NUREG–1365
Rev. 1

Severe Accident Research Program Plan Update

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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ABSTRACT

In August 1989, the staff published NUREG-1365, "Revised Severe Accident Research Program Plan." Since 1989, significant progress has been made in severe accident research to warrant an update to NUREG-1365. The staff has prepared this SARP Plan Update to: (1) Identify those issues that have been closed or are near completion, (2) Describe the progress in our understanding of important severe accident phenomena, (3) Define the long-term research that is directed at improving our understanding of severe accident phenomena and developing improved methods for assessing core melt progression, direct containment heating, and fuel-coolant interactions, and (4) Reflect the growing emphasis in two additonal areas--advanced light water reactors, and support for the assessment of criteria for containment performance during severe accidents.

The report describes recent major accomplishments in understanding the underlying phenomena that can occur during a severe accident. These include Mark I liner failure, severe accident scaling methodology, source term issues, core-concrete interactions, hydrogen transport and combustion, TMI-2 Vessel Investigation Project, and direct containment heating. The report also describes the major planned activities under the SARP over the next several years. These activities will focus on two phenomenological issues (core melt progression, and fuel-coolant interactions and debris coolability) that have significant uncertainties that impact our understanding and ability to predict severe accident phenomena and their effect on containment performance. The SARP will also focus on severe accident code development, assessment and validation. As the staff completes the research on severe accident issues that relate to current generation reactors, continued research will focus on efforts to independently evaluate the capability of new advanced light water reactor designs to withstand severe accidents.

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EXECUTIVE SUMMARY

In May 1988, the staff prepared an "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147) for the Commission. Many of the elements of this integration plan require a basic understanding of severe accident phenomena. In fact, one of the major elements of the plan is a Severe Accident Research Program (SARP). Following the issuance of SECY-88-147, the NRC staff, with input from DOE laboratories and consultants from universities and industry, identified and prioritized significant technical issues in order to focus the research efforts needed to close severe accident issues.

In August 1989, the staff published NUREG-1365, "Revised Severe Accident Research Program Plan." This plan was organized in two parts. One was oriented toward the short-term resolution of issues related to early containment failure (i.e., direct containment heating, BWR Mark I liner melt-through) and methodologies to evaluate and analyze these phenomena (i.e., scaling analysis, severe accident codes). The second part of the SARP Plan was oriented toward providing long-term confirmatory information to support the assessments of a broad spectrum of severe accident phenomenology associated with the major elements of SECY-88-147.

The overall near-term goals of the plan were to provide the technical bases for assessing containment performance over the range of risk-significant core melt events and to develop the capability to evaluate the efficacy of generic containment performance criteria. The long-term goals of the plan were to provide an improved understanding of the range of phenomena expected during severe accidents and to develop improved methods for assessing fission product behavior and release in the event of containment failure during severe accidents. Since 1989, significant progress has been made in severe accident research to warrant an update to NUREG-1365. The staff has prepared this SARP Plan Update to:

1. Identify those issues that have been closed or are near completion,

2. Describe the progress in our understanding of important severe accident phenomena,

3. Define the long-term research that is directed at improving our understanding of severe accident phenomena and developing improved methods for assessing core melt progression, direct containment heating, and fuel-coolant interactions, and

4. Reflect the growing emphasis in two additional areas—advanced light water reactors, and support for the assessment of criteria for containment performance during severe accidents.

The focus of the SARP remains on regulatory research to address technical issues of concern regarding containment performance and release of fission products in the event of containment failure.

The priority for severe accident research depends on whether there is a high likelihood that additional research will result in a significant reduction in uncertainty and on the knowledge necessary for the conclusions to be widely accepted in the technical community. For some issues it may not be necessary to reduce uncertainties further, since other issues may dominate overall uncertainty. For these and other issues for which it may not be practical to attempt to reduce uncertainties further, the staff recognizes that some regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

Table 1 provides a brief description of the program, status, and major milestones for each of 11 major severe accident issues in the SARP. One of the goals of the SARP is to complete all the major severe accident experimental programs within the next 2 to 3 years. Assuming a relatively constant level of funding for the SARP, closure of all severe accident issues is anticipated in 4 years.

Major Accomplishments

Considerable progress has been made in recent years in understanding the underlying physical and chemical phenomena that can occur in a severe accident. The staff developed programs consistent with the Integration Plan for Closure of Severe Accident Issues (SECY-88-147) to obtain key data and information needed to make an informed decision on issues and phenomenological uncertainties associated with accident sequences that could potentially lead to early containment failure. This information is essential for assessing potential safety improvements and for making decisions on whether or not particular improvements are warranted. As pointed out in the Commission's Severe Accident Policy Statement, such decisions should be based on a combination of engineering judgment (i.e., a deterministic method of setting and assessing safety margins) and the application of probabilistic risk assessment techniques to evaluate the likelihood of the occurrence of low probability events.

A summary of the recent major accomplishments and future efforts under the SARP is given below.

1. *Mark I Liner Failure*

   Completion of the initial study of BWR Mark I containment shell failure was documented in NUREG/CR-5423, "The Probability of Liner Failure in a Mark-I Containment" (July 1991). An extensive peer review indicated that the methodology employed to resolve the Mark I liner issue was sound and no major deficiencies or problems were identified that would invalidate the results. The peer review also identified the need for confirmatory evaluation of a number of subjects. That follow-on work is under way in fiscal year 1992.

2. *Severe Accident Scaling Methodology*

A severe accident scaling methodology (SASM) was developed to guide the formulation of experimental programs and analytical methods. Documentation of the SASM and application of the methods to direct containment heating (DCH) was addressed in NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accidents Technical Issue Resolution" (Draft for comment, November 1991). Work on this issue is now complete. Application of the results of SASM to scaling and operation of experimental facilities and modeling development for reactor analysis are pursued under the direct containment heating (DCH) research program described in Section 2.1.

3. *Source Term Issues*

The NRC has sponsored numerous experimental and analytical research projects on fission product release and transport. Early experiments and analytical work tended to focus on release from fuel material under severe accident conditions. Later, experimental data were obtained on the behavior of aerosols in reactor coolant systems and containment. These data were used to develop aerosol deposition and transmodels to analyze fission product behavior in the reactor coolant system (RCS) and containment. Currently, fully integrated model (VICTORIA) are in the process of being completed to analyze in-vessel release from fuel and retention in RCS. The CONTAIN code is being developed to analyze for the ex-vessel source term, including the transport of fission products, condensation of vapors, agglomeration and settling of aerosols, and chemical reactions in the containment.

Technical evaluations to support the revision to current methods for specifying the chemical form of the iodine entering the containment following a severe accident (TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962) were completed. The Oak Ridge National Laboratory performed analyses using a chemical kinetic model to determine the equilibrium distribution of the iodine, cesium, hydrogen, and steam species entering the containment. The results indicate that iodine entering the containment is at least 95% CsI (Cesium Iodide) with the remainder I (elemental iodine) or HI (hydrogen iodide). Once in the containment, CsI is expected to deposit onto surfaces and dissolve in water pools forming I^- (iodide) in solution. Subsequently, iodine behavior within the containment depends on the time and pH of the water solution. If pH control is available and maintained, little of the dissolved iodine will be converted to elemental iodine.

One remaining area to complete the test series is related to severe accident situations where air ingressed into the core either by natural circulation or from the residual heat removal system will react exothermically with the cladding producing high temperature in the fuel. Large vaporization of ruthenium, tellurium, and molybdenum will occur.

4. *Core-Concrete Interactions*

The NRC has conducted an extensive program of analytic and experimental research to obtain an improved understanding of core-concrete interactions. In FY91, an interim version (CORCON MOD3) of the core-concrete interaction code, CORCON, was released. The final version of CORCON MOD3 is expected to be released by fall 1992. In addition to improved thermal-hydraulic modeling, CORCON MOD3 incorporates the VANESA model of aerosol generation and radionuclide release during core-concrete interactions. Also, the NRC's experimental program on core-concrete interactions has been completed. The remaining work on this issue is to validate CORCON MOD3 against the experimental data. This validation is currently underway.

5. *Hydrogen Transport and Combustion*

Computer codes on hydrogen transport and combustion have been developed to evaluate the static or dynamic pressure loads from hydrogen combustion and detonation in containment. Results of these evaluations should enable the staff to make regulatory decisions to assess the potential threat to containment integrity. Since combustion processes are complex, many aspects are still not well understood. For example, for a premixture of hydrogen-air-steam in containment at elevated temperatures, but below the auto-ignition temperature, flame acceleration and high-speed combustion may lead to a transition to detonation. Research on some of these phenomena is continuing. Although the current research is intended to reduce uncertainties, reduced uncertainties are not required in some cases to make regulatory decisions. A joint agreement was reached for a cooperative program with the Ministry of International Trade and Industry of Japan and the Nuclear Power Engineering Center. Under this program, high-temperature, high-speed hydrogen combustion research will be conducted for next 4 years.

6. *TMI-2 Vessel Investigation Project*

The objectives of the TMI-2 vessel investigation program are to investigate the condition and properties of material extracted from the lower head of the TMI-2 reactor pressure vessel, to determine the extent of damage to the lower head, and to determine the margin of structural integrity that remained in the pressure vessel. Significant progress has been made on these overall objectives. Examinations of the vessel samples from the TMI-2 lower head indicated that a small region of the lower head (approximately 2 feet in diameter) experienced inner surface temperatures of about 1350 °K which exceeded the transition temperature of the steel. The examinations also indicated that the temperature 2 inches into the wall was about 100 °K lower than the inner surface temperature. (The lower head thickness was 5 inches.)

7. *Direct Containment Heating*

Since the publication of NUREG-1365, considerable research on direct containment heating has been undertaken to provide new insights and an improved data base to answer the questions, What is the nature of the DCH threat, and what mechanisms and configurations exist ex-vessel that will mitigate or eliminate it? In order to quantify the pressure and temperature loads generated by DCH, phenomena and processes that mitigate or augment DCH loads to the containment must be considered.

To assist the NRC and its contractors in developing an experimental program and interpreting and analyzing test results, the staff convened a peer review group to evaluate the NRC's program to resolve the DCH issue. The peer reviewers meet regularly to assess progress and ensure that the research program objectives are being met.

Integral testing was initiated in fiscal year 1991 to investigate the containment loadings resulting from DCH. The experimental program will explore integral DCH phenomena at different scales for representative reactor designs. Separate-effects testing to confirm the validity of the assumptions employed in the scaling analysis was initiated in FY92. Details of the testing are provided in Section 2.1.

Major Ongoing Activities

Over the next several years, the major share of NRC's severe accident research program will focus on two phenomonological issues and on code development, validation and assessment. These issues, core melt progression, and fuel-coolant interactions and debris coolability, have significant uncertainties that impact our understanding and ability to predict severe accident phenomena and their effect on containment performance. Severe accidents codes will continue to be developed to reflect the current understanding of severe accident phenomena and will be validated against experimental data. Decisions on when code development is completed require a balance between the level of precision needed and the level of uncertainty or variability in models of severe accident phenomena that are acceptable for regulatory decision making.

1. *Core Melt Progression*

In-vessel core melt progression describes the state of an LWR reactor core from core uncovery up to reactor vessel melt-through in unrecovered accidents, or through temperature stabilization in accidents recovered by core reflooding. Melt progression provides the initial conditions for assessing loads that may threaten the integrity of the reactor containment.

A great deal of information has been obtained on the processes involved in the early phase of melt progression that extends through core degradation and metallic (but not ceramic) material melting and relocation. This information has come from integral tests in the following test facilities: (1) the Power Burst Facility (PBF); (2) the Annular Core Research Reactor (ACRR); (3) the Canadian National Reactor Universal (NRU); (4) the Japanese Nuclear Safety Research Reactor (NSRR); (5) the French Phebus test reactor; (6) the Loss of Fluid Test (LOFT) facility; and (7) the German CORA ex-reactor fuel damage test facility. Laboratory separate effects experiments have also contributed information on significant phenomena. Most of the available information on late phase melt progression has come from the post-accident examination of the Three Mile Island, Unit 2 (TMI-2) reactor.

A comprehensive draft, "Research Plan for Melt Progression Issue Resolution," was prepared in fiscal year 1991 to address the remaining core melt progression issues. The plan was subjected to expert peer review, and is currently being revised. This plan and comments of the reviewers were used as a basis to develop the plan discussed in Section 2.2 of this update.

Reactor pressure vessel lower head failure maps have been developed for local penetration failure and for local and global creep rupture failure as a function of the characteristics of the lower head debris and of the lower head structure. The analytical basis for these maps is supported by the results of the metallurgical analyses of the samples in the TMI-2 lower head program. A draft report, NUREG/CR-5642 "Light Water Reactor Lower

Head Failure Analyses," on this analysis was issued for comments in March 1992 and has undergone peer review. The final report will be issued in the later part of 1992.

2. *Fuel-Coolant Interactions and Debris Coolability*

There are three specific topics under this issue: fuel-coolant interaction (FCI) energetics, quenching in water pools, and adding water to a degraded core. Because of the large variability of scenarios and parameters that impact FCIs and debris coolability, the task of closing this issue is difficult. The plan involves fundamental, long-range elements, as well as assessments applying these fundamentals to specific reactor designs and accident scenarios. Over the last 10 years, the emphasis has gradually shifted from "energetics" aspects to "coolability" aspects relevant to accident management. However, the need for assessing containment integrity for advanced light water reactors (ALWRs) indicates that a balanced approach with continued research on energetic aspects is still warranted. These fundamental energetics aspects are being pursued at the University of California at Santa Barbara and at the University of Wisconsin. Results of this research are also expected to contribute to the understanding of the coolability experiments at FARO facility in Italy that NRC is sponsoring.

Two experimental research programs addressing ex-vessel debris coolability were initiated during FY91. The Advanced Containment Experiment (ACE) program, conducted at Argonne National Laboratory (ANL), is an internationally sponsored program with NRC participation. One phase of this program, Melt Attack and Debris Coolability Experiment (MACE), deals with melt coolability issues. So far, three tests, including a scoping test, have been performed under the MACE program. While the MACE tests involve prototypic debris composition, a separate NRC-sponsored program WETCOR, conducted at Sandia National Laboratories (SNL), supplements the MACE program and investigates the coolability issue of metallic and oxidic core debris under heated side wall conditions to determine the limits of coolability. Two tests were conducted in FY91 and in FY92 under the WETCOR program.

3. *Severe Accident Codes*

The principal integrated computer codes that support the evaluations of nuclear plant responses to postulated severe accident scenarios are being independently reviewed for their adequacy to meet NRC objectives. The first such comprehensive peer review was completed in FY91 for the MELCOR code; SCDAP/RELAP5 peer reviews started in FY92; and the CONTAIN and the VICTORIA code will be undertaken in the second quarter of FY93. The results of the peer reviews will focus on critical needs for completing code development.

Advanced Light Water Reactors

Research programs have been initiated to independently evaluate new ALWR (AP600 and SBWR) features, in particular the adequacy of these features to withstand severe accidents. These efforts will provide the technical basis for NRC's support of ALWR plant design certification.

Long-Term Plan

Even though we anticipate that closure of all major severe accident issues will be accomplished in 4 years, additional work will continue on severe accident research, although at a reduced level of effort. Residual issues will still need to be addressed. In addition, results of severe accident experiments and research that are being conducted world-wide will be used to continue to update and validate the severe accident codes. Severe accident research on ALWRs will continue as designs for these plants continue to evolve. It is likely that additional severe accident issues may arise in the future. Therefore, the NRC will continue to support severe accident research, at a somewhat reduced level, in order to improve our technical understanding and maintain the level of expertise needed to address future issues in this area.

Criteria for Termination of Research

The degree to which a severe accident technical issue must be resolved depends on the needs of the related regulatory decisions and on the necessary knowledge for the conclusions to be widely accepted in the technical community. Ideally, an issue is considered closed when the NRC can pronounce that sufficient experimental and analytical research has been completed to allow, for the purpose of IPEs (individual plant examinations), accident management studies, or containment performance evaluations, for the prediction of reactor plant response. In addition, uncertainties have been reduced sufficiently that regulatory decisions are either insensitive to further reductions in uncertainty or that the residual level of risk is considered low enough that further work is not justifiable.

In practice, deciding when work is completed requires a subjective assessment of the potential benefits of further research. Although these judgments are not easy, two different approaches are possible. The first approach is based on developing the analytical capability for predicting the complete evolution of severe accidents (under given initial and boundary conditions) in some reasonable level of detail. The other is focused on particular processes, relevant to specific containment integrity issues, assuming that these processes can be adequately characterized within a rather general assessment of the overall sequence. Accordingly, the emphasis in the first approach is on computer code development at the system level (i.e., MELCOR, CONTAIN, SCDAP/RELAP) and associated code validation or verification activities. In the second approach, the emphasis is on addressing specific physical processes and through them identify specific safety concerns, then focus the research to address the specific concern. An example of the second approach is that taken for the assessment of Mark–I liner attack in NUREG/CR–5423.

These two approaches are not, certainly, mutually exclusive; the two approaches will work well together. There is already a lot of momentum on the first approach; prototypic testing has been used to validate computer models, which in turn would be exercised over the range of conditions and uncertainties associated with the data and accident analysis. In fact, this approach may be even necessary for a widely available capability for severe accident assessments. However, it is also true that such use is only possible after the main issues have been resolved, and a research effort that is driven by this approach tends to be rather inefficient in addressing the issues themselves.

