reactor safety.

This report describes the revised Severe Accident Research Program (SARP) and how the revisions are designed to provide confirmatory information and technical support to the NRC staff in implementing the staff's Integration Plan for Closure of Severe Accident Issues as described in SECY-88-147. The revised SARP addresses both the near-term research directed at providing a technical basis upon which decisions on important containment performance issues can be made and the long-term research needed to confirm and refine our understanding of severe accidents. In developing this plan, the staff recognized that the overall goal is to reduce the uncertainties in the source term sufficiently to enable the staff to make regulatory decisions on severe accident issues. However, the staff also recognized that for some issues it may not be practical to attempt to further reduce uncertainties, and some regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

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1. Introduction

In a memorandum dated July 8, 1988, the Director of the Office of Nuclear Regulatory Research charged the Division of Reactor and Plant Systems (now the Division of Systems Research) to develop "a detailed plan identifying individual goals, discrete products, and anticipated schedules for the Severe Accident Research Program (SARP) that clearly demonstrates the implementation of the SARP portion of the staff's Integration Plan for Closure of Severe Accident Issues (Integration Plan) (SECY-88-147). [Preparing such a plan will require] a detailed review of the current activity of the SARP against the framework of the Integration Plan and the detailed activities and schedules of the other five elements of the Integration Plan, [together with] an assessment of what redirecting/reprioritizing/rescheduling within the SARP may be needed to fulfill its portion of the Integration Plan and how and when such can be accomplished." This revised Severe Accident Research Program plan reflects the evolution of the severe accident research program and was developed in response to that July 8 memorandum. It will be used to guide the formulation and conduct of severe accident research, addressing in the near-term those issues pertinent to the implementation of the Integration Plan: viz., individual plant examination (IPE) and containment performance improvement (CPI) activities. The plan also will be used to guide the formulation and conduct of the program of long-term research needed to support the NRC's accident management activities and to confirm the Commission regulatory decision on severe accident issues. The plan provides a description of the general approach to the needed research. Naturally, many of the details associated with specific experiments or analyses have not yet been developed and can only be developed after the specific research is identified. Therefore, for each near-term issue, a detailed research plan will be developed, describing each experimental and analytical program and how these programs fit together and lead to a resolution of the issue. The plan may also give the impression that we are just now entering the arena of severe accident research; the facts are quite the contrary. A tremendous amount of severe accident research now exists, and this plan is intended as a needed step to the critical and focused reexamination of our further needs and the regulatory questions involved.

This plan is organized as follows. Section 2 presents the goals of the SARP and a sketch of its structure. In Section 3, the work directed toward achievement of the near-term goals articulated in Section 2 is described. Research dealing with the longer term goals is covered in Section 4. Appendix A to this plan provides background information on the existing SARP and the relationship of SARP to other elements of SECY-88-147, and Appendix B is an exposition of the role of scaling in the SARP. Estimated costs and significant milestones for implementing the revised SARP are presented in Table 1.

2. Goals

In developing the revised SARP plan, the staff recognized that the overall goal is to reduce the uncertainties in source term sufficiently to enable the staff to make regulatory decisions on severe accident issues. However, the staff also recognized that for some issues it may not be practical to attempt to further reduce uncertainties, and regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

As the Severe Accident Research Program represented in this plan is a goal-oriented program, it is critical that such goals be clearly stated at the outset. The staff expects to use safety goal policy and objectives, as appropriate, in determining what potential improvements in the technological base are needed for closure of severe accidents and in order to ensure that the safety of existing plants are reasonably consistent with the safety goals. The goals of the revised SARP are as follows:

1. Provide the technological base for assessing containment performance over the range of risk-significant core melt events.

2. Develop the capability to evaluate the effectiveness of generic containment performance improvements.

3. Provide an understanding of the range of phenomena exhibited by severe accidents that includes the impacts of generic accident management schemes.

4. Develop improved methods for assessing fission product behavior and availability for release in the event of containment failure at various phases of severe accident sequences.

Goals 1 and 2 above are primarily near term, while goals 3 and 4 are longer term objectives that aim for additional depth of understanding of both accident evolution and final consequences. These goals are consistent with NRC's Integration Plan for Closure of Severe Accident Issues (SECY-88-147) and the guidance provided in Generic Letter 88-20 for individual plant examinations. It should be noted that there are not two SARPs, a near-term and a long-term. Rather the revised SARP draws from and focuses specific tasks of the continuing program of severe accident research experiments and analyses to address the near-term implementation of SECY-88-147.

2.1 Near-Term Work

Subsequent to the Three Mile Island Unit 2 accident in 1979, the U.S. Nuclear Regulatory Commission

Table 1 SARP Milestones and Estimated Costs

Task	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-92)
Issue 1:	**Scaling**				
1	Scaling Analysis	Develop and apply severe	3/89	6/91	1800K
TOTAL	ISSUE 1				1800K
Issue 2:	**Depressurization and DCH**				
2.1	Containment Challenges Research	Assess both experimental and model development programs to determine the appropriateness of additional expenditure	6/89	12/89	300K
2.2	Natural Circulation	Confirm natural circulation calculations	9/89	9/90	1800K
2.3	DCH Scaled Experiments	Experiments to study effect of structures and influence of water in the reactor cavity and containment atmosphere and to verify analyses of Task 1	11/89	9/91	2500K
2.4	Depressurization (intentional)	Calculations exploring efficacy and consequences of possible methods of intentional depressurization	9/89	9/90	600K
2.5	RPV Bottom Head Failure	Analyses over a range of melt configurations. Confirmatory experiments	9/89	6/90	800K
2.6	Material at Lower Head at Time of Breach	Determine quantity, composition and timing of molten core arrival at bottom head	8/89	8/90	600K
2.7	Upgrade DCH models	Incorporate test results into DCH analysis model	9/89	4/90	600K
2.8	Depressurization Cost/Benefit Analysis	Provide bases upon which to recommend for or against depressurization	6/89	12/89	220K
TOTAL	ISSUE 2				7420K

*This task will be conducted under the accident management research. The allocated $ is for code validation/development.

Table 1 (Continued)

Task	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-92)
Issue 3:	**BWR Mark I Containment Shell Melttrough**				
3.1	Lower Plenum Study	Analytical/experimental studies to determine sensitivity of BWR RPV failure to characteristics of melt in lower plenum	12/89	9/90	1000K
3.2	Effect of Water on Melt Spreading	Analytical and experimental work scoping melt-water interactions	12/89	6/90	900K
3.3	Mark I Containment Shell Failure	Examination of past experiments and conduct of new experiments to verify expected heat transfer to drywell shell in Mark I under expected ex-vessel melts	12/89	6/90	600K
TOTAL	ISSUE 3				2500K
Issue 4:	**Adding Water to Degraded Cores**				
4.1	H_2 Generation Upon Reflood	Analysis of melt-water interaction, design of needed experiments	9/89	6/90	1000K
4.2	Stability of Crust Crucible	Analysis to examine formation and stability (mechanical and thermal) of crust crucible	9/89	1/90	800K
4.3	Recriticality Investigation	Analysis to determine the extent of recriticality concern	9/89	9/89	200K*
TOTAL	ISSUE 4				2000K
Issue 5:	**Use and Status of Severe Accident Codes**				
5	Role and Future Development of Severe Accident Codes	An in-depth assessment of the entire area of NRC-sponsored severe accident codes development	9/89	3/90	855K
TOTAL	ISSUE 5				855K

*This task is being done under the accident management research. The allocated $ is for defining different critical masses expected for degraded BWR core.

Table 1 (Continued)

Long-Term Goals (Confirmatory program with projection of 3 years)			
Task	**Title**	**Description**	**Expected Cost (FY90-92)**
	Role of Uncertainties in Regulatory Decision and Future Research	Seek further depth of understanding of accident phenomenology, validate analysis tools. The issues are described below.	
Long–Term Research			
Issue L1	Modeling Severe Accidents		16,000K
Issue L2	In-Vessel Core Melt Progression and Hydrogen Generation		17,000K
Issue L3	Hydrogen Transport and Combustion		3,000K
Issue L4	Fuel-Coolant Interactions		3,000K
Issue L5	Molten Core-Concrete Interaction		6,668K
Issue L6	Fission Product Behavior and Transport		4,538K
Issue L7	Fundamental Data Needs		500K
TOTAL	LONG TERM		50,706K
TOTAL			65,281K

undertook a broadly based research program to develop an understanding of severe accident behavior. Over the past decade, major experimental and model (code) development programs have been performed that provide a greatly improved understanding of severe accident phenomena, reflected in an ability to model those phenomena. Now, as the Commission is preparing to close on severe accident regulatory questions, the immediate priority of the NRC's SARP is to support the closure process. That process* is displayed schematically in Figure 1. Hence, over the next 3 years, a significant portion of the severe accident research effort will be directed toward issues that relate to the dominant topic area of the Integration Plan, viz., the accident sequences that lead to early containment failure: direct containment heating (DCH), BWR Mark I containment shell meltthrough, molten fuel-coolant interactions in BWR Mark II and Mark III containments, and hydrogen detonation in BWR Mark III and PWR ice condenser containments. Note that the hydrogen behavior issues for accidents in which the core is degraded but is not into the severe accident domain have been extensively investigated, which resulted in the resolution of USI A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." The additional hydrogen research discussed in this revised SARP is intended to address the hydrogen threat from severely damaged cores and will serve to reduce uncertainties in the threat to containment integrity resulting from various hydrogen combustion mode(s) for a wide range of accident conditions. In the near term, severe accident research will also support the resolution of broader, more general questions regarding the response of various containment designs to severe accidents. Task descriptions have been developed for the near-term work on these issues and are presented in Section 3 of this plan. The majority of the near-term severe accident issues are considered relevant to evolutionary LWRs currently under review by the NRC. Therefore, these near-term efforts are equally important to evolutionary LWRs. We are currently not aware of any features of the evolutionary LWRs under development that are unique with respect to severe accidents and require special research programs. As such, no funding has been allocated for work in this area. However, we will keep abreast with the development of these evolutionary LWRs and notify the Commission if our current perception changes.

*In SECY-88-147, the staff identified the steps that each licensee is expected to take to achieve a closure on severe accidents for its plant, namely:

1 - Completion of the individual plant examination (IPE) and identification of potential improvements,
2 - Development and implementation of a framework for an accident management program that can accommodate new information as it is developed, and
3 - Implementation of any generic containment performance improvements with respect to severe accidents.

2.2 Long-Term Work

Achievement of the longer term goals demands a broadly based severe accident research program. To be responsive to regulatory needs, particularly confirmation of closure of severe accident issues, that program must both explore severe accident phenomena and develop the methodologies appropriate to quantitative assessment of severe accident risks. The plan to guide such a program must reflect a balance among the risk importance of phenomena being investigated, the desire for technical completeness, the expected likelihood that the research will result in a significant reduction in the uncertainty of risk, and the need to promote the maintenance of technical expertise and analytical capabilities. The features of the SARP that address the long-term goals are presented in Section 4 of this plan. Figure 2 depicts the elements of this revised SARP and the relationships among them.

2.3 Results of Previous Work

Review of information available from previous and ongoing research, including that sponsored by U.S. industry or foreign organizations, is being undertaken by the staff and expected to be completed by the end of FY1989. The purpose of this review is to identify and redirect the research program as appropriate to ensure that the needed information to reach closure or resolution of severe accident issues important for regulatory decisions is developed. This review will also ensure that the various research projects are consistent and well integrated among themselves with a common goal of ultimately leading to closure of severe accident issues and that the long-term confirmatory research is properly focused.

3. Meeting Near-Term Goals

3.1 Perspectives

This section of the SARP plan presents those issues of immediate priority that need to be and can be addressed in the next few years in order to achieve the near-term goals set forth in Section 2. These are the issues that are pertinent to the NRC's programs of individual plant examinations and containment performance improvements (direct containment heating (DCH) and hydrogen transport and combustion, issue 2, and BWR Mark I containment shell meltthrough, issue 3). The research to be performed to address those issues is also presented. This research is primarily directed to assessing containment performance under severe accident conditions and is constructed to enable as definitive a judgment as is possible about the potential threats and likelihood of early containment failure in the event of a severe accident given the present paucity of (or sparse) data describing core melt progression. Also included in this section is a discussion of the

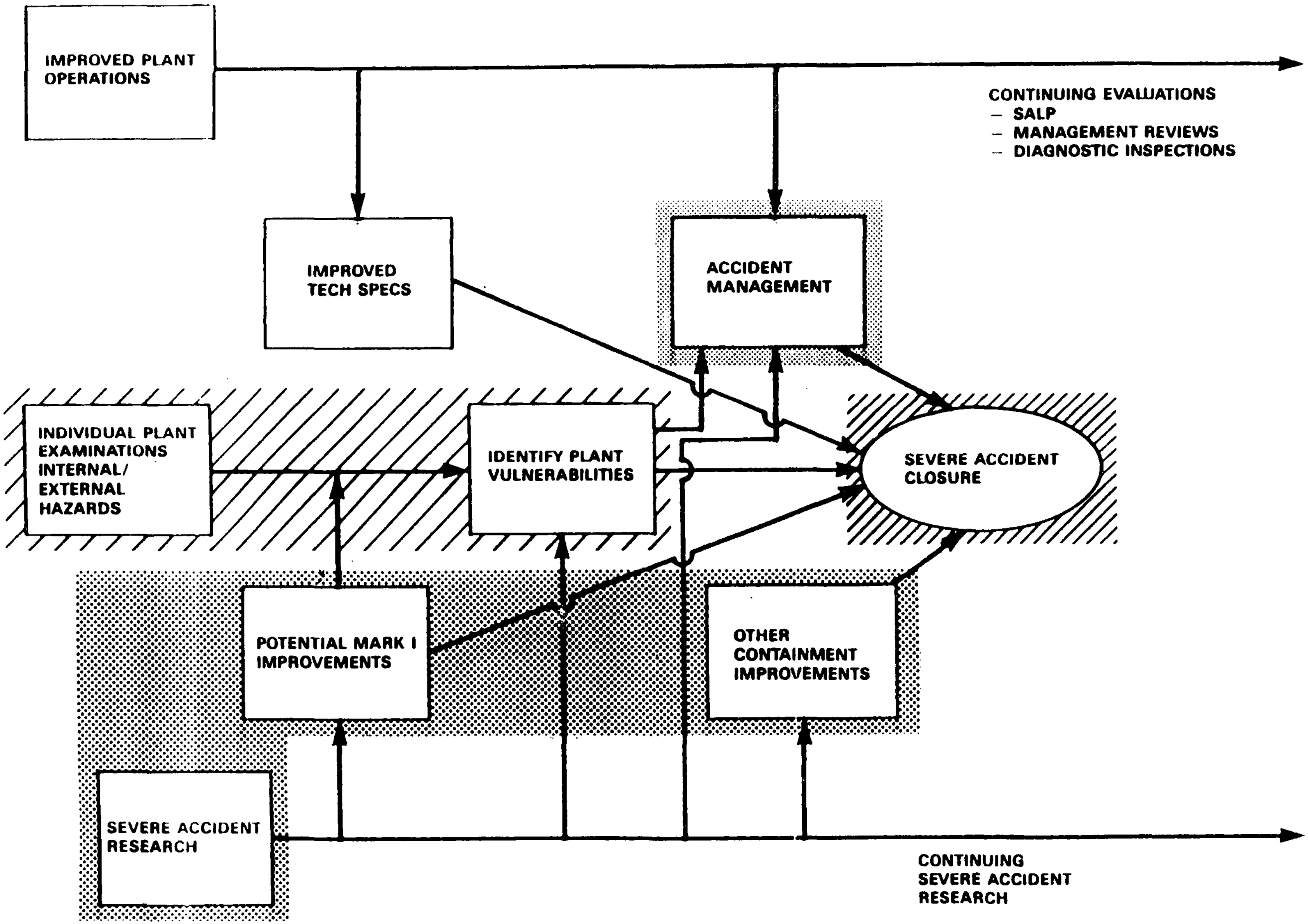

Figure 1 Severe Accident Program—Schematic

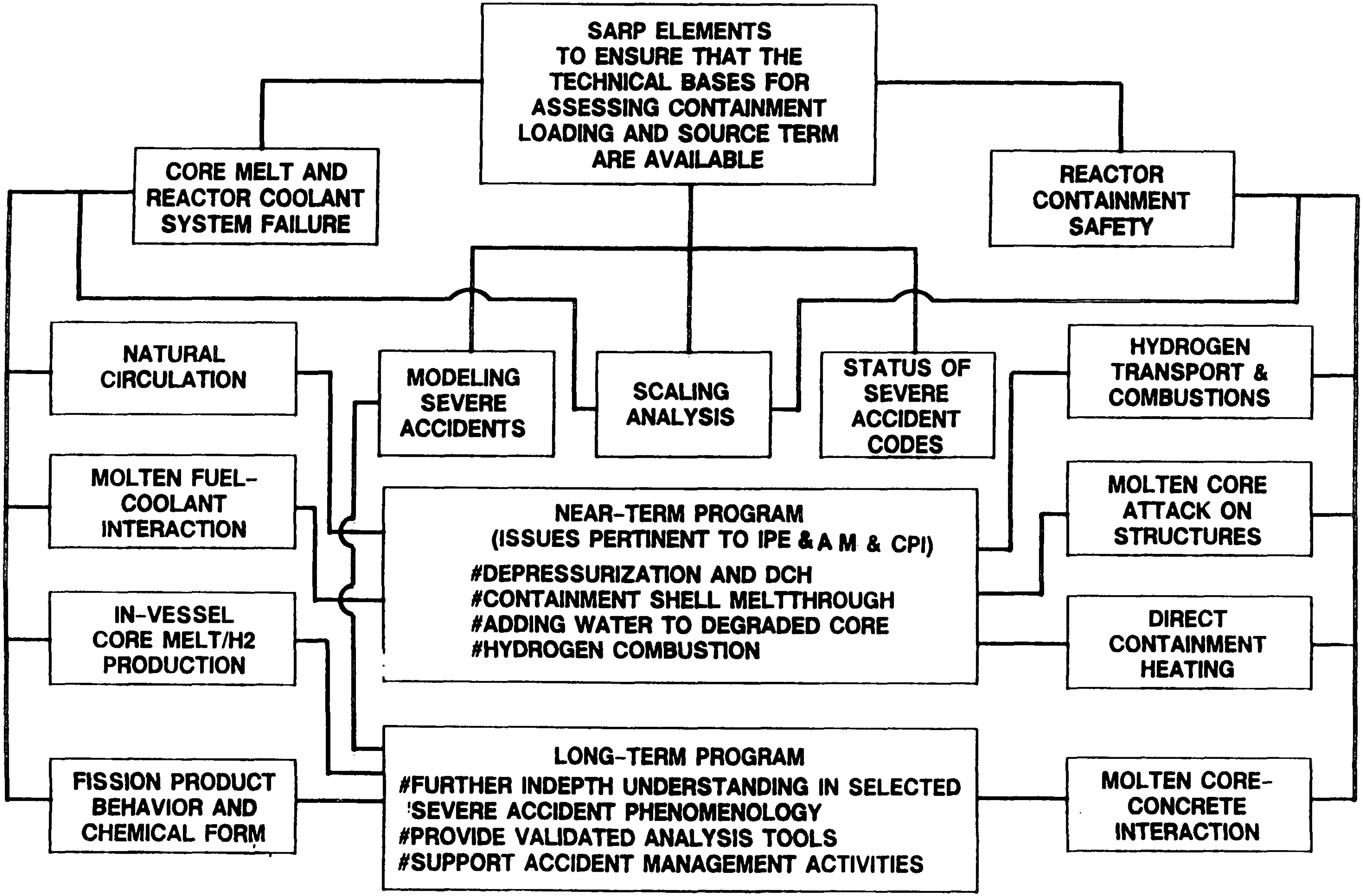

Figure 2 Revised SARP Near- and Long-Term Programs

issue of scaling of severe accident experiments and analyses (issue 1). Obviously, the question of verisimilitude in severe accident research is critical, and scaling is one element of the same. Whether the correct physical and chemical processes are being investigated is another and is appropriately dealt with on a case-by-case basis. But scaling as an issue is common to all areas of severe accident research since full-scale experiments simply are not feasible. For this reason, a systematic examination of scaling of severe accident experiments and analyses is included among the key elements of the near-term research. This element is not intended to constrain closure expected of the other near-term issues herein but is expected to be of benefit to decisions made on recommending further experiments over the near- and long-term periods addressed in this plan.

In addition, this section contains a discussion of the issue of molten fuel-coolant interactions (issue 4) and a discussion of the issue of the use and status of severe accident codes (issue 5). The molten fuel-coolant interaction issue is included here because it is directly relevant to considerations of intentional reactor vessel depressurization to cope with DCH, as well as low reactor coolant system pressure accident sequences in general. Specifically, the question of what happens when water is added to molten fuel (in-vessel or ex-vessel) appears to be significant to understanding accident management and some severe accident sequences. How to answer that question is not immediately obvious, and some initial work needs to be done to allow development of a rational approach to the research. As for the issue of codes, it is recognized that code development is integral to the achievement of long-term goals, since the understanding of severe accident phenomenology ultimately is reflected in those calculational tools. However, achieving the near-term goals will necessitate reasoned judgments based essentially upon our present understanding of severe accidents and an assessment of just how well present codes describe the key behavior of severe accidents. The confidence that can be placed in the severe accident codes also directly translates into confidence that can be vested in the near-term judgments about containment performance. For this reason, the code assessment issue is included in this section.

3.2 Severe Accident Phenomena — Core Melt Accidents

Before the issues and research tasks to address them are discussed, a general description of the important phenomena associated with core melt accidents to be dealt with in the near term is presented. This is done to provide a perspective and context for the issues and tasks.

In light water reactors, core melt accidents can occur either at low reactor coolant system pressure or at high reactor coolant system pressure. At low pressures, the core heats up more or less independently from the rest of the primary system, and the process is accelerated by metal-water reactions. At high pressures, energy redistribution by steam from natural circulation may be significant, transferring core heat to the remainder of the system, including upper vessel internals and, for PWRs, steam generators. This transfer of energy affects the progress of the accident in several ways. It delays somewhat the onset of gross core degradation. It affects the steam availability in the metal-water reaction process, tending to limit hydrogen generation. It affects radionuclide release and transport within the primary system. Perhaps most importantly, it also raises the possibility of competing high temperature and stress-related failures of the primary system pressure boundary (e.g., RCS piping, steam generator tubes) in PWRs with attendant depressurization, hence avoiding the possible ejection of molten core material from the reactor vessel while under high pressure. Current understanding suggests that for BWRs natural circulation is restricted by the fuel assembly wall channel boxes. Although natural circulation may affect the core melt progression, failure of the reactor coolant system due to excessive heating from high-temperature naturally circulating gases would not be likely. However, because of the enhanced reliability of the automatic depressurization system (ADS) suggested from the containment performance improvement program, high-pressure scenarios are not likely to be dominant contributors to early containment failure in risk assessment of BWRs. Hence, high-pressure scenarios in BWRs are not being addressed at this time.

Although the failure of the reactor coolant system (RCS) by natural circulation appears to be possible, the ability to predict its occurrence has not been established to the point that confident conclusions regarding the likelihood of RCS failure can be made. Moreover, while analyses to determine the extent to which natural circulation occurred in the TMI-2 accident are under way, it must be noted that no evidence of high temperatures in the range of interest to cause failures in the primary system components has yet been seen. Thus, competing RCS pressure boundary failures from high temperature cannot be presumed with a high degree of confidence, and scenarios involving gross core degradation or core melt at high or intermediate RCS pressure must be considered. Since such high-pressure PWR scenarios have the potential to produce ejections of molten core material into the lower containment following lower vessel head failure, a variety of concerns regarding the possibility of containment overpressurization have been raised, in particular, direct containment heating. In fact, the high-pressure scenario seems to influence risk in terms of the variety of mechanisms with potential to overpressurize the containment and thus importantly contributes to the uncertainties in predicted risk. In addition, core melt accidents at low RCS pressures that lead to early containment failure have been identified as potentially important to risk, in particular, BWR Mark I containment shell meltthrough.