The second approach provides for self-correcting focusing on specific technical issues (such that they are well-posed for the research program), and it provides for gradual synergism and eventual convergence. This convergence constitutes closure of an issue and thereby provides the basis for terminating further research. We are expecting to use such an approach on all containment integrity issues. This latter approach on issues relating to containment integrity and will temper the approach by cognizance of the issue significance in terms of its overall risk impact.

An important element in the issue resolution process is the role of peer review. For all of its major programs, the SARP routinely incorporates the feedback from peer review groups both in the area of experimentation and in code development. The breadth of experience and expertise brought to bear on the problem by the review groups assists both the staff and its contractors in achieving closure.

Table 1 SARP Status and Milestones

Issue	Description	Status	Expected Completion	Reference Section
	Major Program			
Mark I Liner Failure	Develop probabilistic method to integrate SARP results into conditional liner failure probability. Results indicate that without water, liner failure is nearly certain; with water, liner failure is physically unreasonable.	Complete (NUREG/CR-5423)		Appendix A.1
	Residual Issues:			2.6
	1. Liner failure criteria	Complete		
	2. Melt superheat	Complete		
	3. Melt spreading analysis	Complete		
	4. Recalculation of the liner failure probability	Anticipated	December 1992	
	Major Program			
Direct Containment Heating	Develop analytical models for predicting entrainment of core debris from reactor cavity and its interaction with the containment atmosphere.			
	1. Integral-effect tests to simulate fundamental synergetic effects of high-pressure melt ejection.	Ongoing	December 1992	2.1
	2 Separate-effect tests to confirm rate or detailed spatially dependent information employed in the scaling analysis (e.g., corium jet disintegration, liquid film formation, entrainment, capture of corium in compartment).	Ongoing	February 1994	2.1
	3. Development and validation of system-level code	Ongoing	December 1992	2.1, 2.4
	Residual Issues:			
	Development of generalized assessment criteria for different cavity and compartment designs employed in LWRs.	Anticipated	February 1994	2.1
	Major Program			
Scaling	Develop severe accident scaling methodology to guide the experimental program.	Complete (NUREG/CR-5809)		Appendix B.3
	Residual Issues:			
	None			

Table 1 SARP Status and Milestones (continued)

Issue	Description	Status	Expected Completion	Reference Section
	Major Program			
Source Term	Obtain better understanding of fission product release and transport mechanisms in LWRs under severe accident conditions.	Complete		Appendix B.1
	Residual Issues:			
	1. Complete analysis of VI6 fission product release test conducted in FY92	Ongoing	September 1992	2.7
	2. Complete VICTORIA code development	Ongoing	September 1992	
	3. Validate source term codes with Phebus data	Anticipated	1999	
	4. Complete the fission product release data for severe accident associated with air ingression condition	In planning	September 1994	
	Major Program			
Core-Concrete Interaction	Obtain better understanding of the amounts of noncondensable gas generated from the interaction between molten core debris and the concrete and radioactive aerosols.	Complete		Appendix B.2
	Residual Issue:			
	Compare code predictions to test results.	Ongoing	January 1994	Appendix B.2
	Major Program			
Hydrogen Combustion and transport	Develop data base and computer codes to assess the containment performance due to hydrogen combustion. Data were obtained on subsonic and sonic combustion modes. Data were also obtained on hydrogen transport and mixing.	Complete		Appendix A.3
	Residual Issues:			
	Data obtained to date is limited to hydrogen-air-steam mixtures at ambient or low temperature. Obtain better understanding of the combustion phenomena of hydrogen-air-steam mixtures at high temperature represen-tative of severe accident conditions.	Ongoing	1996	2.6

Table 1 SARP Status and Milestones (continued)

Issue	Description	Status	Expected Completion	Reference Section
TMI–2 Vessel Investigation Project	*Major Program*			
	1. Extract specimens from the TMI–2 vessel	Complete		Appendix A.4
	2. Investigate the condition and properties of material extracted from the lower head of the TMI–2 reactor, determine the extent of damage, and determine the margin to failure.	Ongoing	June 1993	
	Residual Issues:			
	None			
Core Melt Progression and Hydrogen Generation	*Major Program*			
	Provide initial conditions (melt mass, rate of release, composition, and temperature) for assessing containment performance; describe the state of LWR core from core uncovery up to reactor vessel melt-through.			
	1. Early phase: core uncovery, clad ballooning, oxidation and hydrogen generation, eutectic material interactions, Zircaloy relocation.	Complete		Appendix A.2
	2. Late phase			
	a. Conditions for formation of metallic melt blockage in BWR vs. metallic melt drains from core when formed.	Ongoing	1995	2.2
	b. Conditions for ceramic pool melt-through from blocked core; determines the melt mass and characteristic of the melt released to the reactor vessel lower head.	Ongoing	1995	2.2
	3. Mode of vessel failure; determine rate of melt release into containment.	Complete		2.2
	Residual Issues:			
	1. Application of experimental results to models of governing phenomena in severe accident codes.	Ongoing	1996	2.4
	2. Model development for severe accidents recovered by core reflooding.	Ongoing	1993	2.2

Table 1 SARP Status and Milestones (continued)

Issue	Description	Status	Expected Completion	Reference Section
	Major Program			
Fuel-Coolant Interactions and Debris Coolability	Develop models or correlations and the technical bases for those models for predicting the broad range of resultant phenomena of interactions of degraded core materials with coolant (including steam and hydrogen generation rates) for various reactor geometries, meltdown scenarios, and timing of coolant addition.			Appendix A.5
	1. FCI energetics—relevant to in-vessel interactions in PWR.			
	a. Premixing experiments to determine the extent of fuel and coolant intermixing that causes the region to become pressurized.	Ongoing	September 1992	2.3
	b. Fragmentation experiment to determine the rapid increase of fuel surface area causing vaporization of more coolant and increasing local pressurization.	Ongoing	September 1992	2.3
	c. FCI yield experiment to determine the expansion work associated with the triggering and explosion of premixture.	Ongoing	September 1993	2.3
	2. Fuel melt quenching experiment to determine the conditions under which FCI leads to coolable configuration as occurred at TMI-2.	Ongoing	January 1994	2.3
	3. Reflooding—relevant to in-vessel and ex-vessel issues related to operator adding water to achieve stable coolable configuration.			
	a. In-vessel reflooding issues—includes recriticality and hydrogen generation	Complete		2.3
	b. Ex-vessel reflooding to mitigate core-core interactions	Ongoing	March 1993	2.3
	Residual Issues:			
	The models or correlations devel oped under the FCI research program will be validated against available experimental data. Perform selective plant analyses.	Anticipated December 1993		

Table 1 SARP Status and Milestones (continued)

Issue	Description	Status	Expected Completion	Reference Section
	Major Program			
Severe Accident Codes	1. Develop, validate, and document computer codes to analyze severe accident phenomena and issues for light water reactors. The code development program is an ongoing iterative process; codes are developed and assessed against selected experimental data; model improvements are identified; new models for new phenomena are implemented and assessed.	Ongoing	December 1994	2.4
	2. Extensive assessment and validation against experimental results from domestic and international programs will continue to increase the confidence in the codes.	Ongoing	1999	
	Major Program			
Advanced Light Water Reactors	1. Assess the methodologies used to evaluate containment cooling concepts, including natural circulation cooling and external spray cooling of containment.	Ongoing	September 1992	3
	2. Assess the methodologies used to evaluate the effects of phenomena resulting from ALWR fuel design on severe accidents.	Ongoing	December 1992	3
	3. Apply severe accident codes (e.g., SCDAP/RELAP, MELCOR, CONTAIN) to the analysis of ALWRS.	Anticipated 8–12 months following receipt of vendor's design information		3
	Residual Issues:			
	None.			

1 SEVERE ACCIDENT RESEARCH PLAN

1.1 Background

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 (CPI) 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 (SARP).

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.

All these elements are mutually supportive and interrelated, but the severe accident research program is the common source of information, and hence of central importance. Elements 1, 2, and 6 particularly depend on our understanding of severe accident phenomena and phenomenological sequences, and it is the purpose of this program to provide this understanding.

Following the issuance of SECY–88–147, the staff established several groups of experts to help identify and define the status of SARP activities (item 4 above). Each group was composed of contractor personnel from DOE laboratories, consultants from universities and industry, and NRC staff. These groups considered detailed technical issues to define their status and the need for further research, to identify and focus research necessary for sound regulatory decisions to be made within the framework of the integration plan, and to prioritize the research activities needed to close severe accident issues. The staff used the experts' input as a basis for preparing the revised SARP plan that was published in August 1989 as NUREG–1365, "Revised Severe Accident Research Program Plan." This plan was structured in two parts: one oriented to the short-term resolution of certain pressing issues, and the other oriented to the long-term confirmatory support, and perhaps refinement, of current assessments in a broad-spectrum-coverage of the whole severe accident phenomenology. The specific issues identified in the short-term part of the plan were: Mark I liner attack, direct containment heating, severe accident codes, and scaling—the last two being primarily of methodological thrust.

As stated in NUREG–1365, the overall goals of the revised SARP plan are to provide the technological base for assessing containment performance over the range of risk-significant core melt events, develop the capability to evaluate the efficacy of generic containment performance improvements, provide an improved understanding of the range of phenomena exhibited by severe accidents, and reduce the uncertainties in the source term sufficiently to enable the staff to make regulatory decisions on severe accident issues. Since the issuance of NUREG–1365, the SARP has generated a large amount of data and insights into the progression of severe accidents. According to the revised SARP plan, the research efforts in the past 3 years focused on the short-term issues listed above; however, significant progress was also made on certain elements of the long-term plan, including in particular corium-concrete interactions and the chemical form of radioiodine in severe accidents. In addition to the improved understanding of severe accidents achieved in the past 3 years, the present plan (referred to as SARP plan update) was motivated by the following considerations:

1. Identify those issues that have been closed or near completion,

2. Describe the progress in the maturation and evolution of our understanding of important severe accident phenomena,

3. Define the long-term research that is directed at improving our understanding of severe accident phenomena and developing improved methods for assessing core melt progression, direct containment heating, and fuel-coolant interactions, and

4. Reflect the growing emphasis in two additional areas—advanced light water reactors, and support for the assessment of criteria for containment performance during severe accidents.

1.2 Discussion

As in NUREG–1365, the focus of the SARP remains on research efforts to address technical issues of concern regarding the health and safety of the public. Technical issues regarding containment performance are of particular regulatory significance. The two issues identified in NUREG–1365 that could potentially lead to early containment failure are the Mark I liner attack and direct containment heating (DCH). At this time, the Mark I issue is near resolution; a plan to bring closure to the DCH issue will be in place by the end of CY92. With respect to the long-term research, it is important here to achieve a definition that will produce the sought-after understanding within a reasonable time schedule. At the same time, it is important that this understanding is directly relevant to safety to adequately support elements 1, 2, 5, and 6 of the Integration Plan. Carrying out the long-term research plan requires a lot of judgment and an atmosphere of open communication among all parties involved. It also requires clear communication and a deliberate approach in planning, reviewing, and conducting research. One purpose, then, of this plan is to address the methodological and practical aspects of meeting these requirements in future research efforts.

Consistent with this philosophy, the staff has proceeded with the implementation of the revised SARP, making extensive use of debate and deliberation of expert peer review to address a good number of issues. The expert reviews include the technical program group on severe accident scaling methodology and review panels for core melt progression research, for direct containment heating experiments, and for severe accident codes. In addition, the Mark-I liner attack study (NUREG/CR-5423) was subjected to extensive peer review and to smaller specialized panels for follow-up activities. These activities have proven very fruitful in sharpening the staff's judgment concerning the practical aspects of implementing the revised SARP and will be continued in the implementation of this SARP update.

1.3 Benefits of Additional Research

The phenomena under investigation and their combination into severe accident scenarios are highly complex—usually involving multiphase or multicomponent interactions, rapid transients, and multidimensional behavior. Since full-scale experiments are not possible, the in-depth understanding presents quite a formidable task. On the other hand, experience has shown that dominant mechanisms and evaluation methodologies can be found to provide an understanding quite adequate for the intended purposes. The special challenge in planning and conducting the severe accident research plan is to identify and use effective approaches, focusing quickly and studying in depth all the essential aspects of the behavior. In fact, a continuing focusing and refocusing process is required, as new research results provide the basis for improved judgments as to where to expend future efforts.

Which severe accident phenomena are studied, how conclusions about these phenomena are reached and defended, and when research on specific phenomena and issues should be stopped have traditionally been highly charged areas of debate. The impact of improved understanding in a certain area of phenomenology on risk reduction is not always readily apparent. One example is core melt progression research, which is one of the most complicated technical areas in severe accident phenomenology. Core melt progression impacts hydrogen generation (timing and quantity) and core debris relocation into the lower plenum (timing, quantity, rate, and composition of debris). The latter aspect, in turn, impacts fuel-coolant interaction (FCI) during and after the relocation (energetic, coolability), thermal loading on the lower head (mode and timing of failure), and release of core debris materials to the containment. Eventually there are important implications for containment integrity (i.e., energetic FCI, DCH or Mark I liner attack). As NUREG/CR-5030, "An Assessment of Steam-Explosion-Induced Containment Failure" (February 1989), and NUREG/CR-5423 have shown, conservative treatment of uncertainties is feasible to yield bounding and acceptable results for the alpha-mode containment failure and Mark I liner attack issues, respectively. For these two issues, the main benefit of learning more about core melt progression phenomena will be related primarily to improved understanding of the margin of conservatism in the analyses to resolve these issues.

From a containment integrity standpoint, learning more about core melt progression provides the input for containment loading for DCH. If DCH were amenable to the same approach as the alpha-mode containment failure and containment liner attack issues (see section 2.1), the value of core melt progression research would be in improved knowledge in the area of accident management when core melt progression is considered in the presence of water as it would in a recovered scenario. The payoff would be in improved estimates of risk, even though the accident scenario is of exceedingly low probability and the risk is already low, if we can demonstrate that accidents can be terminated even at relatively advanced states of their progression. The excellent safety record of nuclear power plants leaves only such exceedingly low probability, high consequence events to be of concern. Again, even though the accident sequence leading to vessel failure is of low probability, lack of adequate understanding of core melt progression leads us to use conservative assumptions to provide answers to a set of rather intangible questions that we are often asked. For example, for ex-vessel debris coolability, a core debris coolability criterion of 0.02 m^2/MWt was proposed. This criterion is intended to be used in sizing the reactor cavity for ALWRs. The existing data base for sustained melt-concrete-coolant experiments suggests that debris depth and the formation of crust on the surface of the melt may inhibit water ingression into the melt, which is necessary to produce rapid cooling. If more realistic assumptions are used, the debris bed depth would be substantially smaller than is currently investigated experimentally (75%–100% of total core). The core melt progression research can provide valuable information on the amount of molten material that can enter the containment at the time of vessel breach.

As was the case when NUREG-1365 was developed, the staff recognizes that for some issues it may not be practical to attempt to reduce uncertainties further, and some regulatory decision or conclusion will have to be made with full awareness of existing uncertainties. Expert elicitation (as done in NUREG-1150) in the Level II PRA serves only to reveal rather than to resolve issues since it deals with phenomenological issues at a high level. Quite often experts rely on parametric codes in which the author has programmed his opinion of what happens; real important phenomena may never be discovered. Furthermore, the degree of judgment, and therefore the degree of confidence associated with it, depends on the particular issue and the nature of the phenomenology involved. The priority and manner in which severe accident research is conducted depend on whether there is a high likelihood that such research will result in a significant reduction in risk uncertainty and on the depth necessary for the conclusions to be widely accepted in the technical community.

This research plan is structured to examine the issues in depth and develop the necessary methodological tools that can soundly address these issues. Therefore, one of the goals of the SARP is to complete all the major severe accident experimental programs within the next 2 to 3 years. Results of severe accident experiments and research that are being conducted world-wide will be used to continue to update and validate the severe accident codes.

1.4 Criteria for Termination of Research

The degree to which a severe accident technical issue must be resolved depends on the needs of the related regulatory decisions and on the necessary depth for the conclusions to be widely accepted in the technical community. Ideally, an issue is considered closed when the NRC can pronounce that sufficient experimental and analytical research has been completed to allow, for the purpose of IPEs, accident management studies, or containment performance evaluations, for the prediction of reactor plant response. In addition, uncertainties have been reduced sufficiently that regulatory decisions are insensitive to further reductions in uncertainty.

In practice, deciding when work is completed requires a subjective assessment of the potential benefits of further research. Although these judgments are not easy, two different approaches are possible. The first approach is based on developing the analytical capability for predicting the complete evolution of severe accidents (under given initial and boundary conditions) in some reasonable level of detail. The other is focused on particular processes, relevant to specific containment integrity issues, assuming that these processes can be adequately characterized within a rather general assessment of the overall sequence. Accordingly, the emphasis in the first approach is on computer code development at the system level (i.e., MELCOR, CONTAIN, SCDAP/RELAP) and associated code validation or verification activities. In the second approach, the emphasis is on addressing specific physical processes and through them identifying specific safety concerns, then focusing the research to address the specific concern. An example of the second approach is the one taken for the assessment of Mark-I liner attack in NUREG/CR-5423.

These two approaches are certainly not mutually exclusive; the two approaches work well together. There is already a lot of momentum on the first approach; prototypic testing has been used to validate computer models, which in turn would be exercised over the range of conditions and uncertainties associated with the data and accident analysis. In fact, this approach may be necessary for a widely available capability for severe accident assessments. However, it is also true that such use is possible only after the main issues have been resolved, and a research effort that is driven by this approach tends to be rather inefficient in addressing the issues themselves.

The second approach provides for self-correcting focusing on specific technical issues (such that they are well posed for the research program), and it provides for gradual synergism and eventual convergence. This convergence constitutes closure of an issue and thereby provides the basis for terminating further research. We expect to use such an approach on all containment integrity issues.

An important element in the issue resolution process is the role of peer review. For *all* of its major programs, the SARP routinely incorporates the feedback from peer review groups both in the area of experimentation and in code development. The breadth of experience and expertise brought to bear on the problem by the review groups assists both the staff and their contractors in achieving closure.

1.5 Organization of the Report

Our intent is to make this research plan scrutable; thus, when pertinent technical details cannot be given by reference, they are included in appendices at the end.

Chapter 2 points out the main directions for future severe accident research, and Chapter 3 discusses planned research efforts for ALWRs. Appendices A and B of this report discuss all the major phenomenological areas of severe accidents and progress to date on these areas.

This report has benefited from reviews by the Advisory Committee on Reactor Safeguards, Nuclear Safety Research Review Committee, and the Office of Nuclear Reactor Regulation.

2 RESEARCH PLAN

Introduction

The programs described in this chapter of the updated SARP are the high priority items that need to be addressed in the next few years, namely

1. Core melt progression,
2. Direct containment heating,
3. Fuel-coolant interactions and debris coolability, and
4. Severe accident codes

This chapter presents the research needs, our current research programs, and anticipated results for each of the above issues. Research programs on other residual issues are also discussed in this chapter. Appendices A and B present a brief summary of the current state of knowledge on the core melt progression and fuel coolant interaction issues listed above as well as other severe accident issues. The focus of this research is to provide validated codes to assess the performance of nuclear power plants with severe accidents and to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases. NUREG–1150 and other PRAs have shown that events that lead to early containment failure (such as several hours after severe core damage progression starts) have the greatest risk significance. As the accident progresses through its various stages, various kinds of containment loads are a consequence of the prevalent material interactions. The intensity of these loads depends on the quantities, composition, and temperature of the material involved and the overall configuration of contact.

While the NUREG–1150 study did not produce quantitative measures of the risk importance of specific melt progression phenomena, the results show that both the magnitude and uncertainty of melt processes are important. For example, the observation that the risk significance of DCH has decreased relative to earlier studies can in part be attributed to an improved understanding of melt progression that led to improved DCH testing procedures. Further exploration of the NUREG–1150 information base together with a sensitivity study has produced quantitative information on the risk importance of melt progression phenomena.[1]

Several of the conclusions from this study are:

1. In-vessel issues directly influence the low containment failure probabilities and the low risk values obtained for Surry (depressurization of the primary system and arrest of core damage before vessel breach). The regression analyses would not reveal this sort of insight.

2. Although the regression analysis had limitations, and care should be taken in interpreting the results, the regression analysis showed that the following in-vessel issues are important to the uncertainty in NUREG–1150 results: the fraction of each radionuclide group in the core released to the vessel before vessel breach, pressure rise in the containment at vessel breach (implies that pressure of the reactor coolant system (RCS) at the time of vessel breach, the mode of vessel breach, and the fraction of the core released at vessel breach are important), in-vessel hydrogen production, and alpha-mode failure.

3. The probability of failure of the Surry containment is low. However, within the conditional containment failure probability that does exist, the contributions from different modes of vessel breach are extremely varied—the conditional probability of containment failure at vessel breach for vessel failures at high pressure is several orders of magnitude higher than the conditional probability of containment failure at vessel breach for vessel failures at low pressure. This indicates that in-vessel issues could be critically important to containment failure probability at plants that are more susceptible to overpressurization than Surry.

4. The magnitude of source term releases at Surry are significantly impacted by the mode of vessel breach. The risk of prompt fatalities from a high-pressure melt ejection mode of vessel breach and an early containment failure is several orders of magnitude higher than the risk of prompt fatalities from vessel breach at low pressures with an early containment failure. This difference is due to the high rate of aerosol generation from the molten debris during a high-pressure melt ejection. This risk at Surry is low, but could indicate a high source term sensitivity to the mode of vessel breach at the other plants.

5. In-vessel issues are especially important from an accident management point of view. The timing of vessel breach is important for planning operator actions. Also, during recovery stages, the predicted response of the molten core to water injection would be useful information for the operator to have.

In summary, the evidence in NUREG–1150 indicates that in-vessel melt progression issues are important to risk and the uncertainty in risk.

[1]Letter from Fred Harper, SNL, to Brian Sheron, NRC, dated January 24, 1992.

In general, core melt progression research provides important input for all containment failure issues, such as the amount, composition, and temperature of the melt released to the containment and the mode of vessel failure. In addition, since accident management is considered to be a practical measure for mitigating the consequence of severe accidents, an understanding of degraded core cooling and other phenomena related to fuel-coolant interactions is needed to assess accident management strategies. Sections 2.2 and 2.3 provide detailed NRC research programs to address these issues.

At the time the revised SARP was issued in 1989, the issues of high-pressure melt ejection and DCH had been the subject of considerable study over the preceding several years. The results of these previous studies increased concerns about these issues, and it did not then appear likely that further research could lead to their resolution. Recent understanding of the core melt progression, the completion of the lower head failure analysis program, the completion of the severe accident scaling methodology, and restructuring of the DCH experimental program are all contributing significantly to the near-term closure of this issue. Section 2.1 provides the detailed NRC research program to address the direct containment heating issue.

Section 2.4 discusses the codes being developed and maintained by NRC. The MELCOR code has become the NRC-supported systems analyses code that replaced the source term code package. A few detailed mechanistic codes will also be maintained that are applicable to specific severe accident phenomena.

Sections 2.5, 2.6, and 2.7 discuss the BWR Mark I containment liner, hydrogen combustion, and source term issues, respectively.

2.1 Direct Containment Heating

2.1.1 Research Needs

The ultimate needs from research on DCH are analytical models or correlations and the technical basis for those models for predicting the entrainment of core debris from the reactor cavity and its interaction with the containment atmosphere and blowdown gas during a high-pressure melt ejection.

Thus, in addition to predicting the dispersal of debris from the reactor cavity, DCH modeling must also be capable of evaluating or suitably accounting for:

1. Debris fragmentation—the surface area for heat transfer and reactions, translated into a particle diameter
2. Debris to gas heat transfer—the embedded debris velocities, residence time
3. Debris trapping—the capture of debris within subcompartment volumes or on intervening structures
4. Hydrogen generation—resulting from oxidation of metallic debris constituents
5. Hydrogen combustion—resulting from reaction of any preexisting hydrogen in the containment along with hydrogen generated during the high-pressure melt ejection
6. Water vaporization—as a consequence of debris-water interactions in the cavity or other containment regions.

Ideally, modeling should also address parameters, process times, and spatial dependencies as well as their relationship to, in the SASM parlance, control parameters (e.g., RCS pressure).

While not strictly related to DCH, the precursor processes that influence melt ejection from the reactor vessel, gas blow through, and hole ablation are also necessary components of the overall calculation.

Modeling of DCH must also reflect the effects of the diversity of plant designs insofar as plant-specific features influence the phenomena. There are some 18 different cavity and lower containment subcompartment designs in use in the U.S. commercial PWR nuclear power plants.

Past research has indicated that while the cavity geometry influences the extent of dispersal from that region at elevated RCS pressure, differences in design configuration (at least for the designs experimentally investigated) have less impact on the entrainment fraction. As part of the SASM technical program group efforts to develop a scaling methodology for DCH testing, a correlation for entrainment fraction (i.e., the fraction of debris dispersed from the reactor cavity) was developed and assessed against a range of experimental data at different scales (1/42, 1/25, 1/10 linear scales) for different designs (Surry, Sizewell, Watts Bar, Zion). This work suggests that the entrainment fraction can be correlated against a number of parameters with a notable dependence on the initial Euler number, and thus, the pressure ratio defined by the initial RPV pressure/cavity pressure. Figure 2.1.1, taken from NUREG-5809, illustrates the dependence, as predicted by the correlation of the entrainment fraction on the initial RPV to cavity pressure ratio for two different designs, Zion and Surry.

The general conclusion that debris dispersal from the cavity is nearly complete at elevated RCS pressures has been qualitatively demonstrated in many of the past DCH experimental programs, regardless of the direct

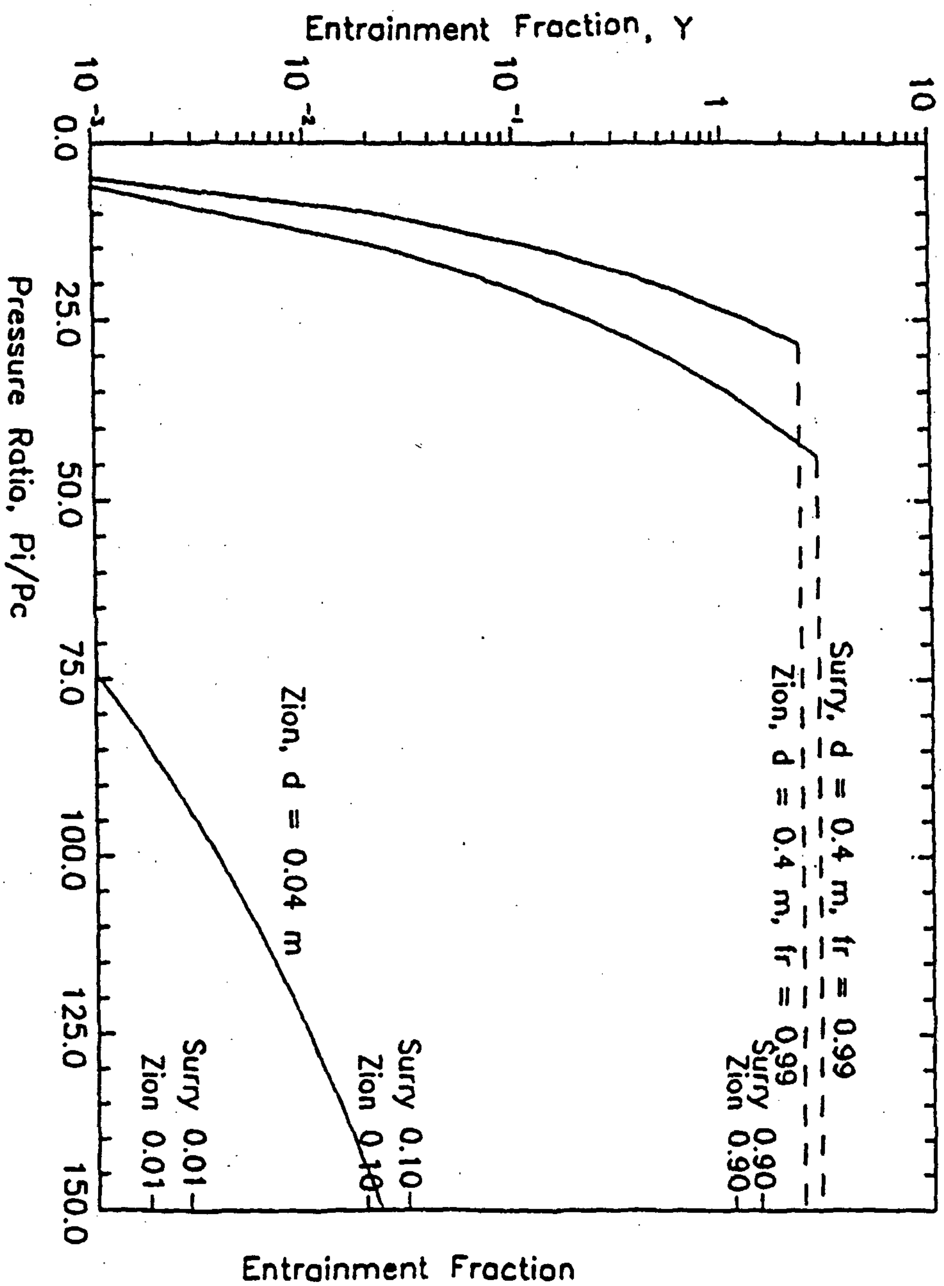


Figure 2.1.1 Entrainment Versus Pressure Ratio for Steam Blowdown.

scalability of individual tests. While this preliminary conclusion needs to be confirmed by additional testing and scaling analysis, the results of work to correlate entrainment fraction culminating in the plant extrapolation shown in Figure 2.1.1 indicate that research is needed to address the other major processes, e.g., subcompartment trapping and water interactions, which may have a significant effect on DCH loadings. Furthermore, correlation of entrainment fraction data indicates that testing need not represent the full range of possible RCS pressures (up to the power-operated relief valve setpoint), since complete dispersal out of the cavity is potentially achievable at much lower pressures. Interestingly, the work to correlate entrainment fractions also suggests that relatively simple models, or even quantitative empirical criteria, may be substituted in lieu of detailed mechanistic models.

While the details of the cavity configuration, for the limited range of variations considered, have been shown to have less effect on debris dispersal at elevated pressures, it is intuitive and has long been argued that compartmentalization and structures downstream of the cavity would trap or deentrain debris from the flow stream exiting the cavity on its way to the bulk containment. Since the free volume of the lower containment compartmentalized region of the containment is a small fraction of the total containment volume (~15% in the case of Zion) and the compartmentalized region immediately adjacent to the cavity exit is yet a smaller fraction of the total volume, it is apparent that trapping of debris in those regions would significantly decrease the ability to rapidly heat, by debris-gas heat transfer, the bulk atmosphere.

In addition to trapping or deentrainment of dispersed debris, the other first-order mitigative effect on the high-pressure melt ejection (HPME) process is the potential interaction of debris with water, either in the cavity or in the containment. Large quantities of water present in the containment during certain severe accident sequences can potentially quench or cool debris and thereby reduce containment pressurization.

While direct containment heating research has principally been concerned with the basic mechanisms by which molten debris could give up its latent and sensible heat to the atmosphere, a significant contribution to the pressurization of the containment arises from the potential oxidation of the metallic constituents of the core debris and the subsequent combustion of any hydrogen generated through that reaction. Assuming roughly 15% (by mass) of the core debris is unreacted zircaloy cladding, the corium thermal energy (sensible plus latent heat) is roughly equivalent to the chemical energy, on the order of 1 MJ/kg. Energy released from the combustion of the hydrogen produced by oxidation of that unoxidized cladding is also equivalent to roughly 1 MJ/kg of melt. If the hydrogen generated is consumed at the same rate at which it is released and mixed into regions containing sufficient oxygen, no hydrogen accumulates and the resulting combustion mode is a diffusion flame, either a buoyant plume or jet. Similarly, another requirement for this scenario would be the presence of an ignition source, hot debris particles in the case of an HPME, or the release and mixing of hydrogen at temperatures above an autoignition temperature. If the hydrogen is released at a rate faster than which it can be mixed with oxidant and burned or if there is no mechanism for initiating combustion upon initial release, the hydrogen may accumulate and the subsequent combustion, which can be presumed to occur after eventual mixing of gases to combustible levels, may take the form of detonation.

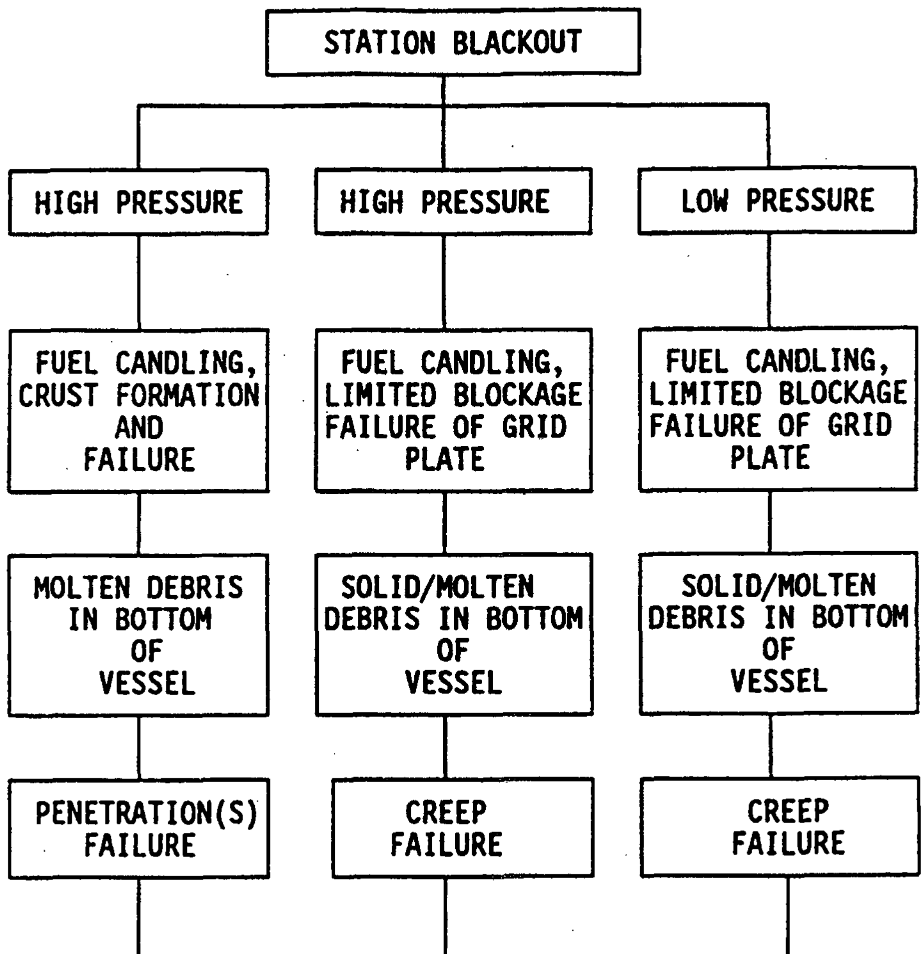
In order to provide the answers to the research needs described above and resolve the DCH issue, the NRC has developed a research program composed of three major elements, all of which are judged necessary to achieve closure: (1) integral testing, (2) separate effects testing, and (3) development and validation of a systems-level code (CONTAIN).