The following observations are intended to supplement the above general perspectives:

1. At this time, competing primary boundary failure locations, size, and timing are highly uncertain, particularly considering the role of relocated fission products and the possibility of loss of integrity of steam generator tubes.

2. Core slump at high pressures may yield significant fuel-coolant mixing. Triggering of massive steam explosions at such high pressures is generally considered highly unlikely by the experts, but the potential driving forces associated with such core slump interactions may not be altogether ignored in the process of understanding fission product relocation and the potential for altering the accident progression.

3. Natural circulation may, for example, cause a weakening of the upper internals and upper vessel head (e.g., seals), with the concomitant result that competing relief paths may be formed that influence both the fission product relocation and the potential for direct containment heating.

4. Lower vessel head specific failure modes and the timing remain highly uncertain. Local failures of instrument guide tubes could occur. However, melt attack of the vessel lower head forging (in the absence of local penetrations) could lead either to a local failure of the forging or to creep rupture of a large portion of the lower vessel head when heated to 600° to 700°C. Large failures at high RCS pressure can potentially create thrust loads and threaten failure, for example, through overstress or tearing at piping junctures.

5. At high pressure, a large coherent failure of the lower head would likely be accompanied by prompt expulsion of melt, steam, and hydrogen. High-pressure expulsion has been postulated to lead to direct heating of the containment atmosphere and an increased potential for hydrogen combustion and perhaps even detonation.

6. The ex-vessel course of accidents in which reactor vessel breach occurs with low pressure in the reactor coolant system is highly dependent on the mass, thermal, and chemical properties of the melt upon exit from the reactor pressure vessel. Direct attack of the Mark I containment shell; molten fuel-coolant interactions in the BWR Mark II and Mark III containments; and hydrogen combustion in BWR Mark III containments are examples of where low system pressure scenarios are likely to dominate the probability of a severe accident but also give rise to early containment failure threats.

3.3 Issues and Tasks

Issue 1—Scaling

Full-size severe accident experiments are impractical or simply not feasible for many reasons, particularly costs. Hence, severe accident research must rely on small-scale experiments that are carefully crafted so that the conclusions drawn from those experiments about the phenomenology of severe accidents can be appropriately applied to predict severe accident progression in typical nuclear power reactors. The confidence in this application is greatly enhanced through scaling analyses. A key element of the SARP, therefore, is a scaling methodology for severe accident experiments. In those cases where an acceptable methodology can be developed, it will be applied to relevant experimental programs. As necessary, recommendations regarding revisions to experimental programs will be made, and needs for new facilities will be addressed in terms of the regulatory value in pursuing this research. A useful scaling methodology should be capable of providing the following:

1. A scaling rationale and similarity criteria.

2. A procedure for conducting comprehensive review of facility design, test specifications, and results.

3. A measure or index to indicate the applicability of correlations or models based on test data from subscale facilities to full-scale nuclear power plant conditions.

4. Quantification of the effects of scale distortion.

5. Quantification of the effects associated with extrapolating correlations or models beyond their data base. This information is needed for quantifying code uncertainty.

Although scaling analyses have been performed for some of the severe accident research experiments, there as yet is no systematic application of a scaling methodology that would allow confident extrapolation of observations made in intentionally scaled experiments to severe reactor accidents. The work to be done is to first determine whether a scaling rationale and similarity criteria for severe accident research can be developed to provide assurance that the correlations or models developed can be properly scaled to reactor conditions. Second, if such a methodology is developed, a systematic application of the same will be made to appropriate experiments and computer codes carried out under the SARP.

Task 1 Develop and apply a general severe accident scaling methodology.

Research Approach

Scaling methods will be reviewed. This review will be of scaling methods in general and methods used in the experiments of the SARP to date. From examination of the nature of the phenomena being studied, the experiments being conducted to study those phenomena, and the review above, scaling methodology will be developed for the SARP. The efficacy of the methodology will be tested initially by applying it to the direct containment heating experiments that have been conducted in the SARP to date.

This task will not address directly whether the correct phenomena are being investigated in a particular experiment—that is done for each task throughout the SARP—but rather whether the experiment will indeed yield information about the phenomena being investigated relevant to behavior at the scale of an actual reactor.

Usables and Use

To the degree that this work is successful, the SARP will have a scaling methodology that can be applied to the experiments of the SARP to provide a more rigorous and systematic link between experiment and accident phenomena than is present today. The application of the methodology will help in the planning and conduct of the experiments directed at the longer-term goals. In the near term, application of the methodology should help establish the level of confidence that can be placed in the present understanding of containment performance from scaled experiments and code analyses performed to date.

Issue 2—Depressurization and Direct Containment Heating

Present understanding of the in-vessel progression of a PWR core degradation accident is limited but suggests the formation of melting-solidifying fronts as the molten core materials "candle down" into the lower, colder portions of the fuel bundles and solidify. This "frozen crust" of previously molten core material may form in a layer that inhibits steam flow from the lower plenum. Unsupported fuel pellet stacks may collapse onto the crust and melt. As this process proceeds, a crucible-shaped crust supporting a molten mass of core material may form in the central region of the core. Failure of this crust at a radial location ("sidewall") is expected from considering thermal convection of the melt within the crucible. In fact, this is what was concluded to have occurred at TMI-2 upon examination of the reactor core. Upon crust failure, molten core material would flow downward, possibly ablating through the core shroud and flowing onto the flow distribution plate, eventually reaching the lower head. However, very little experimental information on this process exists, and computer code predictions are at best uncertain. Hence, prior to dismissing crust failure on the bottom portion of the crucible as not a plausible failure mode, this issue will be assessed. The main difference between these two failure modes seems to be the amount of molten material that pours into the lower head and is ultimately available to be released to the containment. Our current understanding indicates anywhere from around 25 to 60 percent of the core could relocate to the bottom head upon crucible failure. Because of the difficulty of conducting appropriate tests to better characterize these phenomena, questions associated with core relocation are likely to remain highly uncertain in the near term, although modeling of these processes will continue under the SARP. Thus, the question of lower head failure mode can be approached by considering a sufficiently broad range of debris quantities, composition, temperatures, and relocation rates. The TMI lower head inspection plays an important role in understanding the relocation process and the threat to the lower head integrity. On the other hand, depressurization of the RCS, either by structural failure of the pressure boundary (e.g., piping, steam generator tubes) resulting from natural circulation or deliberate depressurization by operator action, may alleviate concerns over direct containment heating (DCH). In this regard, any analysis with respect to depressurization would be assessed in terms of the current state of operator training, including information available to the operator (instrumentation), and it would include the dominant accident sequences anticipated for a given plant.

As part of the development of the reactor risk assessment document (NUREG-1150), the DCH issue was presented to an expert panel that included experienced severe accident analysts. Monte Carlo analysis that integrated all the pertinent phenomena was performed. The resultant likelihood of containment failure was found to be small, and the chance that containment would survive the loads associated with DCH is much greater than had been originally estimated and reported in the first draft of NUREG-1150. The experts based their judgments, including the likelihood of RCS failure by natural circulation leading to RCS depressurization, on the current body of research evidence.

Consistent with the above, in the near term the SARP will address the DCH issue by a two-pronged approach:

1. Before we can rigorously reach a conclusion on the benefit of intentional depressurization of the primary systems of PWRs, an improved understanding of the challenge to containment integrity must be obtained. The existing research efforts will be critically examined with respect to the following:

   a. What is the nature and character of the information most likely to be provided on high-pressure core melt accidents in the next several years? Are continued commitment of time and resources likely to result in a commensurate advancement of understanding of DCH phenomena?

b. Are the existing research programs addressing DCH optimized with respect to providing the best information for closure of the DCH question in the near term?

2. The efficacy of PWR depressurization will be examined. In particular, both beneficial and detrimental aspects will be explored. Review of relevant emergency procedures, dominant accident scenarios, and timing of required operator actions to depressurize also would be included in this examination.

Using the results from 1 and 2 above, an updated estimate of the risk associated with high-pressure core melt accidents will be made, along with an estimate of the risk reduction that would be obtained by intentional depressurization, taking into account any increase in risk associated with depressurization. Based on a careful weighing of the net benefits of depressurization, a recommended course of action will be proposed.

Some of the key questions that the SARP will attempt to answer in the near term in order to make the above determinations are:

1. What is the likelihood that the RCS will fail by natural circulation prior to lower head failure? If so, are these failures of much lesser concern to overall risk? (Task 2.2)

2. Is there a low-pressure cutoff below which there is no DCH threat? (Task 2.3)

3. If so, will this pressure be reached by natural circulation-induced failure of the RCS, or through operator action to depressurize, or both? (Tasks 2.1 and 2.5)

4. If operator action is necessary, is there time available for this action? Given that operator action is to be taken, what are the hardware and procedure specifications that would enable successful depressurization actions? (Tasks 2.3 and 2.4)

5. Are there adverse consequences to early depressurization? If so, what are these, and what is their potential significance in terms of causing either earlier core damage or earlier containment failure? (Tasks 2.4 and 2.8)

6. What is the nature of the DCH threat, and what mechanisms (e.g., sprays) and configurations exist ex-vessel that will mitigate or eliminate it? (Tasks 2.4–2.7)

Logically an answer to this last question should precede the rest. However, even with DCH apparently presenting much less of a risk than previously believed, there is not yet sufficient confidence in this belief that it warrants delay in seeking answers from the other relevant tasks, especially in light of the 3-year horizon of the Integration Plan.

Task 2.1 Evaluate current research program addressing high-pressure melt ejection challenges to containments.

Research Approach

Current research programs, both experimental and model development programs related to DCH and hydrogen production during high-pressure melt ejection (HPME), will be reviewed and assessed to quantify the current level of uncertainties to determine if any of the questions raised in the following tasks have already been answered. In addition, the assessment will determine the nature and character of information likely to result from these programs over the next 3 years. Progress made to date, current research topics, major uncertainties, and methods being used, both experimental and analytical, will be examined. This task will be carried out prior to the planning and conduct of any new experiments or analyses.

Usables and Use

Based upon the results of the above assessment, a judgment will be made as to whether a significant increase in understanding the threats to containment from high RCS pressure accidents will be achieved in the next 3 years. In particular, the question of whether a significant reduction in uncertainties will be achieved only with a significant additional expenditure of time and resources will be addressed.

Task 2.2 Assess the likelihood of RCS structural failure by natural circulation.

Research Approach

Calculations of RCS failure by natural circulation will be scrutinized and alternative models and assumptions examined. Comparisons of calculations with available experiments (Westinghouse) will be made to identify potential weaknesses in both experiments and models, particularly with regard to scaling. An assessment will be made as to why there were only minor natural circulation effects in the upper vessel structure and none in the hot leg piping at TMI.

High RCS pressure PWR sequences will be selected by initiating event, characteristics of the early stages of the sequence, and potential for evolving to intermediate or low RCS pressure sequences (e.g., potential failure of a safety valve in a station blackout during core heatup). This selection should include loop seal clearing and pump seal failures. Codes predicting energy redistribution due to natural circulation will be applied to these sequences and will be verified, validated, and examined for underlying assumptions that could affect the results. Analyses and comparisons of the selected sequences as to likelihood, location, and time of failure of the RCS boundary will be made. The effects of fission product deposition and revaporization as an additional source of thermal energy during natural circulation will be considered.

Usables and Use

The result of this task will be an estimate of the likelihood of RCS failure from natural circulation in high RCS pressure scenarios for typical primary system geometries. It is possible that this item will alleviate concern with the threat of direct containment heating.

Task 2.3 Investigate the influence of cavity and containment compartment structures on DCH, low-pressure cutoff for DCH, and hydrogen production from HPME.

Research Approach

The scoping calculations performed for NUREG-1150 to estimate the DCH load from the dispersal of melt into the containment correlated DCH load with melt quantity, ejection pressure, and ejection rate. The results of this analysis identified the key unknowns in the prediction of DCH containment loads from melt dispersal that need to be determined by experiment—particularly, the effect of lower containment compartment geometries; the effect of water (co-dispersal with the corium or added by spray) on reducing the magnitude of DCH pressure; and the production of hydrogen by HPME.