As described in Appendix B.3, "Scaling," a necessary precondition for the resumption of DCH testing was the development of a scaling rationale to guide the design and operation of planned tests. It was further determined that scaling evaluations would be based on the evaluation of conditions for a specific accident sequence and for a specific plant design in order to assure a more scrutable and meaningful comparison of the scaling results with reactor design and analysis.

The starting point for this scaling evaluation of appropriate initial conditions was the technical program group (TPG) review of existing core melt progression analyses for a station blackout at the Surry plant in order to assess the amount of core debris present in the lower head at the time of reactor pressure vessel failure. Three synthesized sets of initial conditions for DCH were developed; a set of conditions associated with a penetration failure of the bottom head and two sets of conditions (for both low pressure and high pressure) associated with creep failure of the bottom head. The quantities, compositions, and amounts of material involved are summarized in Figure 2.1.2., taken from NUREG/CR-5809. The evaluation of melt conditions for a station blackout at the Surry plant yielded the following general insights.

- About 40% of the core mass would be ejected in molten form.

- From 30 to 70% of the molten material could be in a metallic form. If failure of the crust or flow blockage from structural weakness is considered more realistic than crust or flow blockage melting, about 30 to



INITIAL CONDITIONS			
TIME OF FAILURE	~3 HOURS	~4 HOURS	~10 HOURS
EJECTED KGS	~40,000	~70,000	~80,000
MOLTEN KGS	~40,000	~40,000	~40,000
OXIDIZED ZR	~40%	~40-50%	~60-70%
EJECTED METALLIC CONTENT	~30-40%	~40-60%*	~50-70%*
TEMPERATURE	~2500 K	~2500 K	~2500 K

*INCREASED METALLIC CONTENT DUE TO MOLTEN STEEL

Figure 2.1.2 Initial Material Conditions for Direct Containment Heating.

50% of the molten material will be in a metallic form. (In both instances, the majority of the metallic component is steel; Zr is 10–25% of the molten debris.)

- A significant amount of solid debris (over one-half of the molten mass) will be present in the lower head for scenarios in which vessel failure is delayed until creep failure occurs. The majority of the solid debris may be retained in the vessel, at least in the time scale related to HPME. The mass of molten material for a creep failure scenario was virtually identical to that associated with a penetration failure.

To proceed with testing and validation of analytical models, the staff has assumed initial conditions associated with a station blackout sequence and a penetration failure of the reactor pressure vessel (RPV) bottom head. It is fully recognized that there is uncertainty surrounding the precise initial conditions for DCH; nonetheless, for prescribing scaled test conditions and conducting integral tests, we conclude that the best available information indicates adopting this approach. Simulation of creep failure, or for that matter multiple penetration failures, can be addressed by variation of the hole size used for testing.

2.1.2 Direct Containment Heating—Current Research Program

As summarized in section 2.1.1, the staff has outlined a research program consisting of (1) integral testing, (2) separate effects testing, and (3) analytical model development and validation.

The objective of integral testing is to simulate the fundamental synergetic effects of high-pressure melt ejection through experiments that include entrainment or sweepout of corium from the cavity, transport and trapping of debris in the lower containment subcompartments, oxidation of metallic constituents in the corium, combustion of hydrogen produced as a result of that oxidation of metals, and vaporization or heat transfer to the existing water inventory in the reactor cavity and containment. This integral testing is designed to investigate *both* the mitigative processes and phenomena that lessen the impact of an HPME as well as the inherent mechanisms contributing to the pressurization by DCH.

In terms of scaling, we sought to simulate the pressurization of the containment to the extent practicable without introducing distortions that significantly impact the simulation of the dominant phenomena influencing the fundamental behavior. As an example, tests conducted at less than full scale will produce conditions in which transported debris has a shorter residence time in the atmosphere; if debris velocities are preserved, a test at 1/10 scale would result in a much shorter residence time for debris to heat the atmosphere. Thus, preservation of debris-gas heat transfer in scaling may lead one to distort or even neglect principal mitigative mechanisms in an integral test. When those conflicts arose, scaling was driven by the objective of simulating the transport of debris to the bulk containment as opposed to direct simulation of pressurization. As a check on the possible distortion of phenomena, including concomitant atmosphere pressurization, counterpart integral tests using the same scaling principles are run at different scale.

Following the basic objective and guidelines for integral testing outlined above, the simulated RCS and containment features and volumes were geometrically (i.e., linearly) scaled, the RPV aspect ratio as well as the length scale from the RPV to the cavity was preserved. In keeping with our objective to simulate fundamental or global behavior in a consistent fashion and as a consequence of using a simulant material (iron-alumina thermite with chromium metal) rather than reactor materials, the melt mass employed in the tests was scaled based on the equilibrium energy/volume ratio. Because of the lower density of simulant relative to corium, scaling on an energy/volume basis results in a distortion of the amount of melt, volumetrically scaled, by approximately 48%. The melt mass employed in the 1/10 scale representation of Zion was 43 kg.

The staff has convened an independent peer review group to evaluate the program to resolve the DCH issue and specifically assess the program of integral testing. With the concurrence of the peer review group on our basic approach, the staff has proceeded to conduct integral tests at Sandia National Laboratories (SNL) in the 1/10th linear scale Surtsey facility and at Argonne National Laboratory (ANL) in the 1/40th linear scale COREXIT facility. The integral experimental tests currently under way at SNL and ANL are considerably different from the tests proposed prior to the development of the SASM. Specifically, these tests are improved because (a) the initial and boundary conditions for the tests are scaled for specific accident scenarios for specific plants, (b) important scaling groups have been matched or the distortions have been minimized for those tests in which matches cannot be achieved, (c) scoping tests have been conducted and instrumentation and procedures required to carry out HPME/DCH experiments are reliable and reproducible, and (d) test conditions include more prototypic conditions (i.e., steam-driven melts, realistic containment compartmentalization, sources of water, potential for hydrogen combustion, etc.). Integral effects tests are based on detailed modeling of the Zion and Surry plants, designs that are representative of two large classes of reactor designs. Scaled models of the reactor pressure vessel, reactor cavity, in-core instrument tunnel, and subcompartment structures were constructed. The subcompartment structures and equipment modeled included the crane wall, steam generators, reactor coolant pumps, seal table room, biological shield wall, refueling canal, radial beams and grating, and the operating deck.

On February 14 and June 1, 1992, the staff reconvened the DCH peer review group with the purposes of (1) describing the progress and results of the initial counterpart tests at 1/10 and 1/40 scale, (2) discussing preliminary conclusions and observations regarding scale effects, and (3) proposing the balance of our integral test program. Table 2.1.1 presents that portion of our integral test program proposed to be carried out at SNL. For completeness, tests that have already been conducted are listed along with the data of the test. Note that Table 2.1.1 identifies tests at a third and larger 1/6th scale that are proposed to be conducted. These tests would be performed using the existing Containment Technology Test Facility at SNL that has been used in the past to investigate containment integrity (containment leakage and failure beyond design pressure). As evident from the test matrix, the approach in integral testing is systematic; major synergetic effects are added in the progression of testing. For example, in order to experimentally measure hydrogen generation directly and to measure and observe the incremental effects of hydrogen combustion, the first integral test was conducted in an inert (99% N_2) atmosphere with later tests (IET–3,4,5,6) allowing combustion with an oxygen-bearing atmosphere. In similar fashion, the effects of cavity water inventory and containment water heat removal are included in the test program. Proposed counterpart testing in the 1/40 scale COREXIT facility at ANL was originally designed to mirror the Surtsey tests through IET–7. Other options under consideration include use of the ANL facility to explore issues related to the use of a simulant melt—the ANL facility has the capability for conducting tests with UO_2-based thermitic material. At this stage three tests have been conducted in the 1/40 scale facility, counterparts to IET–1, IET–3, and IET–6.

Relative to the research needs identified in subsection 2.1.1, the integral tests provide direct indication of the fraction of debris swept from the cavity (entrainment fraction), subcompartment trapping, hydrogen generation, and hydrogen combustion. Indirect information is derived for water vaporization, debris-to-gas heat transfer, and debris fragmentation. In the case of debris fragmentation, posttest analysis provides a direct measure of recovered debris, however, this is not a measure of the debris particles upon inception of entrainment nor is it a measure of characteristic debris geometry or surface area for melt that can not be recovered in particulate form (because of agglomeration of molten or semi-molten material).

While the integral tests provide valuable information and allow for a global evaluation of DCH, because of the difficulties associated with use of high-temperature reactive melts (instrumentation limitations) and their added complexity, these tests do not provide the most efficient mechanism for determining rate processes or detailed spatially dependent information. Though integral tests provide a posttest measurement of entrainment fraction, there is no ready means to measure entrainment rates within the cavity or to determine the characteristic process by which debris is swept from the cavity. Integral tests also do not provide for measurement of the characteristic particle size range associated with the entrained debris in the cavity. Resolution of subissues related to HPME and DCH, such as the location and rate at which material is oxidized, depend in part on this information. Furthermore, the examination of rate parameters provides for the confirmation of behavior observed over a time interval, e.g., measurement of an entrainment rate allows the confirmation of an entrainment fraction.

In order to obtain the general information identified in the preceding discussion, the NRC has initiated a separate effects program to be conducted at Purdue University. Six specific phenomena will be studied in detail:

1. Corium jet disintegration immediately after discharge
2. Liquid film formation upon impact of the jet
3. Entrainment and drop formation by streaming gas
4. Liquid film transport from pressure and shear
5. Liquid film ejection and disintegration
6. Capture of liquid mass in compartment

Each phenomenon will first be studied experimentally using the preliminary scaling parameters. The proposed mechanism and model for each phenomenon will be systematically tested at more realistic scales than the original data base. A preliminary study indicates that a 1/10 linear scale experimental facility for the separate effects tests is adequate when water and woods metal are used as simulant material. This is also attractive since this scale is compatible with the SURTSEY integral test facility at Sandia. Flow visualization and detailed instrumentation are possible by using the simulant fluid pairs. Either of them is quite difficult or impossible in the integral facility using more prototypic high-temperature material. The experimental data are then used to verify or scale-up the proposed correlations. For this purpose, several basic existing models and correlations, as well as the ones developed by previous NRC programs, will be systematically examined. The study will be carried out by a detailed mechanistic modeling study, together with visual observations through high-speed photography and the data from the local instrumentation.

Overall, the separate effects testing and model assessment is projected to be complete within 2 years, with completion of the first phase of testing, using a water-air system, scheduled for the end of calendar year 1992.

Table 2.1.1

Test	Test Initial Conditions	Target Date
IET–1	Melt Mass: 43 kg Fe oxide/Al/Cr thermite Driving pressure: 7.1 MPa (IET–1), steam Hole size: Tube ejection with ablation, 3.5 cm **Atmosphere: inert, N_2** Cavity water: condensate level, 0.9 cm Containment water: none	9/13/91
IET–2A	Thermite temperature measurement	11/1/91
IET–3	**Atmosphere: reactive, 0.2 MPa air/N_2** The rest of the initial conditions are the same as IET–1.	12/13/91
IET–1R	Repeat of IET–1 except driving steam pressure of 6.2MPa	2/7/92
IET–4	**Containment water: condensate level** The rest of the initial conditions are the same as IET–3.	3/20/92
IET–2B	Repeat of IET–2	4/28/92
IET–5	**Atmosphere: 0.2 MPa air, CO_2, H_2** The rest of the initial conditions are the same as IET–4.	5/13/92
IET–6	**Atmosphere: reactive, 0.2 MPa air, N_2, H_2 (scaled)** The rest of the initial conditions are the same as IET–3.	6/18/92
IET–7	**Atmosphere: reactive, 0.2 MPa air, N_2, H_2 (scaled)** The rest of the initial conditions are the same as IET–4.	7/09/92
IET–8	Cavity water: 1/2 fill with water The rest of the initial conditions are the same as IET–3.	7/30/92
	Conversion to Surry geometry and subcompartment	9/92
IET–9	Melt Mass: #kg (scaled) – Fe oxide/Al/Cr thermite Driving pressure: 13 MPa steam Hole size: Tube ejection with ablation, 3.5 cm **Atmosphere: reactive, 0.16 MPa, air/steam/H_2 (scaled)** **Vessel without annular gap** Cavity water: condensate level (scaled) Containment water: condensate level (scaled)	9/24/92
IET–10	Counterpart to IET–9 in 1/6 scale CTT facility Scale parameters accordingly.	10/15/92
IET–11	**Atmosphere: reactive, 0.16 MPa, air/steam/H_2 (scaled)** **Vessel with annular gap** The rest of the initial conditions are the same as IET–8.	11/12/92
IET–12	Counterpart to IET–11 in 1/6 scale CTT facility Scale parameters accordingly.	12/10/92

The third major element of our DCH research program is the development and validation of a systems-level code, i.e., the CONTAIN code. The need for a systems-level code, and specifically CONTAIN to assess the consequences of a high-pressure melt ejection, reflects several considerations.

1. The diversity of reactor designs and the range of possible accident conditions leading to an HPME preclude the experimental confirmation of plant response for all reasonable combinations. Therefore, some modeling approach is necessary to address the potentially wide range of outcomes.

2. A highly complex detailed analytical treatment of DCH by a multidimensional finite difference or field code is not practical, given the capabilities of available models (e.g., KIVA, CONCHAS-SPRAY, HMS, COMMIX) and the effort required to apply these codes even in their present incomplete form. Furthermore, application of such codes at this stage would introduce the additional uncertainty associated with general validation of the codes themselves.

3. HPME is only one phase in the overall progression of a severe accident that proceeds to vessel failure at high pressure, but the resulting DCH has an influence on the subsequent plant response by altering containment atmosphere conditions from which later phenomena, e.g., long-term pressurization or core-concrete interactions, will influence containment performance. Thus, calculation of the integrated containment response to a severe accident over the entire course of the event dictates that a severe accident code serve as the vehicle for the assessment. CONTAIN, the NRC's severe accident containment response code, is the logical choice for modeling. As a practical consideration, past efforts to model DCH have enabled the NRC to modify the code with a reasonable level of effort.

4. Interpretation and analysis of integral tests require an analytical model capable of calculating the synergetic effects associated with DCH. Analysis of the integral tests with the tool to be used in plant analysis is also essential to most directly address the scalability of test results and models.

The principal challenge in applying the CONTAIN code to DCH is to incorporate models that are sufficiently mechanistic to predict the influence of major plant and accident parameters, yet are consistent with the overall engineering, correlational modeling used in CONTAIN. It is also necessary to allow some flexibility in modeling given present uncertainties.

In accordance with our program objectives, the NRC has initiated efforts at SNL to quantitatively assess, implement, and document CONTAIN models for RPV and cavity phenomena, relying wherever possible on past activities including the SNL preparation of a DCH models and correlations document and those tasks of the TPG in which model assessment was undertaken. Given the relative abundance of modeling approaches that have been proposed in the past, no new model development is being undertaken; the individual model selected will be made following expert peer review of these models.

For RPV phenomena, models will be incorporated that are consistent with those models recommended by the TPG and used by SNL in scaling the integral tests. For the cavity processes such as entrainment rate, a variety of models will be included that reflect the variety of proposed modeling approaches (Whalley-Hewitt, Ricou-Spalding, Kataoka-Ishi correlations). Incorporation of more than one model for a process allows for a comparative assessment of model adequacy without a significant penalty since most correlations rely on similar variables calculated internally in CONTAIN. The implementation and initial assessment of CONTAIN DCH models is scheduled to be completed by the end of CY 1992.

2.1.3 Anticipated Results

Our planned research on direct containment heating is directed to produce generalized modeling, incorporated into the CONTAIN code, capable of predicting the consequences of a high-pressure melt ejection for the range of applicable accident conditions. The modeling approach will have been validated against available integral and separate effects tests results. Scaling issues will have been addressed by comparison and analysis of test results for experiments conducted under this program at three different scales, as well as experimental data generated from other research. Selected analysis of full-scale reactor plant analysis will be performed to gauge the expected results for some subset of plant and accident conditions. Plant analysis will be performed in part during the program as an iterative process in order to assess the demands for accuracy in modeling, i.e., to determine how good or accurate an answer is needed and the effect on plant response to variations in modeling.

Another expected outgrowth of this research will be the development of generalized assessment criteria that may be used in lieu of more detailed analysis for a more specific set of accident and plant parameters. Such assessment criteria can serve as the basis for evaluating the adequacy of specific calculations based on other models and can serve as the basis for the development and assessment of simpler models adopted for use in full plant severe accident codes such as MELCOR and MAAP.

While our planned research is anticipated to provide for resolution of the issues that have long been associated with DCH, it is important to note that our research program will not fully resolve questions surrounding the phenomena related to corium-water interactions that may result from the ejection of debris at high pressure into a deep water pool that may be in the cavity in some designs in certain severe accident sequences. While our testing program includes a very limited investigation of the effects of cavity water inventory for the accident conditions simulated, a complete resolution of this general issue, to the extent it bears on plant performance, relies on our research program on fuel-coolant interactions.

2.2 Core Melt Progression Research

In-vessel core melt progression describes the state of an LWR reactor core from core uncovery up to reactor vessel melt-through in unrecovered accidents, or through temperature stabilization in accidents recovered by core reflooding. Melt progression provides the initial conditions for assessing the loads that may threaten the integrity of the reactor containment. The significant results of melt progression are the melt mass, rate of release, composition, and the temperature (superheat) of the melt released from the core and later from the reactor vessel at vessel failure. Melt progression provides the in-vessel hydrogen generation and the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system. Melt progression also provides the core conditions for assessing accident management strategies.

2.2.1 Research Needs

Understanding core melt progression phenomena, including an assessment of uncertainties, is important in determining the margin of conservatism for those severe accident issues that have been resolved (e.g., BWR Mark I liner failure) and in resolving the remaining severe accident issues (e.g., DCH). The objective of the research on melt progression is to provide quantitative tools for assessing the mass and other characteristics of the melt released from the core and from the reactor vessel, as well as hydrogen generation and the conditions for fission product release, transport, and retention. This research is focused on four general issues or technical areas:

1. Determining whether a metallic blockage of the core (or core plate), similar to that which occurred at TMI–2, would occur in BWR accidents, or alternatively, whether the metallic melt drains from the core (and core plate) when formed. The core blockage or melt drainage branch point and the different melt progression sequences for the two different pathways are shown in Figure 2.2.1.

2. The conditions for ceramic pool melt-through from a blocked core that determine the mass and other characteristics of the melt released from the core into the lower plenum.

3. The mode of vessel failure that determines the rate of melt release into the containment and the timing of the release.

4. The application of experimental results in models of the governing phenomena for severe accident safety assessments. These models are also to be used, usually in simplified form, in the severe accident systems codes SCDAP/RELAP5 and MELCOR.

If during the course of this melt progression research the need arises for special research results outside the above areas of focus, for example, specific material properties, then such research will be undertaken after a peer review of the need. A strong effort has been made in planning this research program to focus on the key areas of uncertainty in issue resolution, and not to try and cover a large list of unprioritized technical uncertainties.

The research needs in each of these areas are discussed in the remainder of this section.

Whether a metallic blockage develops across the core during a severe accident, as occurred at TMI–2, or the metallic melt (and later the ceramic melt) drain promptly to the lower plenum from the core (and BWR core plate) has a major impact upon subsequent melt progression, as shown in Figure 2.2.1. The mass, rate of release, and other characteristics of the ceramic melt released from the reactor core into the vessel lower plenum impact the overall consequences of the accident. Analysis indicates that in BWR accident sequences with automatic depressurization, the blowdown lowers the water level below the core (and core plate) so that core heatup occurs under dry core (negligible steam flow) conditions. All the available experimental data on melt progression are for wet core conditions (i.e., water in the bottom of the core). No data currently exist for the dry core conditions that would result from the automatic depressurization that is called for by the emergency operating procedures for U.S. BWRs. Analysis indicates that core heat up, particularly the rapid oxidation transient, and metallic melting, relocation, and blockage formation are significantly different for dry core and for wet core conditions.

The current experimental and analytical research program to determine whether core blockage occurs or does not occur under BWR dry core accident conditions is discussed in Section 2.2.2. Analytical modeling of the process of melt relocation and blockage formation is needed to provide a tool for extending the experimental results to the full range of risk-significant severe accident conditions for both BWRs and PWRs.

The second research area concerns the characteristics of the ceramic melt that drains from the core (and BWR

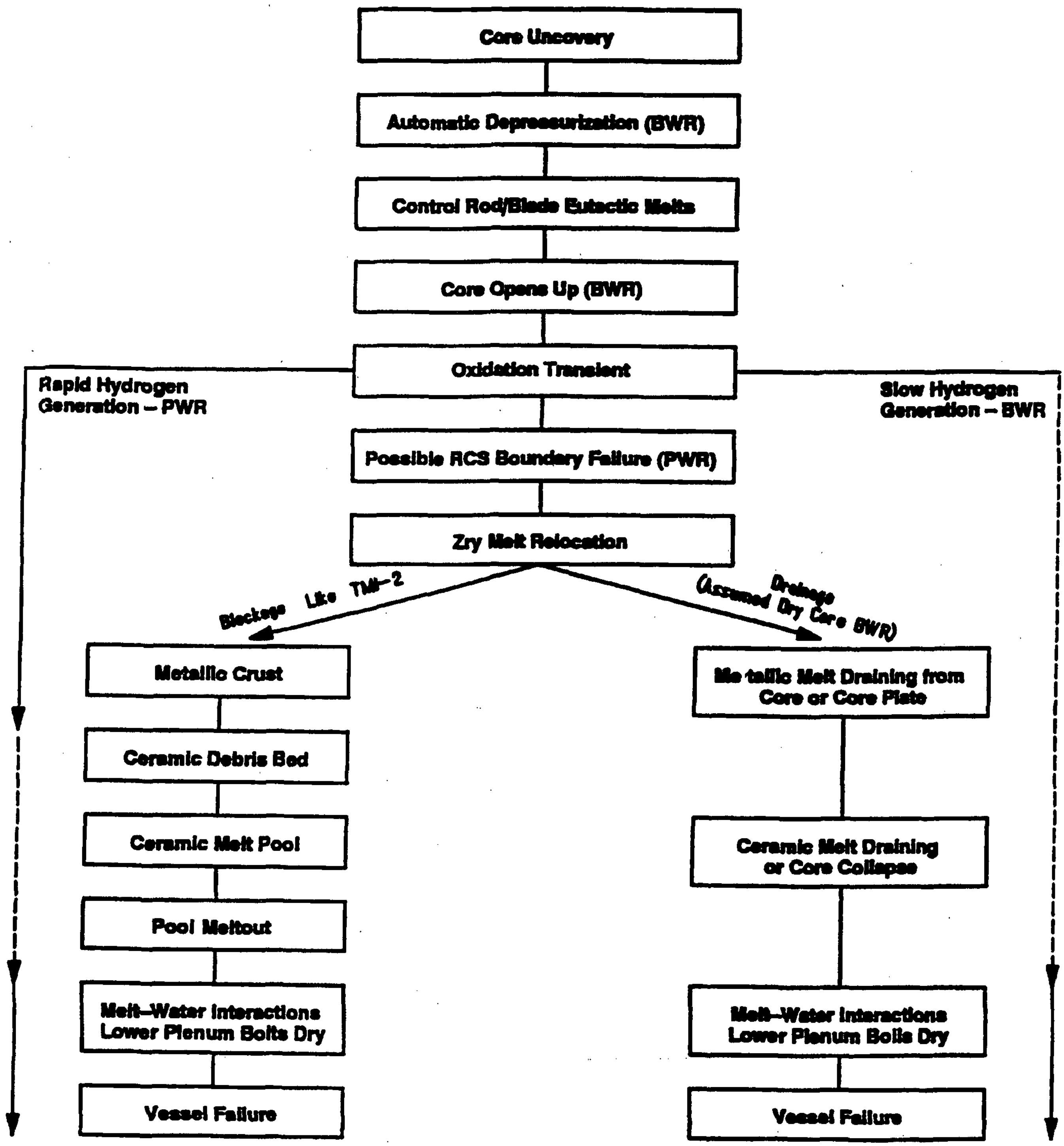


Figure 2.2.1 Core Melt Progression Sequence Showing Blockage or Drainage Paths.

core plate) into the lower plenum in a blocked core accident. The significant melt characteristics (mass, rate of release, temperature (superheat), and composition (metal content) are largely determined by the threshold and location of melt-through of the metallic core blockage (and secondary ceramic crust) by the growing ceramic melt pool. These characteristics are very important in assessing the impact of core-melt accident sequences on containment loads. Most of the information currently available on the late (ceramic melt) phase of melt progression, including blockage melt-through, has come from the TMI-2 core examination. This information, however, is not sufficient to validate modeling of the melt-through threshold and location and the characteristics of the released ceramic melt.

Ceramic pool melt-through from a blocked core is a whole-core phenomena and is not amenable to direct experimentation. Therefore, experimental research needs to be conducted on separable key parts of the problem with integration of the results by analysis and codes. For example, the surface heat flux distribution of the internally heated ceramic melt pool that results from natural circulation in the pool is an important phenomenon in determining the meltthrough threshold and location in blocked-core accidents. An assessment will be undertaken to determine whether or not correlations derived from the existing data at lower Rayleigh numbers and in different geometry than expected in a core melt accident are adequate for characterizing the surface heat flux distribution of the ceramic melt pool. The research program to determine the threshold and location of melt-through is discussed in Section 2.2.2.

The third research area concerns the mode and timing of vessel failure in core melt accidents. A primary question is whether, for a given reactor type and accident scenario, vessel failure is by penetration failure or by global or local creep rupture failure. These different failure modes give very different results on the rate of melt ejection from the vessel and also on the failure threshold. The research on lower head failure models reported in NUREG/CR-5642, "Light Water Reactor Lower Head Failure Analyses," is discussed in Section 2.2.2.

The fourth area concerns severe accident systems codes, such as SCDAP/RELAP5 and MELCOR. These codes model the key physical processes that occur during invessel melt progression, often in simplified form, in order to determine the global response of the reactor during a core melt accident. Detailed phenomenological models are also used to perform analyses of the experiments in order to gain a firm understanding of the important physical processes. This understanding forms the foundation for the simplified models in the severe accident codes. The detailed phenomenological models and the thorough analyses of the experiments form the basis for assessing the adequacy of the simplified models in the systems codes. Section 2.4 presents the NRC research to develop and validate severe accident codes.

While it is recognized that appropriate scaling analyses are important to ensure that the experiments are performed under the proper conditions and in the proper parameter range for application to full-scale reactor accidents, it may be that a specific issue is not completely amenable to scaling analysis. In these cases, experiments will not be used to develop empirical correlations, but rather to identify important dominant physical/chemical processes and assure that they are properly represented in the analytical models. Analytical models of the key phenomena are also necessary for the results to be applicable over the range of severe accident conditions.

The primary current development activities in the analysis of melt progression processes involve the DEBRIS porous media late-phase melt progression model and the MERIS model of metallic melt relocation and blockage formation that is an extension of the porous media framework of DEBRIS. Rather than trying to describe the detailed geometry changes during melting, melt relocation, and the freezing and remelting of crusts, DEBRIS and MERIS use a much simpler porous media framework that keeps track only of material relocation and state (liquid, solid, composition, temperature). DEBRIS provides a 2D(r,z) mechanistic treatment of the key phenomena involved in late-phase (ceramic melt) melt progression in blocked core accidents (Refs. 1,2). Specifically, DEBRIS treats the melting dynamics of a particulate ceramic debris bed supported by a metallic crust across the fuel rod stubs in the lower region of the core. The modeling includes ceramic crust formation around the melt pool and pool melt-through of the supporting crust system and melt drainage from the core into the lower plenum. MERIS uses a 3D(x,y,z) formulation to treat the complex BWR core geometry and a time-varying anisotropic effective permeability to account for melt relocation and freezing (Ref. 3). DEBRIS and MERIS respectively, have shown good agreement with the very limited experimental data currently available, on ceramic and on metallic melt relocation and crust formation.

An assessment is needed of the ability of the DEBRIS porous-media model to predict the late-phase melt progression behavior in the TMI-2 accident. An assessment is also needed of DEBRIS with the experimental data from the melt progression (MP) experiments in ACRR on the relocation and failure dynamics of the ceramic and metallic crust system that supports the growing ceramic melt pool. Also needed is an assessment of the MERIS porous media type model of metallic melt relocation and blockage formation with the experimental results of the planned ex-reactor experiments on this subject and with any other relevant available information. If the assessments of DEBRIS and MERIS, with possible improvements, are successful, their modeling, probably in simplified form, can be incorporated into SCDAP/RELAP5 and MELCOR.

2.2.2 Current Research Program

As described in Appendix A.2, a great deal of information has been obtained on the processes involved in the early phase of melt progression that extends through core degradation and metallic material melting and relocation. Except for the BWR experiment in the Canadian NRU reactor, which has been delayed because of a leak in the heavy water system of NRU, current NRC research on melt progression is concentrated on two major uncertainties or issues. The first issue is the BWR accident conditions, if any, in which a metallic core blockage, similar to that in the TMI-2 accident, would not be formed. This is the major branch point in the melt progression sequence shown in Figure 2.2.1. The second issue concerns the conditions for the meltthrough of the growing pool of ceramic melt that is supported by the metallic blockage. Research on the mode of vessel failure in core melt accidents, which in general is plant-and accident-sequence-specific, is limited to the validation of the models developed and presented in NUREG/CR-5642. Issues related to severe accident core melt codes are discussed in Section 2.4.

2.2.2.1 Core Blockage or Melt Drainage in BWR Accidents

An ex-reactor experimental program has been started to resolve the question of metallic melt blockage or drainage under BWR dry core accident conditions. Unlike the previous ACRR and CORA BWR experiments, the ex-reactor experiments do not start from the initial intact rod geometry. Instead, the ex-reactor experiments accurately reproduce the core conditions at the onset of metallic melt relocation, and the system behavior under these well defined conditions is then measured.

Since internal heat generation in the metallic melt is negligible in reactor accidents, these experiments can be performed ex-reactor without internal heat generation with pours of metallic melts of prototypic composition and flow rates (dribbles). The test assemblies have prototypic dimensions, materials, structures, heat capacities, and temperature distributions. Because of the axial temperature distribution, illustrated in Figure 2.2.2, blockage formation is only possible in the lower quarter of the core and in the core plate region. The metallic melt comes from the upper three-fourths of the core and the metallic pour represents that part of the core in these experiments. The ex-reactor experimental apparatus represents the lower quarter of also the core and also the core-plate region. The pour of metallic melt represents the incoming metallic melt from the upper three-quarters of the core to give overall full-length simulation. The mass, flow rate (dribble), and composition of the metallic melt poured into the experiment test sections are obtained by analysis of the flow of metallic melt from the upper three-quarters of a full-scale BWR core under dry core accident conditions. This analysis is consistent with the phenomenology observed in the ACRR DF-4 and CORA BWR tests. These tests were performed, however, for BWR wet core conditions. For most of the tests, there is pre-oxidation of the Zircaloy (and stainless steel) surfaces by furnace-heating in steam to provide oxide film thicknesses that are representative of operating BWRs. The initial temperature distribution in the core and core-plate structure is produced by preheating the system with downward flowing argon. There is compensation for radial heat losses with external heaters on the insulation of the test assembly. A detailed description of these experiments and their objectives is given in Reference 4.

A cross-section of the test fuel bundle is shown in Figure 2.2.3. This has full-scale radial simulation of half a channel box unit cell in a BWR core. This simulation includes the control blade in the gap between channel boxes, the roughly equal horizontal length of unbladed gap that is potentially important for melt drainage, and an additional two rows of fuel rods outside the gap. These additional rods provide a prototypic temperature distribution in the channel box walls, in the gap, and in the control blade, and they also provide a volume for diversion of the melt flow following channel box failure.

This simulation has 64 fuel rods, channel box walls, and bladed and unbladed sections of the gap between the channel box walls. In initial operation of the experiment and for debugging, it is planned to perform two XR1 experiments in the simplified geometry of the blade, gap, and channel box walls shown in the upper schematic drawing of Figure 2.2.3. These experiments will also provide basic information on melt relocation flows before channel box failure and on the effects of pre-oxidation on the interaction with and failure of the channel box walls by the control blade melt.