Experiments at different scales (using existing facilities insofar as possible) with high temperature, thermitically generated melt and steam as a pressurizing fluid will be performed. Measured quantities will be identified based on the scaling analysis of Task 1. Variables believed at this time to affect the magnitude of the challenge to the containment are melt mass expelled from the cavity and its distribution within the containment, containment atmosphere pressure and temperature, melt metal content of the dispersed melt heat transfer rates to structures, and hydrogen production in the dispersing melt-steam mixture and its transport and combustion in the containment.

Usables and Use

Direct comparisons of containment pressurization observed in tests at different scales (existing facilities) provide insight on the dominant factors affecting the scaling up of the results. The data may be used to estimate melt entrainment as a function of quantity of melt and ejection rate. The melt entrainment will then be used to calculate at what pressure DCH from core ejection does not threaten the containment integrity (low-pressure cutoff).

Task 2.4 Explore the feasibility of intentional RCS depressurization over the relevant spectra of PWR severe accident scenarios.

Research Approach

The overall framework for analyses should be based on a comprehensive treatment of representative scenarios, including possible perturbations by operator actions (based on the current state of operator training, Emergency Procedures or Functional Restoration Guidelines, and timing for when the operator is required to depressurize) and equipment failures that may have a bearing on the issues addressed here. The research should consider, but not be limited to, the normally available power-operated relief valves (PORVs) as a depressurization device. One such strategy is to decrease the system pressure sufficiently to allow the accumulators to dump prior to significant cladding oxidation, so that gross fuel degradation is approached at a pressure that will not threaten the containment integrity should the vessel lower head fail. The approach is essentially analytic, using thermal-hydraulic codes such as RELAP5 or TRAC for the most part, although selective use of core degradation codes may also prove useful. The use of models in these codes that have not been tested in circumstances peculiar to this application, e.g., reflooding of a highly overheated core would be scrutinized to facilitate judgment as to the meaningfulness of the results. The research will examine both beneficial and detrimental effects of early intentional depressurization.

Usables and Use

The results of this work will be a defined relationship between depressurization strategies and final system pressure at the time of gross fuel degradation, or the time of lower head failure, and an identification of any detrimental effects of intentional early RCS depressurization.

Task 2.5 Determine the mode of bottom head failure of the reactor pressure vessel in a high RCS pressure sequence.

Research Approach

Assess and determine the likely modes of PWR pressure vessel bottom head failure considering the reasonable range (Task 2.6) of quantities, composition, and timing of core melt that is arriving on the bottom head. From the above assessment, identify those experiments that may need to be conducted to determine the values of parameters that are keys to the assessments and/or confirm those assessments. Application of the scaling methodology (Task 1) will be made in devising and interpreting the results of the experiments.

The TMI lower head inspection plays an important role in understanding why vessel failure did not occur for this accident. It is not sufficient to look for damage to the lower head. It is essential that the debris on the lower head be carefully characterized in order to interpret the observed condition of the lower head.

Usables and Use

This work will result in an understanding of the sensitivity of reactor pressure vessel (RPV) lower head failure mode to the quantity, composition, and timing of arrival of molten corium on the lower head. This understanding may be quantified in the form of system level codes that predict the mode and timing of RPV failure in a manner consistent with the predicted melt progression and thermal-mechanical loading of the RPV. The results of this task will contribute to the work of Task 2.3.

Task 2.6 Determine the likely range of quantity, composition, and timing of molten core material arriving on the bottom head of a PWR during a core melt accident at high and intermediate RCS pressures.

Research Approach

Review and assess existing experiments and associated analyses to ensure that the analysis tools represent the important phenomena. A series of core degradation and melt relocation calculations will then be made using boundary conditions and initial conditions that span the range of core melt sequences. For each set of conditions considered, the calculations should yield the quantity, composition (proportion of metals and oxides), temperatures, and timing (rate of arrival) of melt reaching the bottom head. Based on these calculations, best estimates will be made on the characteristics of melt reaching the bottom head of a PWR in high and medium RCS pressure core melt accidents for assessing DCH. Low RCS pressure conditions will be used in evaluating the potential effects of depressurization, including the effect of molten fuel-coolant interaction on core melt progression.

Usables and Use

The result of this work will be the most likely characteristics of melt to be considered in addressing PWR bottom head failure (Task 2.5).

Task 2.7 Use the results obtained in Task 2.3 to upgrade DCH models.

Research Approach

This task should proceed only if the results from the scaling program (Task 1) and Task 2.3 indicate that further analytical developments are necessary for calculation of DCH loading and closure of this issue. Analysis may be used for extracting detailed information from experiments that can then be used to develop models that sufficiently represent plant-specific geometries. The suitability of the models will be affirmed by comparisons of calculations to the results of experiments. The developed DCH models can then be used to predict containment pressure due to direct heating phenomena in reactor accidents as well as the mitigation of these phenomena by containment pressure control devices and procedures.

Usables and Use

This work will result in DCH models that will be applicable to the evaluations of containment performance, the review of individual plant examination submittals, and the assessment of the efficacy of depressurization and containment pressure control strategies on the magnitude of DCH containment loading.

Task 2.8 Depressurization cost/benefit analysis.

Research Approach

The results of Tasks 2.1 through 2.7, and the results of Tasks 4.1 through 4.3 dealing with fuel-coolant interactions at low system pressures, will be considered in performing analyses of intentional PWR system depressurization as a mechanism to avoid DCH. The analyses will be focused on assessing the efficacy of depressurization to bring about a net benefit to safety for important core melt scenarios (e.g., station blackout and high-pressure conditions associated with loss of feedwater, and small LOCA scenarios). The benefit is the estimated reduction in risk of early containment failure from high-pressure ejection of molten core materials into the containment. Other factors to be considered include equipment costs needed for intentional PWR system depressurization and also any detrimental effects that may result from intentional depressurization. It is recognized that the results of Task 2.2 (Natural Circulation) may indicate that the RCS will fail and depressurize without operator action. If such is the case, whether to continue this task will be reviewed.

Usables and Use

The result of this task will provide additional bases upon which to recommend for or against requiring depressurization of PWRs during core melt accidents to avoid DCH.

Issue 3—BWR Mark I Containment Shell Meltthrough

An accident sequence leading to early containment failure has been postulated for BWR Mark I containments. This sequence involves the deposition of molten core onto the drywell concrete floor at the time of vessel breach and the subsequent spread of the melt to, and its contact with, the drywell steel wall that is the containment boundary, ultimately causing failure of the wall. Some experimental work has suggested that the addition of water in the drywell could provide a mitigative effect by causing fragmentation and quenching of the melt before it reaches the wall or by providing significant cooling of the melt layer. The heat transfer processes from the melt to the wall, the melt-concrete interactions, and the heat removal processes from the melt and from the wall are not completely understood, although some work has been done at Sandia National Laboratories on heat transfer from steel melts to steel structures and on melt-concrete interactions. There have also been experiments on bubbling heat transfer and on melt spreading at Brookhaven National Laboratory, using simulants, and experiments related to melt spreading and quenching have been conducted by Fauske and Associates, Inc. In addressing this issue in the near term, the literature will be carefully reviewed to clearly identify the critical questions to be answered with respect to Mark I shell failure and the capability of the existing data to answer those questions before any new research efforts are begun.

One such question that has been identified is the sensitivity of the time and mode of reactor vessel lower head breach to the quantity, composition, and timing of arrival of molten core on the bottom head of a BWR. The core melt of a BWR may proceed much differently from that of a PWR. For example, one possibility that has been suggested is that the channel box and control blade geometry of a BWR suggests that stable crusts supporting the melt may not form (as in TMI) and that molten fuel will more or less continuously flow onto the lower head. Further, the lower head volume of a BWR is larger than that of a PWR and is densely occupied by the control rod drive structures, and this may result in a different failure process (e.g., less coherent failure) of the bottom head from that of a PWR.

It has been postulated that a BWR core melt progression proceeds by gradually accumulating fully quenched debris in the spaces between the control rod guides in the lower head. Another postulated progression is similar to the TMI-2 accident with melt relocating into the water-filled lower head. In either case, following dryout, debris heatup will be accompanied by lower head heatup and eventual failure of the bottom head. At this time, the composition and characteristics of the debris arriving on the drywell floor beneath the reactor vessel is uncertain. The debris could range from mostly solid oxidic with low superheat to mostly molten metallic with high superheat. The nature of the debris exiting the failed bottom head of the BWR can determine the subsequent course of the accident. However, the bottom head itself may play a significant role in determining the characteristics of that debris. In light of the above, the SARP will address the questions of how sensitive (or insensitive) to the quantity, composition, temperature, and timing of melt arrival is the failure mode of the bottom head of a BWR and to what extent are the hydrodynamic and thermal properties of melt arriving on the drywell floor determined by the mode of bottom head failure.

The BWR Mark I containment shell meltthrough issue, like the DCH issue, was presented to a panel of experienced severe accident analysts who based their judgment on the current body of evidence. The panel was equally divided as to whether failure would occur or would not. When the drywell floor is water covered, the experts were more confident that failure would be avoided. Similarly, when the floor is dry and superheat is high, the experts had less confidence that the containment shell would remain intact. The panel concluded that the degree of belief regarding containment shell meltthrough is 0.33 or 0.87; the lowest failure probability corresponds to cases in which water is assumed to cover the drywell floor; and the highest corresponds to cases in which the drywell floor is dry and debris flow rate, debris superheat, and debris unoxidized metal content are all high. Nevertheless, it was generally agreed that the composition and temperature of the debris exiting the reactor pressure vessel and the presence of water on the drywell floor are important variables in estimates of containment shell meltthrough probability. This issue is, of course, of interest where venting might be accomplished after a severe core damage accident, but it does not bear on the recognized efficacy of venting prior to such an accident in order to prevent the same.

Specific questions that the SARP will attempt to answer in the near term are the following:

1. What is the relationship of the BWR bottom head failure mode to variations in quantity, composition, temperature, and timing of arrival of the melt on the bottom head? (Task 3.1)

2. What is the effect of water on the drywell floor when the melt pours out from the pressure vessel? (Task 3.2)

3. How does the answer to the above question depend on the initial conditions (melt ejection rate, melt superheat, melt composition, initial presence or absence of water) and water addition rate? (Task 3.2)

4. Under what conditions would the crust that forms at initial contact between the melt and the shell be stable and for how long? What is the expected rate of heat transfer between the core melt materials and the shell for various melt conditions? (Task 3.3)

In addition to addressing the questions related to attack of the Mark I containment shell, the results of the following tasks will provide initial conditions for a variety of ex-vessel phenomena that depend on the quantities and dynamics of spreading of the released corium.

Task 3.1 Investigate debris relocation phenomena into the lower plenum of a BWR, including the failure mode of the core plate.

Research Approach

Review and assess existing BWR core relocation experiments and associated analyses. Analyze core plate and RPV failure mode and timing, assuming a range of quantities, composition, and timing of melt. Based on these analyses, identify the key unknowns that can only be determined by further experiments, and assess the feasibility of those experiments. If such experiments are determined to be both necessary and feasible, design and conduct properly scaled out-of-pile tests in which simulated molten core debris is poured onto a scaled core plate or bottom head to verify the failure mode as a function of the nature of the debris (quantity, composition, and timing). The results of this task, including those for any additional experiments carried out under this task, will be used to develop a model to predict lower head failure. All the analyses and experiments should assume expected degraded BWR core conditions (e.g., water in the lower plenum at the time of melt arrival on the core plate and ADS operation.)

Usables and Use

This work will address the expected mode, timing, and temperature of core plate and bottom head failure and passage of molten core debris through core plate and bottom head for use in severe accident analyses. The results of this task will provide crucial initial conditions for a variety of ex-vessel phenomena that depend on the quantities and spreading characteristics of released corium.

Task 3.2 Determine the effect of water on melt spreading, melt cooling, and melt-concrete interactions.