The XR2 experimental test assemblies will include a full radial-scale mockup of a corresponding half unit cell in the core plate region below the core that includes the prototypic complex flow paths, materials, and heat capacities, as well as simulation of the interior section of the BWR channel boxes with fuel rods and of the core plate structure below the core. The complexities of the structures and flow paths in the core plate region are shown in Figure 2.2.4. The core plate itself occupies only 34% of the horizontal cross-section in the core plate region, and most of this cross-section is relatively open with complex flow paths defined by thin, low-heat capacity structures of low-melting stainless steel. Because of these geometrical complexities, rivulet flow complexities, and the effects of eutectic interactions, analysis of the potential for blockage or drainage of eutectic metallic melts of control

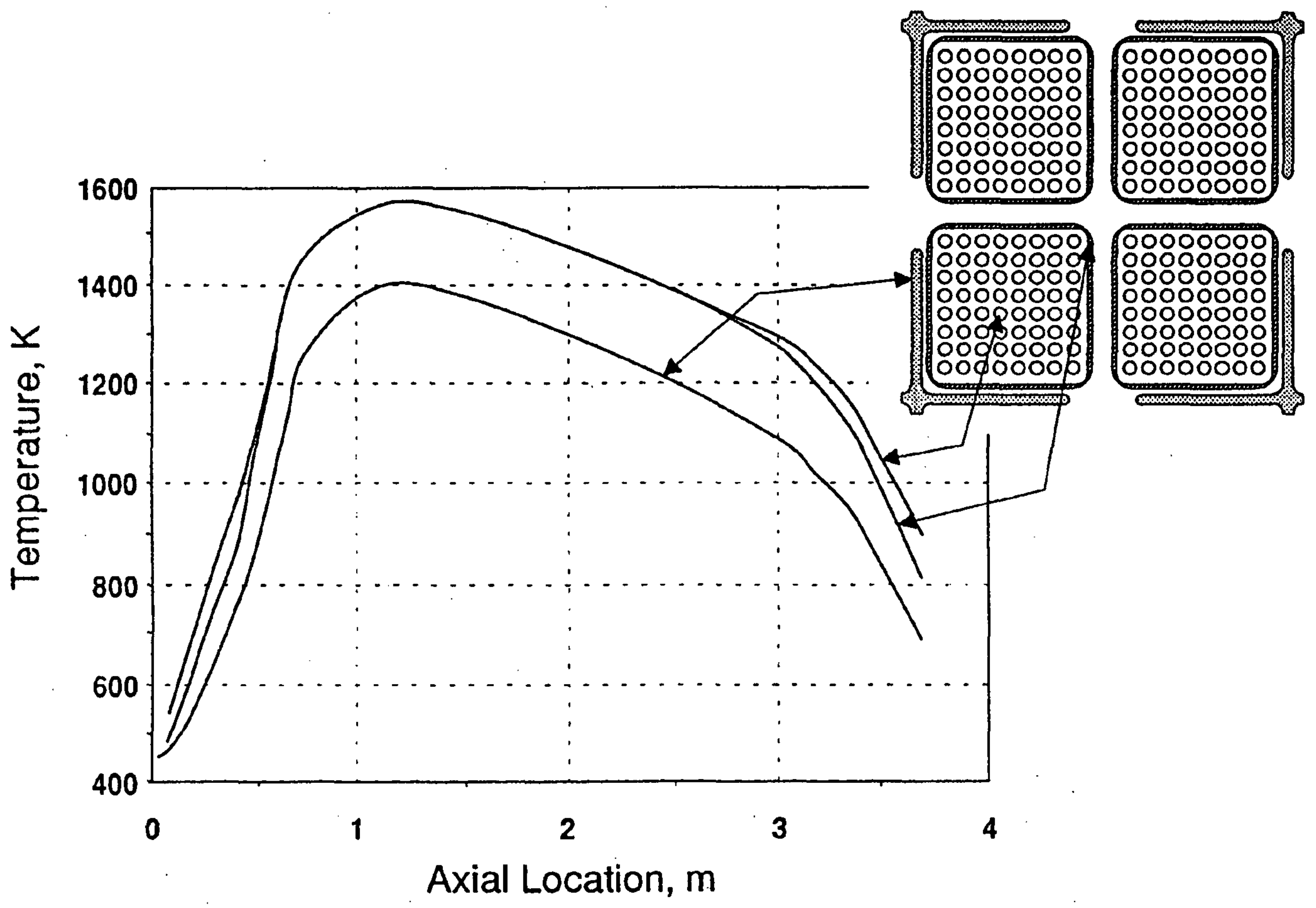
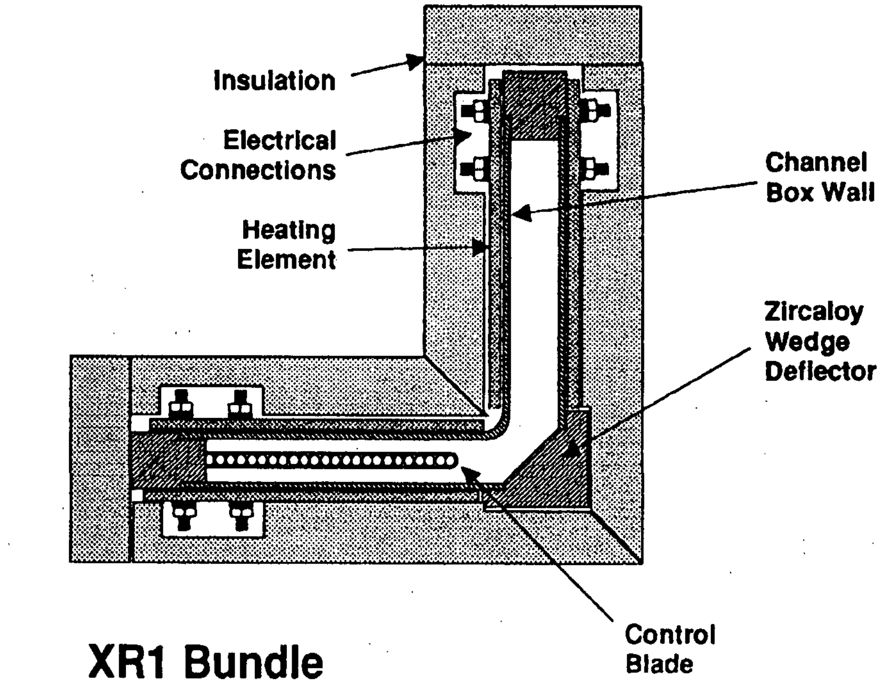


Figure 2.2.2 Axial Temperature Profiles for the Control Blade, Channel Box and Fuel Rods, Indicating Maximum Lateral Temperature Variations



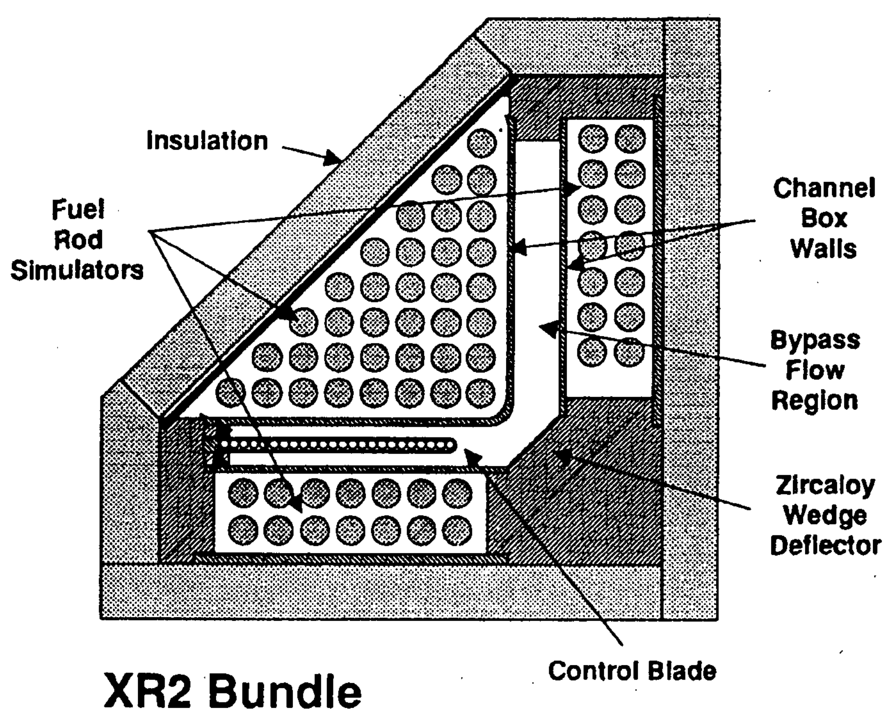


Figure 2.2.3 XR1 and XR2 Test Bundle Cross Sections.

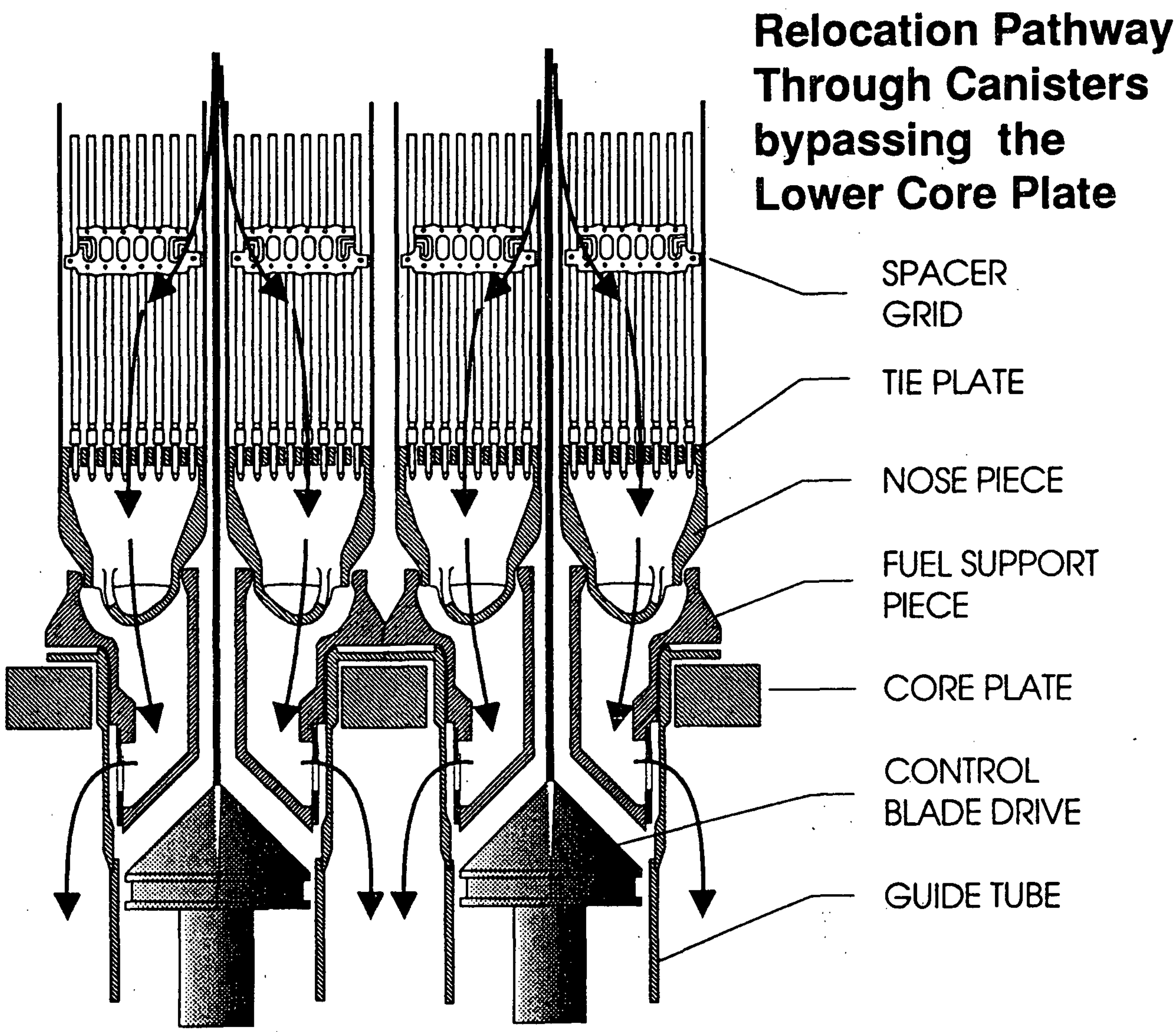


Figure 2.2.4 Melt Drainage Pathways Through the Fuel Canisters Which Bypass the Lower Core Plate.

blade materials and Zircaloy in the core plate region needs guidance from experimental data. It is planned to perform the XR2 experiments in a steam environment.

Analysis with MELCOR has indicated that two separate relocations of metallic melt occur under BWR dry core conditions, the first a melt of B_4C and stainless steel control blade materials with a melting (eutectic) point of about 1500°K, and the second a melt of cladding and channel box Zircaloy with a melting point of about 2200°K. The two simplified gap-geometry XR1 experiments will use a single pour (dribble) of control blade melt. The XR2 experiments, which include the fuel rod region within the channel boxes and the core plate region below the core, will use two separate pours, one with control blade material melt with some dissolved Zircaloy and the other with a higher temperature Zircaloy melt. The melt mass, composition, temperature, pour rate, and the time of separation between the two pours in the XR2 experiments will be determined by analysis and by the results of the dry core test soon to be performed in the German CORA ex-reactor fuel damage facility.

The effect of the pre-accident oxide film on the core Zircaloy (and stainless steel) from normal reactor operation in delaying material interactions and eutectic formation will be investigated in the XR1 experiments. These results will be factored into the XR2 experiments. Additional data on this effect will also be also obtained from CORA experiments. Current plans are for nearly all the XR2 experiments to be pre-oxidized.

The test matrix for the ex-reactor experiments has been formulated to determine, with as few tests as possible, whether there are any conditions within the range of the governing parameters (including uncertainties) for BWR dry core accident conditions under which metallic melt drainage, rather than core or core plate blockage, can occur. The governing parameters are the metallic melt temperatures, the core axial temperature gradient, and the presence of an initial oxide film on the Zircaloy structure corresponding to that in normal BWR operations. After the two initial simplified XR1 experiments to check out the experimental techniques and to determine the effect of pre-existing oxide films on the channel box Zircaloy, the first of the four planned XR2 experiments will be performed with all the above parameters at the end of their range that favors melt drainage. If core blockage does occur under these conditions, it will then have been determined that the blocked core sequence is to be expected in all dry core BWR accidents. The experimental matrix would then be shortened to focus on the characteristics of the metallic blockage formed under the best estimate conditions. This is needed for assessing the durability of the blockage with continued core heat up and also for assessing blockage melt-through by the growing ceramic melt pool in late-phase melt progression. If melt drainage does occur in the initial XR2 experiment, the preliminary test matrix given in Table 2.2.1 will be followed step by step, increasing the potential for blockage to see if blockage is achieved under these dry core conditions.

The conditions for this test matrix will be reviewed as experimental results are obtained, particularly those from experiment XR2-1. Currently it appears that these experiments and corollary analyses with additional input from the CORA BWR tests and from the BWR FLHT-6 test in the NRU reactor will provide sufficient data to resolve the question of metallic melt blockage or drainage for the relevant range of BWR dry core accident conditions. The last of a series of full-length tests on fuel damage and melt progression during coolant boildown (wet core) is scheduled to be performed in the Canadian NRU reactor in late 1992. The results will provide unique length-effect data for BWR core geometries. Two high-burnup fuel rods in the 14-rod test assembly that is shown in cross-section in Figure 2.2.5 will provide unique data on fission product release in the boron-containing BWR core environment. Post-irradiation examination (PIE) of the test fuel bundle should also provide some information on the effects of high-burnup fuel on early-phase melt progression.

Table 2.2.1 Test Matrix for the Ex-Reactor Experiments on Metallic Melt Relocation and Blockage Formation Under BWR Dry Core Accident Conditions

Test*	Melts*	Grad T**	Melt T	Zry Oxide	Objectives
XR1–1	C.B.	Prototypic	Prototypic	Yes	Oxide Film
XR1–2	C.B.	Prototypic	Prototypic	No	No Oxide Film
XR2–1	C.B., Zry	High	High	Yes	Favor Drainage
XR2–2	C.B., Zry	High	Low	Yes	Decrease Melt T
XR2–3	C.B., Zry	Low	Low	Yes	Decrease Grad T
XR3–4	C.B., Zry	Low	Low	No	Delete Zry Oxide

*C.B. means control blade materials plus some dissolved Zircaloy.
**High grad T means that the ΔT occurs over a short distance, while low grad T means that the same ΔT occurs over a long distance.

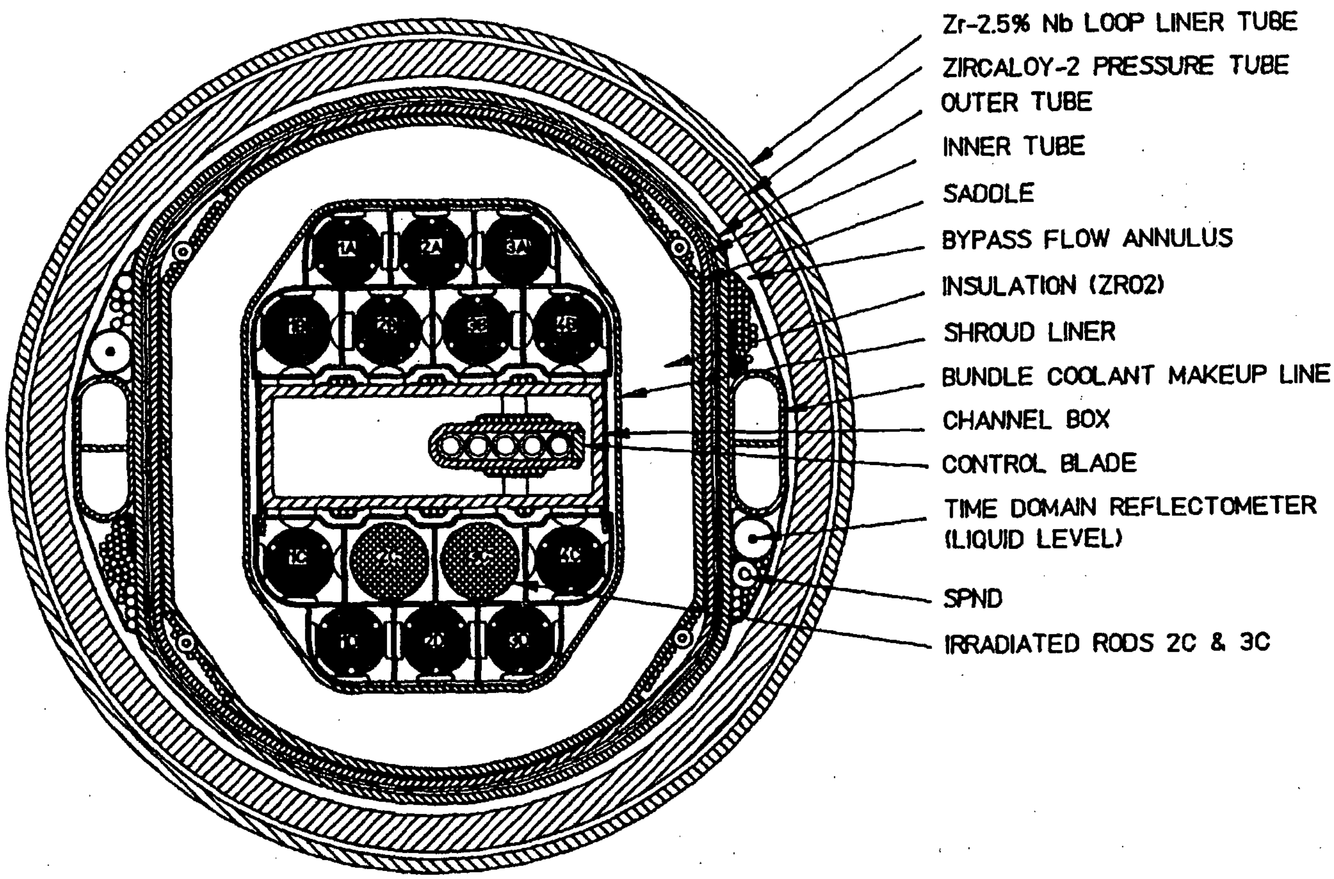


Figure 2.2.5 Test Bundle Cross-Section for the NRU BWR Test

Experiments in the German CORA ex-reactor fuel damage test facility with electrically fuel rods have provided much of the current information base on core degradation and early (metallic melt) phase melt progression. This information includes materials interaction (eutectic) effects and the effects of reflooding damaged cores. The CORA results are available to NRC as part of the international Cooperative Severe Accident Research Program (CSARP) and will be used to further the data base for code validation and assessment. The forthcoming CORA dry core BWR test will provide data on the initial conditions of the incoming melt for the ex-reactor experiments on metallic melt relocation and blockage formation and will complement the results of these experiments.

2.2.2.2 Ceramic Pool Melt-through from a Blocked Core

In blocked core accidents, the primary determinants of the mass and other characteristics of the mostly ceramic melt released into the lower plenum are the threshold and location of meltthrough of the metallic and ceramic crust system that supports the growing, mostly ceramic melt pool. The failure threshold and location are themselves primarily determined by the relocation and failure dynamics of this complex fuel-rod-supported system of ceramic and metallic crusts. Uncertainties as to the surface heat flux distribution of the internally heated melt pool because of natural circulation are also important, particularly for the failure location. An assessment will be made and peer reviewed to determine whether or not the correlations derived from existing data at much lower than prototypic Rayleigh numbers and in different geometry than in those occurring in blocked core accidents are adequate to characterize the surface heat flux distribution of the ceramic melt pool. The TMI-2 melt pool had a Rayleigh number of about 10^{16}, turbulent flow, and hemispheric geometry. Nearly all of the available data are for a Rayleigh number of 10^{11} or less, laminar or unsteady laminar flow, and thin, semi-circular geometry. In addition, there are a few data in thin, rectangular geometry at Rayleigh numbers up to 3 x 10^{13} that agree reasonably well with the low Rayleigh number data. If data at a Rayleigh number of 10^{16} should prove to be needed, an experiment with water as a simulant fluid would be feasible. The time constant for establishing natural circulation flow and heat transfer in the melt pool also needs to be examined. In addition, analysis should be performed to assess the possibility that the metal in the mostly ceramic melt pool can significantly affect the natural circulation in the pool and its surface heat flux distribution.

Very little experimental information currently exists on the melting dynamics of the ceramic melt pool and on the relocation and failure dynamics of the fuel-rod-supported system of metallic and ceramic pool-supporting crusts in the late-phase blocked core. Nearly all the information currently available on these processes has come from the TMI-2 core examination. In the TMI-2 accident, pool meltthrough occurred at the side of the core and reflected the core reflooding in that accident. It is currently not known whether in unrecovered accidents, meltthough would occur at the side or at the bottom of the melt pool. The failure location makes about a factor of two difference in the released melt mass.

The melt progression experiments in the ACRR test reactor have been designed to furnish phenomenological information on the governing mechanisms involved in the relocation and failure dynamics of the fuel-rod-supported metallic and ceramic crust system that supports the growing internally heated mostly ceramic melt pool. The melt attack on the metallic crust involves both thermal attack and possibly chemical (eutectic) interactions among the crust materials and the Zircaloy cladding and UO2 in the fuel rod stubs. The results of these experiments are to be applied at full scale to blocked-core reactor accidents through a mechanistic model or models of the meltthrough process. The DEBRIS porous media model, which treats the governing processes involved mechanistically, will be the primary tool used in analysis and interpretation of the experimental results (Refs 1, 2). The simplified late-phase melt progression modeling in SCDAP/RELAP5 and MELCOR will also be checked against these results.

A schematic drawing of the test assembly for the melt progression experiments, actually that for MP-2, is shown in Figure 2.2.6. Detailed descriptions of these experiments are given in experiment plans for the melt progression experiments and for the MP-2 experiments specifically (Refs. 5, 6). In these experiments, a pre-cast metallic crust with a prototypic low UO_2 content (for the dissolved UO_2 in the metallic crust) bridges a 32-rod array of clad fresh fuel rod stubs and supports a particulate ceramic (UO_2 ,ZrO_2) debris bed.

The purpose of the MP-1 experiment was to provide base-line information on ceramic melt dynamics in the blocked core system and on the failure mechanics of the supporting metallic crust. MP-1 was conducted in October 1989, and the experiment was terminated earlier than planned because a temperature safety limit had been reached internally in the experiment package. The MP-1 experiment provided basic information on heat transfer, densification, and melting in the particulate ceramic debris bed and on the formation and downward propagation of the secondary ceramic crust or blockage below the growing ceramic melt pool. Peak temperatures of about 3200°K were reached in the ceramic melt pool, which gives about 400°K superheat above the UO_2-ZrO_2 eutectic temperature.

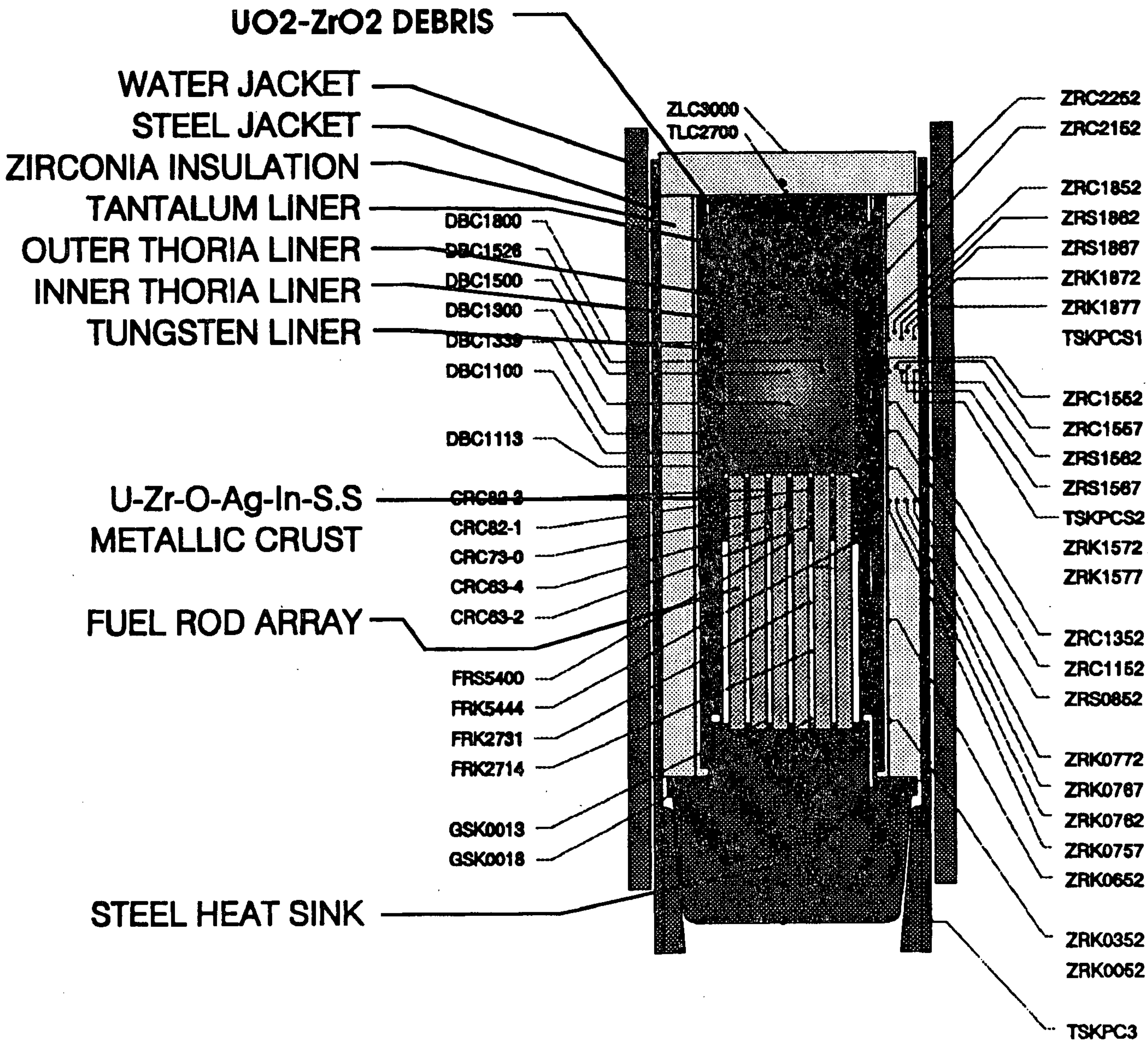


Figure 2.2.6 MP–2 Test Section Showing Major Components and Thermocouple Locations.

MP–2 is to have a metallic crust of TMI–2 composition, which includes PWR control rod materials (Fe, Ni, Cr, Ag, In, Cd). A purpose of the MP–2 experiment is to determine the effects of the eutectics of these control-rod-materials on the crust failure mechanisms and thresholds and to investigate the effects of the fuel rod stubs on the behavior of the secondary ceramic crust. It is not currently known whether eutectic (chemical) interaction of the control-material-containing metallic crust with the fuel rod stub Zircaloy cladding itself is significant. It is planned to run MP–2 to metallic crust failure. The ceramic insulating shrouds and metallic shields in MP–2 have been modified to prevent the premature termination of the experiment that occurred in MP–1 and to allow temperatures in the melt pool well above UO_2 melting (3100°K) as well as the UO_2–ZrO_2 eutectic (2800°K) without reaching the safety limit of the internal tantalum shield that required the shutdown of MP–1.

With the results and interpretation of experiment MP–2 and with the results of melt progression sensitivity studies, an expert peer review group on melt progression will review the need for and the nature of further research in the area of late-phase melt progression. In particular, a determination will be made of the need for further melt progression experiments.

2.2.2.3 Mode of Vessel Failure

The mode and timing of the reactor vessel lower head failure have controlling effects on the subsequent containment loading events in severe accidents. A program that reviews and extends the numerous studies of reactor failure modes made during the past decade is nearing completion. The major potential failure mechanisms include penetration tube heatup and failure, tube ejection, lower head global creep rupture, and localized thermal and mechanical loads on the lower head. This last includes nonuniform distribution of a the debris bed and coherent jets of molten core material that can directly contact and ablate the vessel wall.

Because there are a large number of debris conditions, lower head designs, and accident scenarios, analytical techniques using key dimensionless parameters were used to develop failure maps that indicate the relative potential for failure of the various modes as a function of the dimensionless parameters. Limited numerical finite element analyses were then utilized to benchmark the failure maps.

An analytical method for bulging analysis (local creep rupture) was also developed. The analytical solution is based on an axisymmetric plastic analysis with a pseudo-plane strain treatment of deformation normal to a meridional plane. This and other failure analysis methods require high-temperature creep data. Therefore, a series of tests to acquire high-temperature creep rupture data was carried out for pressure vessel steel. Additional creep rupture data acquisition is planned for the penetration materials (Inconel, stainless steel, and SA105 or SA106 steel). The results of this analysis have been reported in draft NUREG/CR–5642, "Light Water Reactor Lower Head Failure Analysis" (Ref. 7).

The analytical models in this report are undergoing a peer review to ensure that the report is on a firm technical basis. Model validation will be undertaken when results become available from the Swiss CORVIS program of integral tests on the melt pool thermal attack on the vessel lower head and the head penetrations, which will be made available to NRC through the CSARP program, and from the cooperative research program between the USNRC and the I.V. Kurchatov Institute in Russia. An extensive series of experiments and associated analysis has been started at the I.V. Kurchatov Institute on the integrity of the vessel lower head under melt pool attack. A major purpose of this research is to investigate the efficacy of ex-vessel water cooling in preventing vessel meltthrough. The research plans include large scale integral tests with up to 200kg melts of UO_2 or molten salt, as well as smaller scale separate effects experiments and analysis on melt pool thermal hydraulics.

2.2.3 Anticipated Results

The first major area of investigation in the current melt progression research program is the determination of whether blockage of the core by metallic melt or drainage from the core and core plate occurs in BWR dry core accidents. It is anticipated that the currently planned ex-reactor experiments on metallic melt relocation and blockage formation under BWR dry core conditions along with modeling will provide sufficient information to resolve this question. The full-length BWR test in NRU and results from recent BWR tests in CORA should contribute increased confidence to the conclusion. The results of these experiments will be used to validate models in the SCDAP/RELAP5 and MELCOR systems codes for use in predictions of the severe accident behavior of nuclear power plants.

The situation regarding the second major area of investigation, ceramic pool melt-through from a blocked core, however, is far more complex. The results and interpretation of the ACRR MP–2 experiment should substantially increase our understanding of the key processes involved in ceramic melt pool growth and in the dynamics of the relocation and failure of the pool-supporting metallic and ceramic crust system. With the results and interpretation of MP–2 in hand to augment the results of the TMI–2 core examination, sufficient information should be available for the expert review group on melt progression to assess the need for and nature of future work for late-phase melt progression issue resolution. Such future research may or may not include further melt progression experiments. The need for an experiment on the thermal-hydraulics of melt pools with internal heat generation may be established, although this currently appears unlikely.

The report on lower head failure analysis will provide failure maps in terms of dimensionless parameters that may be used to predict failure modes. In many cases, the most probable failure mode may be predictable without further analysis. If further analysis is required, only a limited effort with minimal uncertainty ranges should be necessary.

Section 2.2 References

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2.3 Fuel-Coolant Interactions and Debris Coolability

2.3.1 Research Needs

Since the WASH 1400* quantification of a steam-explosion-induced missile as a possible mode of containment failure (alpha mode), significant progress has been made as reflected in current risk studies, e.g., NUREG-1150 where alpha-mode failure does not seem to be a dominant contributor to early containment failure. However, the progress in understanding this area has been mainly directed at the conditions for in-vessel molten fuel pouring into a coolant pool and its likelihood of causing containment failure by energetic interactions. The shift in emphasis to accident management for a variety of reactor geometries and meltdown scenarios, coupled with the wide uncertainties in the NUREG-1150 expert elicitations of fuel-coolant interactions (FCI), suggest that confirmatory research is needed to learn more about the fundamental mechanisms of FCI to be able to determine the conditions under which FCI is important to severe accident risk and accident management. The emphasis of this research is now shifted to providing the appropriate methodological and analytical tools for evaluating major aspects of the accident sequences, including quantification of steam and hydrogen production, the mode and timing of vessel (or reactor coolant system) failure, and ex-vessel events of potential significance to debris coolability and containment loading. Although all these issues are discussed elsewhere in this research plan, there are fundamental aspects of FCIs that are germane to all these issues.

Three specific issues that require additional information either from experimentation or from analysis are:

1. FCI energetics
2. Fuel melt quenching in water pool
3. Water added to a degraded core (in-vessel as well as ex-vessel)

The FCI characteristics are briefly described below, and the experimental/ analytical data base needed for more comprehensive resolution of these issues is discussed for developing the overall research to address the above three areas.