Research Approach

The results of melt spreading, cooling, and core-concrete interaction experiments will be reviewed to determine the extent to which a consistent picture of critical variables and conditions can be drawn. Among the questions to be considered are the role of superheat, the effect of unoxidized Zr in the melt, crust formation and stability, the generation of aerosols and noncondensible gases from core-concrete interactions, and the effect on the behavior of the melt of the presence of water—both water beneath the RPV upon bottom head failure and water added atop the melt following the arrival onto a previously dry floor. Consistent with the results from the scaling program (Task 1), calculations will be conducted either to confirm the conclusions of the review (or clarify issues that were concluded to be ambiguous) and define those near-term experiments that can be conducted to resolve ambiguities of data.

Usables and Use

This work will produce refined estimates of melt spreading, hydrogen and other noncondensible gas generation from melt-concrete interactions, and the effect of water beneath the RPV. Of equal importance, however, this task will bring to focus those questions that may only be addressable by experiment over the long term since this task goes to some of the basic chemical and process effects of high-temperature interactions of the core-concrete materials.

Task 3.3 Determine the conditions under which the interactions between the spreading melt and the Mark I drywell shell will lead to containment shell failure.

Research Approach

Experimental results of heat transfer from steel melts to steel structures will be examined to determine their applicability to reactor accident conditions. In particular, questions such as the effect of the inclination of the shell to the drywell floor and the effect of concrete decomposition gas agitation of the melt will be addressed. Experiments will be devised and conducted to determine in the near term whether a stable crust can form and persist at the drywell shell. Unless otherwise indicated from the results of Task 3.2, this task will assume the presence of water in the drywell. The staff recognizes that it may well be that this task might not be completed in the near term; however, available insights and data will be used in the closure process.

The presence of Zr metal in the melt arriving at the shell depends on the extent of its oxidation both in-vessel and while the melt is spreading over the drywell floor. If it is determined that significant amounts of Zr remain unoxidized before contact with the shell, then the impact of the presence of Zr will be determined.

Usables and Use

The results from this task will enable the staff to determine the conditions under which the shell of a Mark I containment may fail, and, in particular, determine the proximate melt amount, composition, and superheat necessary for failure.

Issue 4—Adding Water to a Degraded Core

There is little doubt that in a severe accident situation the primary efforts of the operators will be directed toward making water available to the reactor vessel. An important question that needs to be considered in view of such likely efforts is what are the likely consequences of those efforts. Given the uncertainties in core melt phenomena, the uncertainty associated with the operator's knowledge of the condition and location of the core during an actual severe accident, and the intuitive drive to put water onto the core in the event of an accident, it is not likely that an operator ever would be told not to put water into the reactor vessel should water become available during the course of an accident. However, along with the potential benefit of achieving a stable, coolable configuration, restoring water to a core that has been severely damaged can have effects of which the operator should be aware. The operator also should maintain cognizance of possible symptoms and response of the plant to adding water in such circumstances (e.g., molten core-coolant interaction, increased hydrogen generation, increased containment pressure).

A related question that also needs exploration is what are the circumstances in which a grossly degraded core can be prevented from melting its way through the vessel lower head. Although important parameters can be identified as water availability, system pressure, and debris configuration, it is not likely that complete resolution to this issue will be achieved in the near term. Nonetheless, there are questions related to the addition of water to a degraded core that can be addressed in the near term that will provide some insights into the longer-term issues of accident management as well as improve understanding of low RCS pressure core melt accident phenomena. The questions to be addressed are the following:

1. What are the amounts and rates of hydrogen and steam generation during the reflooding of a degraded core and during relocation of the melt onto the bottom head? (Task 4.1)
2. What is the potential that reflooding a severely damaged BWR core will result in a recriticality? What will be the effects as opposed to not adding water? (Task 4.3)

Task 4.1 Determine the effect of water injection on the generation of steam and hydrogen during reflooding of a degraded core.

Research Approach

Identify basic variables governing heat transfer and hydrodynamics of melt-water interaction, including the effect of water injection on debris reconfiguration. The work would be oriented toward scoping the range of hydrogen generation that can be produced during quenching and would assess the impact on containment performance. In consideration of the results of the scaling analysis (Task 1), establish whatever additional experiments may be needed or conduct analysis needed to determine the effect of adding water to a degraded core. We have already performed a lot of TMI–2 analyses and will assess these results.

Additionally, a benchmarking exercise, using available models (to be selected), against the TMI–2 accident beginning at 174 minutes (when core was reflooded by the start of a reactor coolant pump and high-pressure injection) will be undertaken as part of this task.

Usables and Use

This task will produce estimates on the amounts and rates of steam and hydrogen generation as a function of water addition to degraded core geometries. The effect of adding water to degraded cores has an important application to accident management as well as to improving the quantification of the resultant containment loads.

Task 4.2 Stability of melt-supporting crusts.

Research Approach

The formation of a crust (crucible) supporting a melt, the predicted failure of the crust, and the eventual relocation of the melt into the bottom of the reactor vessel depend strongly on the heat transfer coefficients at boundaries of the core melt crucible. The anticipated approach is to review and assess existing analyses for both the growth and demise of a melt-supporting crust and subsequently perform a set of calculations of heat transfer along boundaries of the crust crucible, analyzing thermal and mechanical crust stability as a function of melt accretion.

Usables and Use

The work will affirm whether the heat transfer coefficient and the code treatments of core behavior, including crucible formation and collapse and debris relocation into the lower plenum, are suitable.

Task 4.3 Investigate the possibility and consequences of recriticality in degraded BWR cores.

Research Approach

There are two configurations to be considered with regard to the recriticality problem. One is the possibility that standing fuel pellet stacks will become critical upon reflooding (with control rod materials previously melted out). The other is the possibility that core material will become critical upon relocation to the bottom head.

The approach is to first review and assess existing analyses of both the likelihood and consequences of recriticality of a damaged BWR core. Where deficient, additional calculations will be performed assuming the existence of critical masses for expected degraded BWR core geometries reflooded with unborated water.

Usables and Use

The products of this task would be an estimate of the likelihood and effects of recriticality on accident progression and the alternative design mitigative measures (e.g., minimum boron concentrations in the reflood water) that might eliminate the problem if one is found to exist.

Issue 5—Use and Status of Severe Accident Models (Codes)

As noted earlier, the confidence in the analyses and judgments that will be made in implementing the staff's Integration Plan will depend on how well the tools used in performing those analyses and making those judgments conform to the realities of a severe accident. The tools that have been developed under the SARP, the physical and phenomenological models embodied in computer programs (codes), have had and continue to have a key role in furthering the objectives of severe accident research, viz., support of the U.S. Nuclear Regulatory Commission's policy toward severe accidents. First, and perhaps most critical, is the use of the codes in identifying the important sources of risk. In this capacity the impact of the codes is important since the results of code calculations support the evaluation of risk. The evaluation of risk, in turn, significantly determines both what is researched and the relative priority of what is researched. Second, through an iterative process, exercising the codes helps define the experiments needed to improve the representation of physical phenomena by the codes. Third, the complex mechanistic codes provide a benchmark for less detailed but faster-running codes that are used in applications that require extensive calculations, such as parametric studies for evaluating accident management strategies. This "two-tier" code strategy has been pursued because the use of the detailed mechanistic codes to evaluate entire accident sequences is costly, impractical, and unnecessary.

Any large code development program is always faced with the difficulty of determining when the codes are "good enough." That is, when have they achieved an accuracy sufficient for the application for which they were intended? At this time, an acceptable level of accuracy has not been defined. As a result, the current code development program is continually iterative. Codes are developed and assessed against selected experiments and needed model improvements identified. Overlaid on this process is the identification of new phenomena considered important to typical plants that "must be" incorporated into the code.

In addition to the above, a problem that is not unique to the severe accident program relates to the number of codes that must be developed to have a complete analysis capability. If one phase of the accident is difficult to model, or has inherently large uncertainties or variabilities, or perhaps is best understood as stochastic processes with a distribution of potential outcomes, it is not appropriate to model other phases of the sequence in greater detail, since both the overall variability and uncertainty will be driven by the least precise and least certain processes, respectively. Currently, this appears to be the situation with regard to late-phase core melt progression. The degree to which the early phase is modeled, and those models tested and validated by experiment, must be tempered by the degree to which the late phase through melt relocation and vessel failure can be understood and modeled and that understanding tested by experiment.

Therefore, as part of the code development program, a method must be developed that constantly assesses the state of code development from a number of standpoints in addition to how well a particular sequence or facet of an accident can be modeled. The following questions will be considered:

1. How well do the mechanistic models reflect the phenomena believed to be important to severe accidents? (Are the correct phenomena being modeled?)

2. How well does the interactive program of code advancement and experimentation achieve the objective implied in 1 above?

3. Is the level of detail in the codes appropriate to their use? (Are some codes more detailed than needed, others not detailed enough?)

4. Which stages of an accident need to be modeled by detailed mechanistic codes, coupled to adjacent stages? (While clearly there is an interdependency among stages of a severe core melt accident, a

detailed mechanistic code coupling all stages of accident phenomena from the onset of core uncovery through containment failure is not needed to either understand or make regulatory judgments about severe accidents. Further, a detailed mechanistic model of all aspects of a severe accident may not be possible, let alone necessary.)

5. Is the level of precision needed for regulatory use being considered in the code development program?

6. Is the level of precision from a given code more than needed, and is it suitable to the expected overall levels of uncertainty from an integrated analysis package for applications as discussed above (e.g., accident management)?

Answering the above questions is a major undertaking and is not likely to be achieved in the near term. Nonetheless, some measure of confidence in the tools that will be used in achieving the near-term goals must be made. Therefore, the following task will be done. Although not a complete code assessment, this task is believed to be a workable compromise in the near term and should provide sufficient information to enable the staff to gauge the appropriate level of confidence to be placed in calculations using the codes.

Task 5 Code documentation and review.

Research Approach

For each code developed under the SARP, the developer will be asked to supply the following information:

1. Stages in a severe accident to which the code is intended to apply and the accident sequences in which those stages are found.

2. Current capability of code. (Of the intended accident stages and phenomena, which does the code now model, which does it not.)

3. Limitations of the code. (Portions of accident stages and/or special assumptions under which the code operates.)

4. Statement of why the code needs to be or does not need to be coupled sequentially to adjacent codes to yield sufficiently precise calculations.

5. The degree to which the relevance of the physics, as embodied in the code, to severe accidents has been tested by experiment. (This point questions not simply the conformance of code to experiment, but experiment to severe accident.)

6. What features (physics and chemistry) not presently in the code/model need to be incorporated to enable the code to be a reasonable representation of accident phenomena.

7. To what quality assurance program was the code subjected.

Usables and Use

The above information will be reviewed and scrutinized by the staff, and an assessment will be made of each code as to its present scaleup capability and usefulness in supporting achievement of the near-term goals of the revised SARP. Based on that assessment, recommendations as to use of the present form of the code, further development, or abandonment will be made.

Criteria for judging the suitability of each code are being developed. These criteria will take into consideration the intended end use of the code and the expected user of the code. Deciding when code development is completed would require a subjective judgment and a sense of proportion between the level of precision needed for application and the levels of uncertainty and variability in models of severe accident phenomena.

This latter task is applicable to both the near- and long-term research as is Task 1 on similitude. The conduct of Tasks 1 and 5 is believed logical and necessary to the critical reexaminations of SARP needs. However, neither is intended to constrain closure of the near-term issues described above.

4. Meeting Long-Term Goals

The research described in this section of the SARP plan consists of work primarily directed at reducing uncertainties in the estimation of the risk presented by severe accidents. This orientation is consistent with the direction provided in "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants," NUREG/CR-4883, known as the "Kouts Report." That direction suggested a broad base approach, investigating manifold phenomena with the common goal of reducing uncertainties. Both the approach and the goal remain valid, as does the more specific direction in NUREG/CR-4883 that identifies individual technical questions as contributors to uncertainty in risk estimates. However, there are sources of new information that bear on how to order research and distribute resources among the topics of the SARP. These sources are the revised NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," and the guidance provided in SECY-88-147 that identifies accident management as a strategy that the NRC will pursue in dealing with severe accidents. In addition, the successes of the SARP itself over the years are a source of information providing a clearer picture of which phenomena are critical to understanding severe accidents, insight into what avenues of investigation are likely to be most fruitful, and which topics are likely never to be fully resolved in a reasonable time frame within realistic resources. Taking the above together with the needed balance discussed in Section 2, specific topics have been identified that characterize the SARP efforts directed toward achieving the long-term goals cited in Section 2—providing an understanding of the range of phenomena exhibited by severe accidents, including impacts of generic accident management schemes, and developing improved methods for assessing the "severe accident source term." These topics are the following:

1. Review of the SARP approach to modeling severe accident phenomena
2. In-vessel core melt progression and hydrogen generation
3. Hydrogen transport and combustion
4. Fuel-coolant interaction
5. Molten core-concrete interactions
6. Fission product behavior and transport
7. Fundamental data needs

It is not at all clear that detailed mechanistic models can be, or need be, developed for items 2 through 6. Some phenomena may only be tractable when treated statistically as a population of related events with differing outcomes. In recognition of this possibility, the first item listed is a systematic review of the approach taken in the SARP to modeling severe accident phenomena.

Issue L1—Modeling Severe Accidents.

Because of the difficulty in performing prototypic experiments and the variety of scenarios possible, substantial reliance must be placed on the development and validation of complex computer codes for analyzing severe accidents. A number of "mechanistic" codes (e.g., SCDAP/RELAP, MELPROG/TRAC) have been developed for various stages in severe accidents, both in-vessel and ex-vessel, for both PWRs and BWRs. However, it appears that there are practical limits to the feasibility of deterministically modeling all aspects of severe accident behavior. Moreover, it is not clear that all aspects of severe accidents need to be mechanistically reflected in deterministic codes. The resolution of some severe accident regulatory issues may be achieved with bounding analyses alone. Further, some phenomena may be better addressed employing stochastic rather than deterministic techniques. Priorities for additional development can be determined through use of probabilistic risk assessment techniques. When risk is insensitive to the uncertainty in the phenomena or variabilities in parameter values, there should be no need to attempt to further reduce the uncertainty or variability.

Using the "documentation" provided by Task 5 of Section 3 of this plan as a point of departure, the approach of developing mechanistic, deterministic codes for each stage of severe accidents will be reviewed. For each code development program/phenomenon being addressed, a review of progress to date, magnitude of uncertainties remaining, and degree of validation will be made. Based upon this information, a determination will be made as to whether an alternative approach might be appropriate for dealing with some phenomena. The efficacy of the SARP's "two-tier" code strategy (fast-running lumped parameter system codes benchmarked against a combination of individual standalone and coupled mechanistic, deterministic codes) also will be examined. This review will start at the beginning of FY 1990, and recommendations will be made at the end of FY 1990.

The NRC staff assessment of the modeling program will be assisted by various individuals outside the NRC and will be drawn from universities, national laboratories, and industry personnel who are expert in accident phenomenology (have appreciation for the purpose of the analysis and the uncertainties in all the parts of the analysis), regulatory concerns, and programming and numerical techniques. This effort would identify the strength or weakness of different modeling approaches, quantify the uncertainties in code models, and identify additional research, if any, where uncertainties and risk importance are significant.

Issue L2—In-Vessel Core Melt Progression and Hydrogen Generation.

In-vessel core melt progression concerns the state of the reactor core from the start of core uncovery to reactor vessel failure. Included phenomena are thermal attack by the core debris upon the reactor structure and the reactor vessel, in-vessel hydrogen generation, in-vessel natural convection and heat transfer, and in-vessel steam generation. The details of core melt also determine rates and amounts of in-vessel fission product release and aerosol generation and much of the fission product and aerosol transport (and retention) in the reactor coolant system. (Some aspects of this issue relating to fuel-coolant interaction were discussed in Section 3.)

In considering melt progression, BWRs and PWRs need to be treated separately, mainly because of their different fuel assembly, control element, and lower plenum structures. For example, the effect of water injection on (1) the integrity of upper in-vessel structures, (2) the mode of core relocation, (3) hydrogen generation upon core relocation, and (4) the mode of bottom head failure and the details of the release of core and structural debris into the drywell following failure are questions common to both PWRs and BWRs. However, the answers are quite distinct arising from the physical and geometric differences between BWRs and PWRs. Bounding calculations and a few small-scale experiments may be continued to identify which phenomena are critical for each reactor type and whether further reduction in uncertainties and calculational variabilities related to melt progression and hydrogen generation are needed and achievable.

Areas in which work will continue are the examination of TMI-2 data for whatever insights might be provided, the NRU fuel melt experiments, and participation in the existing cooperative program with the Federal Republic of Germany at the CORA facility. New experiments that evolve in the near-term research may continue as well.

Issue L3—Hydrogen Transport and Combustion.

The major concerns regarding hydrogen in LWRs are that the static or dynamic pressure loads from hydrogen combustion and detonation may breach containment integrity or that safety-related equipment may be damaged as a result of either pressure loads or high temperatures. To assess the possible threat to containment and safety-related equipment, it is necessary to understand how hydrogen is transported and mixed within containment and to determine the likelihood of various modes of combustion. The hydrogen behavior issues have been extensively investigated since 1979, but several important areas of uncertainty remain:

1. High-temperature/high steam concentration combustion, and

2. Deflagration-to-detonation transition.

Research programs on these issues of combustion have been sponsored in the U.S. by the NRC and the nuclear industry and in the international community. However, the combustion processes are sufficiently complex that many aspects are still not well understood. The resulting uncertainties in the threat to containment integrity are unlikely to be reduced significantly by the existing research programs. The intent of this research is to reduce these uncertainties; however, in some cases reduced uncertainties are not required to make a near-term regulatory decision. Each category of uncertainty is discussed below.

1. High-Temperature Combustion

The Zeldorich-von Neumann-Doering (ZND) chemical kinetics theoretical model developed under NRC sponsorship predicts that increasing temperature has a strong effect on the combustion and the detonation of off-stoichiometric hydrogen-air mixtures and on all hydrogen-air-steam mixtures. It is believed that the steam inerting effect is reduced greatly at elevated temperatures. There is a limited supporting data base for this phenomenon. Experiments are necessary to resolve the uncertainties associated with high temperatures and steam concentrations typical of those likely to be encountered in severe accident scenarios. It is also necessary to extend or develop more mechanistic models to predict, by either extrapolation or interpolation, the temperature sensitivity of hydrogen-air-steam mixtures that have not been tested.

The small-break LOCA and station blackout scenarios are two examples of high-temperature hydrogen-air-steam mixtures that may exist below the autoignition temperature (550°C minimum value for stoichiometric, hydrogen-air mixtures). There are two aspects of the high-temperature/high steam concentration combustion problem. The first is the injection of high-temperature hydrogen and steam mixtures that autoignite upon contact with preexisting and premixed hydrogen-air-steam mixtures. The second aspect is the injection of hydrogen and steam mixtures at elevated temperatures that do not autoignite upon contact with preexisting hydrogen-air-steam mixtures. This allows the possibility of a premixed condition to form and a subsequent deflagration or detonation. In both cases, the competition between chemical reaction rates and physical mixing rates will ultimately determine the ensuing combustion mode. Data are needed to determine the hydrogen-air-steam flammability limits and the steam inerting criterion at elevated temperatures if reliable predictions are to be made as to the likelihood and the potential threat resulting from various combustion mode(s) for a wide range of accident conditions.

Therefore, the research approach is to determine hydrogen-air-steam flammability limits, volumetric oxidation (chemical reaction) rates, physical mixing rates, and the competition of these rates during the injection of high-temperature hydrogen and steam mixtures into cooler preexisting and premixed hydrogen-air-steam mixtures. A draft proposal is currently under consideration by the NRC that addresses feasibility; construction design, cost, and schedule; and an experimental test plan. It is our current estimate that if a facility is needed, it can be constructed by early FY 1990 and that experimental results can be generated by late FY 1991 or early FY 1992. Final data reduction and formal documentation should be available in late FY 1992.

2. Deflagration-to-Detonation Transition (DDT)

Direct initiation of a detonation would require a concentrated high-energy source for insensitive, steam-diluted mixtures. This is not considered a credible mechanism of initiation by almost all researchers. However, it is possible to initiate a flame with a low-energy source such as a spark or a glowplug, and the subsequent propagation through orifices and around obstacles such as pipes can result in flame acceleration that culminates in a transition to detonation.

The possibility of DDT in realistic and prototypic containment geometries and conditions needs to be resolved. The uncertainty in this area has increased because of recent experimental and theoretical results that indicate an increased likelihood of detonations at high temperatures and large steam fractions as discussed earlier. The current data base on flame acceleration and DDT suggests that the mixture composition, obstacles, and venting are all important factors. These uncertainties result from a lack of experimental data on flame acceleration and DDT for conditions that include the effects of steam dilution, elevated temperature, large-scale and prototypical obstacle types, and spacing.

At present, no reliable model exists to predict or extrapolate DDT results from small-scale experiments to containment scale. Reactor safety studies and fundamental combustion research are being carried out in the Federal Republic of Germany and Canada. These data along with data generated for space shuttle application will be applied to reducing the uncertainty associated with DDT as well as to assessing the potential threat of DDT to containment integrity. Specifically, this data base will then serve as the basis to improve or develop correlations and models to allow extrapolation of experimental results to reactor scale and accident conditions.

Issue L4—Fuel-Coolant Interactions.

Molten fuel contacting water can give rise to a range of phenomena. Very energetic steam explosions could result in early containment failure (alpha-mode failure). Less energetic interactions do not threaten containment directly but could change the course of the accident and the magnitude of the source term. Among these are sudden coherent failure of the vessel lower head, ex-vessel debris dispersal, and steam generation pressure pulses. (Core debris dispersal and steam and hydrogen generation in some BWR Mark II and Mark III containments also warrant further evaluation as to whether fuel-coolant mechanisms can pose a threat to containment integrity.)

The research to be conducted will be selective, confirmatory in nature, and focused to address some well-defined questions dealing with key parts of steam explosion phenomenology, such as degree of premixing, triggering, and fragmentation within the detonation wave of a steam explosion for various premixture conditions. It is clear that the program will necessitate limited scale experiments to test the predictive capability of calculations of premixing and the fragmentation rate of melt drops in the explosion zone of a propagating explosion. Further, the utility of the calculational tools to assess the effect of (1) reflood of a damaged core or debris bed in-vessel, and (2) fuel-coolant interaction in a suppression pool will be examined.

In addition to providing final confirmation of the staff's position on the issue of the alpha mode of failure, this work also will provide analytical tools, additional data, and insights for use in evaluating the dynamics of molten fuel-coolant interactions in-vessel and in various containment configurations.

Issue L5—Molten Core-Concrete Interaction (MCCI).

This is a subject fundamental to severe accidents. It has received considerable research attention, both experimentally and analytically, mainly addressed at understanding the quasi-steady-state interaction that would develop when the core melt materials form a pool above the concrete. However, some significant uncertainties remain. An important question is the long-term coolability of initially molten corium pools interacting with concrete and flooded with water. Other uncertainties involve the transient (early) stage of core-concrete-water interactions, which include the spreading and relocation of the melt over concrete and the associated thermal-hydraulic characteristics of the attack. Also uncertain is the effect of pour rate on melt spreading.

Specific topics that are addressed by the research include the key characteristics of molten-corium pools interacting with the concrete basemat, the rate of melt cooling in the early stages of MCCI, the energy balance of a corium-concrete interaction in the high-temperature regime, the rate of fuel cooldown as it spreads over the concrete, the effect of water on the spreading behavior and associated heat losses of the corium melts, the erosion and ablation of concrete structures (in particular, the reactor vessel pedestal), and the volatilization of fission products and production of aerosols during MCCI and estimation of the effects of these phenomena on the source term. Several of these phenomena investigations are being carried out under the existing cooperative agreement with the EPRI ACE program and FRG BETA program. The design of any additional experimental and analytical programs will follow from the work discussed in Section 3, Tasks 3.2 and 3.3.

Issue L6—Fission Product Behavior and Transport.

The NRC has had a substantial program to investigate fission product release from fuel in-vessel. Today PRA studies no longer identify this as an area of major uncertainty in risk assessment. The recent NUREG-1150 elicitations on a number of source term issues do indicate, however, that late iodine release, revolatization,and fission product release from core-concrete interactions can have an effect on the overall risk uncertainty. Hence, research in these areas is being continued. Nonetheless, because severe accident issues associated with phenomena that have the potential to result in containment failure have higher priority than modeling of fission product release and transport, source term issues will be addressed in the long-term program with lower priority.

Issue L7—Fundamental Data Needs.

For some ranges of phenomena, fundamental data (physical and chemical properties and constants) do not exist or are so poorly known that their use provides no confidence in the fidelity of experimental or analytic results. Data needs have been briefly touched upon in Issue L2 in connection with core melt progression, but similar needs exist with respect to MCCI. Among these needs are thermal properties (including melting points, latent heats, and thermal conductivities) and phase relationships among the various constituents of the debris. A continuing program to identify and measure (or calculate) the required basic data and to incorporate the results in the codes will be instituted. However, in supporting this basic data effort, we will attempt to strike a balance between the accuracy of the data and its significance relative to regulatory applications.

APPENDIX A

Background on Severe Accident Research Program and Relationship to Other Elements of SECY-88-147

For the past 10 years or so, since the Three-Mile Island accident, NRC has sponsored a research program on severe nuclear power plant accidents as part of a multifaceted approach to safety. Other elements of this approach included improved plant operations, human factor considerations, and probabilistic risk assessments. In August 1985, the Commission issued a Severe Accident Policy Statement (50 FR 32138), which concluded that existing plants posed no undue risk to public health and safety. However, the Commission recognized that systematic examinations of existing plants could identify plant-specific vulnerabilities to severe accidents for which further safety improvements could be justified.

In May 1988, the staff presented to the Commission an Integration Plan for Closure of Severe Accident Issues (SECY-88-147). The Integration Plan consists of six major elements:

1. Examination of existing plants for severe accident vulnerabilities (individual plant examinations).

2. Development of generic containment performance improvements with respect to severe accidents to be implemented if necessary for each of the six containment types.

3. Upgrading of staff and industry programs to improve plant operations.

4. A severe accident research program.

5. A program to define how and to what extent vulnerabilities to severe accidents from external events need to be included in the severe accident policy implementation.

6. A program to ensure that licensees develop and implement severe accident management programs at their plants.

During the past few years, the SARP has generated a large amount of insight into the progression of severe accidents. An extensive experimental program has led to a vastly improved understanding of in-vessel core melt progression and associated phenomena of hydrogen generation and fission product release. Models of core melt progression are being coupled to thermal-hydraulic in-vessel and primary cooling system models to allow more rigorous treatment of the in-vessel stages of a severe accident. The ex-vessel phenomena of melt ejection and direct containment heating in high-pressure accident scenarios and molten core-concrete interactions in lower-pressure scenarios, together with their associated threats to containment integrity, also are becoming better understood as a result of the ongoing experimental and analytical efforts under the SARP. Understanding containment challenges due to steam and noncondensible gases is much improved as well. The "two-tier" code strategy of developing detailed "mechanistic" codes for understanding and faster-running "integral" codes for application has furthered the NRC's ability to calculate the impact of accident sequences and their associated risks to public health and safety. Both levels of codes have been used in the risk assessment document, NUREG-1150.

As can be inferred from the above, the thrust of the SARP up to now has been to establish and refine the technical and scientific base of knowledge in the area of severe accident phenomenology, to apply it at the scale of reactor accidents, and to reduce the uncertainties in this knowledge base and the risk assessments that depend on it. The program to date generally has been appropriate to the above motivations. However, because the needs for the SARP are changing as a result of actions being taken by the NRC to bring severe accident regulatory issues to closure, it is natural to question whether the directions of the program should change accordingly. Obviously there will remain practical and resource-related questions as to how many fronts can be pursued in understanding and characterizing accident phenomenology and which of these fronts will prove the most fruitful ones to follow to achieve closure of regulatory issues.

A.1 Development of Revised SARP

To assist the NRC RES staff in this reappraisal and revision of the SARP, a set of four expert groups was established to help the RES staff identify and define the status of present SARP activities that relate to the Integration Plan, identify and focus research necessary for sound regulatory decisions to be made within the framework of the Integration Plan, and identify and rank the base of confirmatory research activities directed at achieving the long-term goals. Each group was composed of contractor personnel from the Department of Energy (DOE) laboratories, consultants from universities and industry, and NRC staff. Each member was active in severe accident research. The groups consisted of three "working groups" that addressed the technical points of the research and an "integration group" that dealt with programmatic considerations.

The working groups considered detailed technical issues, defining their status and needs for further research within the framework of the Integration Plan and confirmatory research. The integration group took the reports of the working groups and evaluated and synthesized them, along with other considerations, into an earlier draft of a recommended Severe Accident Research Program plan. The NRC staff then used this as a basis for preparing the revised SARP plan discussed in this report.

A.2 Relationship of Revised SARP To Other Elements of Integration Plan

In order to place the revised SARP plan in perspective, it is useful to first discuss in general terms the relationship of SARP to the other elements of the Integration Plan. From the discussion it will be apparent that SARP provides, or will provide, important and often essential data to most of the other elements of the Integration Plan.

1. Relationship of SARP to individual plant examinations (IPEs)

The Integration Plan provides specific objectives for the IPEs that each utility is expected to meet:

a. Reduction of the overall probability of core damage and fission product releases by appropriate hardware and procedure modifications;

b. Development of an overall appreciation of severe accident behavior; and

c. Development of an understanding of the most likely severe accident sequences that could occur at its plant.

It is envisioned that a principal tool for an NRC audit review of IPE submittals will be a relatively fast-running, integrated severe accident analysis code such as MELCOR. (This code in turn is benchmarked and validated against detailed and more mechanistic codes.) All these codes draw or have drawn heavily for their phenomenological modeling on the SARP experimental and analytical efforts (including the foreign and industry contributions under cooperative arrangements). Up to now, these codes have been under continual development and validation. However, it is presumed that, when the staff begins its review of the IPEs, some versions of these codes will be "frozen" for use by the NRC staff.

It should be noted that the new SARP addresses all the phenomenological uncertainties discussed in Appendix 1 to the IPE Generic Letter No. 88-20.

2. Relationship of SARP to containment performance improvement program

The Integration Plan states that the containment performance improvement (CPI) program complements and is closely integrated with the IPE program and is intended to focus on resolving hardware and procedural issues related to generic containment challenges.

A detailed summary of the relationship of SARP and CPI is given in Section 5.2 of the enclosure to the Integration Plan (SECY-88-147).

3. Relationship of SARP to improved plant operations

The improved plant operations (IPO) program includes elements relating to continued improvements of the technical specifications, emergency operating procedures (EOPs), expanding EOPs to include guidance for severe accident management strategies, and industry programs to reduce transients and other challenges to the engineered safety features (ESFs). These elements are obviously closely related to accident management strategies, discussed below.

The relationship of the SARP to these elements of the IPO program involves providing analytical descriptions of severe accident sequences, including the effects of EOPs and other mitigation strategies. These analyses would predict the effects of accident mitigation schemes, including both a determination as to whether they will be effective and identification of possible undesirable consequences. This is particularly true with respect to the later phases of in-vessel (core degradation) progression of a severe accident and the ex-vessel and containment phenomena. Accident sequence analyses, based on SARP results, could be used to judge the appropriateness and effectiveness of EOPs under severe accident conditions.

4. Relationship of SARP to external events

According to the Integration Plan, the evaluation of external events will proceed separately with a different schedule from that of internal events. At the moment, there is no part of the SARP that explicitly addresses external events. However, to the extent that external events such as earthquakes or fires act as accident initiators or constrain the availability of ESFs, safety-related equipment, and other mitigation strategies, the accident progression can be analyzed with the severe accident codes developed under the SARP plan.

5. Relationship of SARP to accident management program

The Integration Plan defined accident management to include measures taken to prevent core damage; to terminate the progress of core damage if it begins; and, failing that, to maintain containment integrity as long as possible to minimize offsite releases. The NRC accident management program concentrates on near-term improvements based on well-understood accident management procedures and strategies. The SARP will provide the data base to allow the staff to examine procedures and strategies whose benefits and adverse effects are not well understood.

As in other areas, the role of SARP in accident management is to provide the tools (i.e., codes) to analyze the progression of severe accidents and to evaluate the effects of given mitigation strategies (both beneficial and detrimental). The staff is currently in the process of defining the analytical tools that are available or should be developed and required for accident management purposes. We expect to complete this effort by the end of FY 1990. It is important that SARP should continue to address certain important issues—in particular, the consequences of adding water, both in-vessel (reflood) and ex-vessel, in attempting to cool a severely damaged core or quench core debris—and that further attention should be given to evaluation of the uncertainties associated with the use of the codes.

APPENDIX B

A Severe Accident Scaling Methodology (SASM)

B.1 Objective and Outline

The objective of this appendix is to outline a Severe Accident Scaling Methodology (SASM) that could be used to address in a systematic and practical manner questions concerning such scaling topics of interest to severe accidents as:

1. The adequacy of the design and operation of a reduced-scale test facility to provide experimental data that can be used in safety analyses of full-scale nuclear power plants.

2. The appropriate initial and boundary conditions for experiments of interest.

3. The use of a set of experimental data or of a correlation based on the data in nuclear power plant safety analyses.

4. The effects of test facility scale distortions (if present) on physical and chemical processes of interest to nuclear power plant safety analyses.

5. The capability of a computer code to scale up physical and chemical processes observed in reduced-scale test facilities to full-scale nuclear power plant conditions, etc.

The role that scaling plays in nuclear power plant safety analyses is discussed in Section B.2 of this appendix. The need for establishing a severe accident scaling methodology (SASM) is discussed in Section B.3, together with the requirements that it would have to meet. The elements of such a methodology are outlined in Section B.4, together with their rationale. The last section describes a program directed at demonstrating the methodology by applying it to the DCH problem.

B.2 Elements of Safety Analyses and Premises

As was the case in studies of LOCAs, analyses concerned with severe accidents will have to be supported by a suitable experimental data base on appropriate models and computer code calculations, and, when appropriate, on bounding calculations. Implicit to this approach are two premises: one pertaining to experiments and the other to codes; both are concerned with scaling.

For reasons discussed below, both premises must be addressed and evaluated before a high degree of confidence in the analysis will exist.

B.2.1 Premise Related to Experiments

In nuclear reactor safety research, experimental data are used (1) to develop correlations and/or models for a particular process or (2) to assess code capability to calculate such a process.

Full-scale test facilities capable of generating the experimental data of interest are prohibitively expensive to construct and to operate. In severe accident research, difficulties and costs are compounded because some phases of an accident scenario involve failures of the reactor vessel and of containment structures. Thus, large (or even smaller) scale integral facilities needed to provide experimental data on processes leading to or attending such failure events become prohibitively expensive.

Consequently, in severe accident research, the majority of experiments will have to be performed in reduced or small-scale separate effects test facilities. The premise made in following this approach is that experimental data from such facilities are applicable and relevant to nuclear power plant conditions. This implies that test facilities, as well as the initial and boundary conditions of experiments, are properly scaled so that distortions (if and when present) will not affect the evolution of physical and chemical processes of interest.

Whether a facility and the initial and boundary conditions of an experiment are well scaled will have to be evaluated for each facility and set of experiments because such an evaluation will determine whether the data can be used in nuclear power plant safety analyses of a postulated severe accident.

B.2.2 Premise Related to Codes

The reliance on computer codes to simulate the behavior of a nuclear power plant during a postulated accident scenario is predicated on three factors. First, it is too costly, and for severe accidents not even feasible, to subject a nuclear power plant to such an event. Second, very often one cannot directly apply results from test facilities to nuclear power plants; this is particularly true for separate effects test data. Third, the study of various plant recovery techniques can only be performed by computer codes.

Implicit to applications of computer codes to analyses of postulated accident events in nuclear plants is the premise that these codes have the capability to scale up phenomena and processes from test facilities to full-scale plant conditions. For reasons discussed below, this premise must be evaluated on a case-by-case basis, that is, for each postulated accident scenario or for a set of scenarios.

It is often stated that advanced codes are "mechanistic" and are based on "first principles." Therefore, ipso facto, they have the scaleup capability. However, as a matter of fact, for the following three reasons this is not the case.

First, because the conservation equations used in computer codes are space averaged and because of their dependency on numerous empirical correlations, computer codes are not based on "first principles." As the capability of a code to model a particular process and/or phenomenon is provided by particular closure relations, the scaleup capability of a code will depend on whether or not the empirically determined closure relations have this capability. If the test facilities that generated the data are well scaled and the experiments are performed with appropriately scaled initial and boundary conditions, the empirical closure relations, and therefore a code, can be used in analyses of full-scale nuclear power plants. Otherwise limitations due to scale distortions must be assessed before meaningful safety analyses can be performed.

Second, because of discretization schemes used to nodalize a nuclear power plant and perform calculations, the computed values are not local but are averages over very large volumes. Consequently, these averages are functions of node size and may affect the evolution or the timing of a physical or chemical process calculated by the code. Furthermore, as nodalization used to model a nuclear plant and small-scale test facilities differ (the latter have most often a much finer nodalization), the events calculated to occur in a full-scale plant may differ from those observed in test facilities. This problem becomes even more serious if a test facility has some scale distortions that could affect a particular process of interest.

Finally, because of "compensating errors" that may be present in a code, there is no assurance that a code has the capability to scale up processes observed in small-scale test facilities to full-scale nuclear plants. "Compensating errors" are generated most often during the code validation process. Since advanced codes have numerous parameters, coefficients, that is, "dials," improved agreement with experimental data is very often achieved by adjusting some of these "dials." This "tuning" process of a code to a set of experimental data can introduce "compensating errors" in the code. The effect of such errors on the scaleup capability of a code becomes even more difficult to assess if scale distortions are present in the facility or if the initial and boundary conditions of an experiment are not properly scaled.

It can be concluded from this brief discussion that if meaningful safety analyses concerned with postulated severe accidents in nuclear power plants are to be performed, then questions related to scaleup capabilities of experimental data and of computer codes will have to be addressed.

B.3 Needs and Requirements

The important role that scaling has in experimental and analytical investigations related to severe accidents was noted in the preceding section. This role establishes a need for a scaling methodology that would be applicable to both experiments and computer codes.

In order to meet the needs of research and development activities conducted for a regulatory agency, the methodology should:

1. Be systematic and practical, auditable and traceable.

When applied to a specific severe accident scenario, the methodology should:

2. Provide the scaling rationale and similarity criteria.
3. Provide a procedure for conducting comprehensive reviews of facility designs, test specifications, and results.
4. Provide a measure or index to indicate the applicability of correlations or models based on test data from sub-scale facilities to full-scale nuclear plant conditions.
5. Quantify the effects of scale distortion.
6. Quantify the effects associated with extrapolating correlation and/or models beyond their data base.
7. Relate in a *systematic* and *unifying* manner
    - test facility design and operation,
    - test data accuracy and applicability, and
    - correlations (or model) accuracy and applicability to (1) computer code scaleup capability and (2) applicability to calculate the postulated scenario for a full-scale nuclear power plant.
8. Provide a quantifiable and traceable procedure for specifying and prioritizing future experiments if needed.

It should be noted that requirements 4 through 7 provide the information needed to quantify the code uncertainty to calculate the postulated severe accident scenarios.

A scaling methodology that meets these requirements is outlined below.

B.4 Elements of SASM and Their Rationale

The proposed Severe Accident Scaling Methodology (SASM) consists of four primary elements as shown in Figure B.1.

The first element: *Identification and Ranking of Phenomena* contains Steps 1-3. In this element, scenario modeling (and therefore scaling) requirements are identified by (1) specifying the scenario (Step 1), (2) establishing the event tree and selecting the accident path (Step 2), and (3) identifying phenomena/processes along this path and ranking their importance (Step 3). Activities carried out in this element are designed to meet the first requirement set forth in the preceding section. These activities are similar to those discussed and developed in References B.1 and B.2, that is, they:

1. Provide a comprehensive, physically based framework for analyzing an accident scenario. This is essential to a scaling methodology that is systematic, auditable, and traceable.

2. Decompose the scenario in elementary components and provide a casual relationship approach. This is essential for identifying modeling (and therefore scaling) requirements and understanding the role of each component (and therefore model) in the selected accident path.

3. Identify and rank processes and phenomena along the accident path. This is important to a scaling methodology that is not only systematic but also practical. The need for this screening process arises from the fact that it is not feasible either to design an experiment or to develop a code that will scale properly all processes occurring during an accident. What is needed, however, is to ensure that physical and chemical processes important to the evolution of an accident are properly scaled.

4. Identify and prioritize research activities directed at resolving potential safety concerns related to an accident scenario. This is essential to an expeditious closure of a safety issue.

The second element: *Identification and Systematization of Scaling Criteria* contains Steps 4 and 5. In this element, scaling criteria are (1) identified by performing a top-down scaling analysis (Step 4) and (2) systematized by applying group theory methods (Step 5) discussed in References B.3, B.4, and B.5. Activities carried in this element are designed to satisfy the second and third requirements listed in the preceding section, that is, they:

1. Structure the scaling process to follow the physically based, hierarchical approach detailed in Element 1. Such a structure is essential as it provides not only the rationale for establishing scaling and similarity criteria but it ensures also that processes important to the scenario are taken into account in the similitude analysis.

2. Provide a structure and procedure that starts from a global, top-down viewpoint and introduces complexity and detail at each lower level. This is important for conducting comprehensive reviews of facility design, test specifications, and results.

3. Provide a systematic and traceable approach to derive and select similarity parameters. This minimizes the arbitrariness, that is, the ad hoc approach used so often in facility design and test specifications.

4. Provide a method that can yield similarity criteria for processes that can limit the operational range of a system.

The third element: *Quantification*, contains Step 6, with activities designed to meet requirements 4 through 6, that is, to quantify the effects associated with scale distortion and/or with extrapolating correlations beyond their data base. Several methods can be used to achieve these objectives. One among them based on the fuzzy-set theory of Zadeh (Refs. B.6 and B.7) appears very promising in view of its successful application by Kubic and Stein (Refs. B.8 and B.9) to a problem concerned with quantifying model uncertainties and system similarity.

The fourth element: *Application,* which consists of Step 7, is designed to satisfy requirements 7 and 8, that is, to provide the same methodology that can be used to:

1. Design and operate test facilities.

2. Evaluate test data accuracy and/or applicability.

3. Evaluate correlations and/or model accuracy and applicability.

4. Evaluate computer code scaleup capability and applicability.

It should be noted that treating experiments and code model/correlation development in a systematic and unifying manner (by applying the same scaling methodology to both activities) is a prerequisite for quantifying efficiently and more accurately code uncertainties to calculate the postulated severe accident scenarios in a full-scale nuclear power plant.

B.5 Application and Demonstration

A program has been initiated at the Brookhaven National Laboratory with the objectives to:

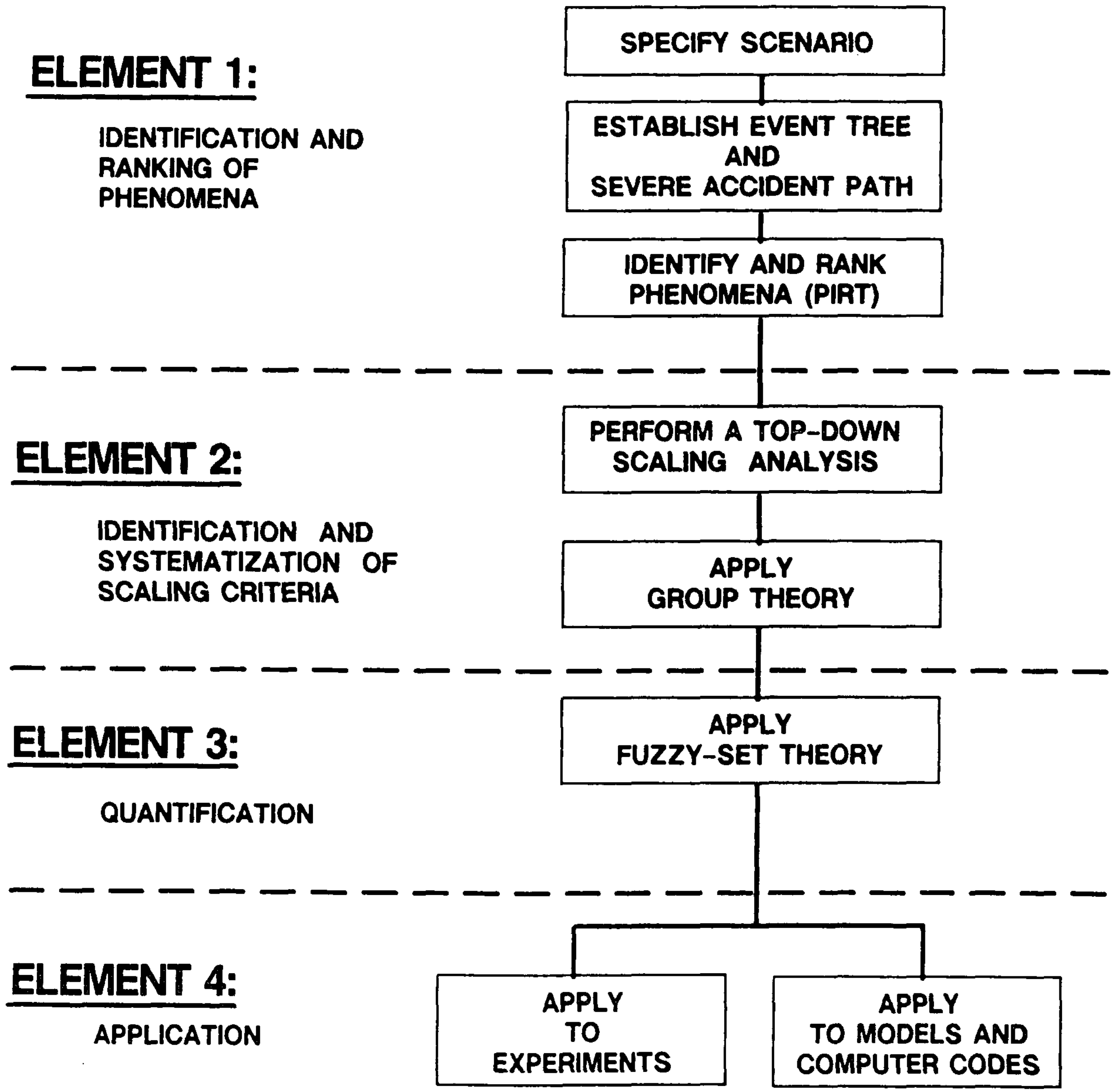

Figure B.1 Severe Accident Scaling Methodology (SASM)

1. Develop a scaling methodology (SASM) that meets the eight requirements listed above, and

2. Demonstrate the methodology by applying it to the DCH problem.

Furthermore, a Technical Program Group (TPG) has been formed to assist the staff and provide guidance to this work. The members were selected on the basis of their:

1. Knowledge of severe accident phenomena and issues, and/or

2. Internationally recognized expertness in modeling and scaling of complex systems and phenomena.

In order to provide input from a broad spectrum of technical sources, the composition of the TPG was specifically designed to include technical talent from universities, national laboratories, and industry.

This work, that is, the development and demonstration of SASM, is expected to be completed by the end of December 1989.

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11. ABSTRACT *(200 words or less)*

The revised Severe Accident Research Program has been prepared by the Office of Nuclear Regulatory Research to support the tasks and objectives discussed in the staff's "Integration Plan for Closure of Severe Accident Issues," SECY-88-147. The revised SARP addresses both the near-term research directed at providing a technical basis upon which decisions on important containment performance issues can be made, and the long-term research needed to confirm and refine our understanding of severe accidents.

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