2.3.1.1 Research Needs—Fuel-Coolant Interaction (Energetics)

The fundamental aspects of FCI are the evolution of liquid interfacial (fuel-coolant) area and associated heat transfer during the contact. When the two liquids first

*U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants," NUREG/-75/014 (WASH 1400), December 1975.

come into contact, the coolant begins to vaporize at the fuel-coolant liquid interface as a vapor film separates the two liquids. The system remains in this state for a delay period ranging from a few milliseconds up to a few seconds. During this time, the fuel and coolant liquid intermix (sometimes called premixture) because of density and velocity differences as well as vapor production.

The vapor film destabilization occurs next, triggering fuel fragmentation. This rapidly increases the fuel surface area, vaporizing more coolant and increasing the local vapor pressure. The vapor formation propagates spatially throughout the fuel-coolant mixture, causing the macroscopic region to become pressurized. Subsequently, the high-pressure coolant vapor expands against the inertial constraint of the surroundings and the mixture itself. The vapor explosion process is now complete, transforming the fuel's internal energy into the kinetic energy of the mixture and its surroundings. Experiments to validate mixing calculations and fragmentation are needed to address in-vessel and, to a large extent, ex-vessel fuel-coolant interactions.

2.3.1.2 Research Needs—Fuel-Melt Quenching

The TMI-2 accident indicated that under certain conditions the fuel melt may be quenched at the time of pouring into a water pool in the RPV lower plenum. Previously it had been assumed that a fuel pour into the RPV lower plenum would result in either settling of the unquenched fuel (and eventual RPV wall failure) or a vapor explosion. Although there have been many integral FCI experiments, there is no data available under these particular conditions. Because of this lack of data, the FARO-LWR experiments are planned. They involve a prototypic fuel mass (50–150 kg of $UO_2/ZrO_2/Zr$ at 3000°K) poured into saturated water at high pressures (5–50 bar for 1–3 meters depth with a chamber diameter of 0.5 to 0.7 m).

2.3.1.3 Research Needs—Adding Water to a Degraded Core

Severe accident management is a natural outgrowth of past emphasis on risk assessment. In fact, there are a number of particular issues that must be addressed when accident management is the main objective. Timing is important to accident management, and only recently have PRA studies considered it in some systematic fashion; e.g., NUREG-1150 considered the effect of the operation of engineered safety features (containment fan coolers or sprays) before core heatup, before vessel failure, or after failure. Inclusion of this "timing" behavior can indicate where opportunity exists for operator intervention to help reach a stable coolable state. Since water is the primary accident management tool and the FCI can alter the course of the accident, it is important to investigate the benefits of adding water to the degraded core with consideration of the possible adverse side effects. This is particularly true for the FCI because many of the fundamental mechanisms are not well understood.

2.3.2 Current Research Program

This section describes the current research program for each of the three specific issues discussed earlier—FCI energetics, fuel-melt quenching in a water pool, and adding water to a degraded core.

2.3.2.1 Research Program to Address FCI Energetics

The objective of this research is to determine under what conditions vapor explosion energetics must be considered and what are reasonable estimates for the energetic yield. As a first priority, we are testing the hypothesis that mixing will be limited by local steam formation and high void fractions, causing water removal from within the fuel coolant mixture. With such data, the experimental results can be compared to the computational models (e.g., IFCI or PM-ALPHA).

To isolate the water depletion phenomenon from particle size distribution effects, experimental premixing simulations are carried out at the University of California at Santa Barbara (UCSB) using clouds of hot solid particles. A new instrument developed specifically for this purpose is used to measure the local liquid fractions in the three-phase mixing zone. The experiments are scaled (1/8-scale), using numerical simulations (PM-alpha) to create water depletion regimes similar to full-scale pours of fixed-size 1–2 cm corium particles. These experiments are currently in progress, and the results are expected to provide the first experimental confirmation of the water depletion phenomena as well as a basis for assessing the accuracy of its predictions in numerical simulations.

Additional melt breakup will occur as a consequence of the hydrodynamic interactions in the mixing zone. Two experimental programs are currently addressing this topic. The one at UCSB is examining the fundamental component of an explosion, a single melt drop, under conditions that effectively simulate the role of such a drop in a real explosion environment. This is accomplished by using a hydrodynamic shock tube capable of generating pressure pulses similar to those of an explosion (design pressure 20,000 psi). With x-ray diagnostics, the detailed time-history of fragmentation can be composed, and the results are the key input in numerical simulations of escalation/propagation. This work is now completed; a report will be published by the end of CY92.

A second set of vapor explosion experiments is being planned at the University of Wisconsin. The purpose of these experiments is to produce a well-controlled one-dimensional geometry in which a fuel simulant (e.g., 2–20 kg tin at 1300°K) pours into a water column, mixes with the coolant, an explosion is triggered, and the explosion expansion work is measured. These experiments are aimed at providing benchmark data to examine the effect of fuel-coolant initial conditions and mixing on explosion energetics. The hypothesis for these tests is that fuel-coolant mixing and explosion processes occur under controlled conditions. Data from these experiments enable comparison among the fuel-coolant mixing and explosion models used to make the case for applicability at reactor scale. This program is expected to be completed by the end of CY93.

It is expected that the above-described work will provide the basis for improved assessments of alpha-failure as well as for several other special-effects assessments as they arise in BWR ex-vessel sequences, where the relevant "sizes" of the explosion and the corresponding level of energetics may be considerably smaller than those involved in alpha-failure considerations and where the need for best estimate, rather than bounding, analyses is of greater importance. For example, the sensitivity calculations in NUREG-1150 suggested that ex-vessel explosions in a Mark-II and Mark-III drywell may lead to drywell failure. However, more detailed analysis is needed to verify that this threat exists.

2.3.2.2 Research Program for Quenching

To calculate quenching, the melt-coolant interface area and the constitutive laws that define the interphase transfers (heat and momentum) must be determined. On the constitutive laws, a reasonable estimate can be made by extending 2-phase formulations. The program involving the determination of interfacial area requires the breakup history of the melt as it descends through the coolant. Quite clearly, such experimental data are next to impossible to obtain; our approach, therefore, has to be largely empirical. Available experimental data in this area are very limited, thus we have joined the program at the FARO facility in Ispra.

The objective of the FARO tests to be performed at the Joint Research Center in Ispra is to observe the integral behavior of fuel-melt quenching at high pressures under likely severe accident conditions. What makes these experiments particularly attractive from a technical standpoint is that they can be performed with real reactor materials (UO_2, ZrO_2, Zr) at temperatures and pressures that are prototypic of actual severe accident conditions and with the proper full-scale water depths for in-vessel accident situations. The instrumentation within the FARO-TERMOS facility is substantial and the data collected is extensive. Thus it is likely that computer code comparisons will be made to gain modeling insights into fuel-melt mixing and melt quenching in water.

Because these experiments are using prototypic materials under realistic initial and boundary conditions at the proper vertical length scales (e.g., water depth) the question of scale only becomes an issue relative to the size of the fuel-melt pour and the lateral dimension of the facility. The FARO facility has the capability to deliver a large mass of oxide melt under a variety of conditions. The fuel pour rate is planned to be within reasonable ranges for accident conditions, i.e., jet diameters 5–10 cm and entry velocities of a few meters per second.

Second, the FARO experiments could be considered representative of two types of geometric situations: (1) a single jet in a large water pool or (2) a unit cell of a multiple jet pour into the lower plenum. In either case, the adequacy of a FARO facility vessel to provide a properly scaled lateral dimension depends to some extent on the degree of melt quenching. It is important to note that the chamber cross-sectional area will be varied by a factor of two in the tests to be performed to specifically address this point. If the results of the scoping test (50 kg of melt) and the base case experiment (150 kg of melt) indicate that melt quenching is minimal (e.g., < 10% of the fuel quenched during the pour), the steaming rate will be low, level swell will be minimized, and the steam superficial velocity will be small. Under these conditions, the lateral dimension of the vessel will not be an important concern regardless of which scenario is considered. Conversely, if the results of the scoping test and the base case experiment indicate significant melt quenching, the scaling of the experiments considering the lateral dimension will be problematic for either scenario. Qualitatively, one would still expect the experiments to be quite informative and valid for reactor safety implications. However, quantitative interpretations of the tests must then account for the lateral dimensions and the likely large steaming rate, superficial velocity and all other consequences associated with substantial fuel and liquid water interactions.

Based on these considerations, it seems clear that the logical sequence is that the scoping test and the base case test should be performed in the FARO-LWR experimental program, then the scaling rationale should be reassessed, and finally, the parameters determined for future experiments. Currently eight experiments are planned for the test series.

2.3.2.3 Research Program for Reflooding

During a severe accident situation, there is little doubt that the primary efforts of the operators will be directed toward achieving a stable, coolable configuration by making water available to the reactor vessel (if loss of cooling was believed to be the cause of accident). An important question that must be considered is the likely consequences of these actions. Given the uncertainties in core melt phenomena and the state of the core during an actual severe accident, and given the intuitive drive to put water into the core in the event of an undercooling accident, it is likely that the operators would inject 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 should be fully cognizant of possible symptoms and responses of the plant to added water under such circumstances (e.g., molten core coolant interaction, increased hydrogen generation, increased containment pressure). The following addresses reflooding research relating to the in-vessel and ex-vessel parts of an accident.

2.3.2.4 Research Program for Reflooding (In-vessel)

The working hypothesis is that adding water would be highly beneficial and likely to terminate the accident at this stage. The fuel-coolant interactions associated with such addition would be benign, yielding quench, with the possibility of hydrogen production in a narrow time window when cladding is hot enough and still in a highly undistributed geometry.

The current research approach to this issue is to review past investigations in which water (or its simulant) has been added to a degraded core and to determine what the adverse effects could be and what the current state of knowledge (data and analysis) suggests. An initial review of past experiments suggests that a few tests have been performed as part of the CORA and PBF experimental program as well as the LOFT–FP2 test. In the past, simulant tests of coolant added to fuel debris were performed at Brookhaven, Argonne, and UCLA. Although limited in scope, these tests address water addition during in-vessel core degradation. The possible adverse effects should be understood and their impact minimized. Possible adverse effects include:

- Hydrogen generation and fuel heatup from exothermic metal-water reactions,
- Energetic FCIs that may adversely affect attainment of a stable coolable state.

The major variables for determining whether water addition would have adverse effects would be the rate and character of water addition and the state of the fuel at the time it is added. To help in focusing this work, qualitative scenarios of the accident were developed with reference to water availability and its effect. On the basis of qualitatively different fuel-coolant contact configurations, the fuel states are:

1. Initial heatup and core degradation (rods intact),
2. Advanced core degradation (core rubble, melt, and relocation).

In order to better understand the accident management issues for this stage of the accident, we are collecting available information on core degradation morphologies and implied permeabilities as affected by fuel rod disintegration or clad relocation.

2.3.2.5 Research Program for Ex-Vessel Debris Coolability

One of the most important phenomenological issues in the progression of severe accidents after the reactor pressure vessel has failed is whether sufficient energy can be removed from the discharged molten debris that the plant can be brought to a stable condition and the challenge to containment integrity, whether by basemat penetration or by containment overpressurization, is avoided. The most commonly available mechanism for removing energy from the ex-vessel molten debris in LWR containments is water addition. Issues that must be resolved to develop debris coolability criteria are (1) the nature and configuration of debris, (2) the heat extraction mechanism from debris by water, and (3) the molten debris-containment structure interaction during the cooling process.

In order to obtain data to support the development of coolability criteria, an experimental program called MACE was developed under the sponsorship of NRC, EPRI, and several OECD countries. The program is intended to determine the ability of water to cool molten core debris during MCCI and to enable characterization of the resulting debris for assessment of permanent coolability.

In August 1989, a scoping test was conducted in the MACE series of tests in which approximately 109 kg of UO_2–ZrO_2–Zr melt in a 30 cm x 30 cm x 15 cm pool interacting with limestone-common sand concrete was flooded with a 50-cm head of water. The melt decay heat was simulated by direct electric heating; however, operational problems caused the input power to significantly exceed the decay heat (by a factor of about 3). The results from the test indicate heat transfer rates of 2.4–3.5 MW/m^2 during the initial period of water interaction. The rates decreased to 0.6 MW/m^2 and down to 150 KW/m^2 for the more quiescent period. The results support the concept of a thickening crust with periodic access of water to the melt and partial melt quenching.

A second MACE test, conducted in November 1991, employed approximately 430 kg of UO_2–ZrO_2–Zr–Fe_2O_3–CaO melt mixture in a test configuration of 50 cm x 50 cm x 25 cm. As in the scoping test, the melt pool had an overlying water with 50 cm head and an underbed of limestone-common sand concrete. The test was terminated after 25 cm of concrete ablation. The results indicate a heat transfer rate of 1 MW/m^2 during the initial period down to about 30–60 KW/m^2 during the quiescent period. The results further indicate formation of a suspended crust thereby preventing continuous melt quenching. The molten material underneath the crust ablated 25 cm of concrete at which point the test was terminated. Based on the post-test analysis of the second

MACE test, it was concluded that the initial and boundary conditions of the test were not prototypic. Thus, a third test was performed in April 1992 with corrected initial and boundary conditions, but otherwise in an identical configuration. Preliminary results from the test indicate some degree of coolability although any firm conclusion must await further analysis of the test data.

Currently, three more tests are planned in the MACE program. These tests will use variable depth of UO_2–ZrO_2–Zr melt mixture in two different geometrical configurations (50 cm x 50 cm x 75 cm) interacting with limestone-common sand concrete and siliceous concrete. Water will be added shortly after the MCCI has started. The objective is to measure the rate of heat removal by water; study the melt cooling process, including crust formation, stability, and growth; and to investigate the effect of various parameters (e.g., debris depth, cavity geometry, power density, concrete composition) on availability.

Concurrent with the MACE program and under the sponsorship of NRC, a second experimental program on debris coolability was initiated at the Sandia National Laboratories in February 1991. The goals of this program, called WETCOR, are to complement and augment the MACE program and to provide a means to support the assessment of existing as well as ALWR designs. The WETCOR program is designed to address two specific issues: (1) the comparative coolabilities of oxidic and metallic debris and (2) the limits of debris coolability in terms of debris composition and depth.

In the first WETCOR test (WETCOR–1) run in September 1991, 35 kg of charge material 80 w/o Al_2O_3—20 w/o CaO was heated to melting at 1850°K within a 32-cm diameter tungsten annulus. The charge and the annulus were surrounded by a cylindrical MgO crucible and supported on a 40-cm diameter by 40-cm high concrete basemat. After 2 to 3 cm of concrete ablation, the molten debris was flooded with approximately 25 cm head of water at a rate of 60 lpm. The results indicate approximately 5 to 6 cm of concrete ablation after 30 minutes of water addition. The results further indicate formation of relatively thick and layered crust, and partial melt quenching with voids between crust and quenched melt. No evidence of fragmentation is apparent from the post test examination of crust and frozen debris.

A second test, called WETMET, was performed in December 1991 to evaluate the potential for prolonged molten debris-water interactions which might result in bulk freezing or quenching of the debris. The charge material for the test was 92 kg of 304 stainless steel and the basemat was made of limestone concrete. The test was executed in two stages. In the first stage, the metal debris was heated to melting at 1990k, ablation began and water was added at 50 lpm for approximately 1 hour before the power was cut off to let the debris solidify and cool. In the second stage, water was added before power was applied to the new quenched meltpool configuration. Power was then increased until melting and concrete ablation were re-established. The results indicated an initial period of vigorous melt-water instability followed by a stable crust geometry with substantially reduced rates of energy transfer.

Results from the limited number of tests described above led the experts to believe that the debris coolability (a coupled FCI and MCCI phenomenon) may be an extremely complex process that demands a more careful examination of various factors contributing to the process. This gave rise to the concept of "morphological" testing, which deals with the determination of debris morphology as a function of experimental variables. The first of such morphological tests currently in the planning stage involves a prototypic charge material on a nonreactive (MgO) basemat.

2.3.3 Anticipated Results

The results of this research will identify the basic variables governing heat transfer and hydrodynamics of melt-water interaction, including the effect of water injection on debris configuration. Models supplemented by appropriate correlation to address all possible phases described above would be validated with available and planned experimental data. These models then can be used to assess the potential for and consequences of fuel coolant interactions, assess the efficacy of accident management strategies, assess the effect of FCI on altering accident sequences, and provide estimates for the rate of steam and hydrogen generation following reflooding by water.

2.4 Severe Accident Codes

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 accident phenomena or planning accident management strategies. A number of codes (e.g., SCDAP/RELAP5, MELCOR, CONTAIN, CORCON, COMMIX, HECTR, MELPROG/TRAC, VICTORIA, and BWRSAR) have been developed for various stages in severe accidents, both in-vessel and ex-vessel, for both PWRs and BWRs. However, as a result of a review of NRC-supported codes and associated documentation, the staff has terminated support for three severe accident analysis codes, HECTR, MELPROG, and BWRSAR. BWR-specific models developed under BWRSAR sponsorship are being incorporated into MELCOR and SCDAP/RELAP5. The HECTR and HMS codes were developed to test models of hydrogen mixing and combustion within reactor containments. MELCOR and CONTAIN have incorporated the models that were part of

HECTR for assessing hydrogen challenges from severe accidents. The HMS code is being documented, and no further development is planned. Support for MELPROG was terminated on the basis that it would be duplicative of SCDAP/RELAP5 as a detailed mechanistic in-vessel severe accident code, however, the IFCI module will be extracted from MELPROG to be used for fuel-coolant interactions analyses. SCDAP/RELAP5 is a less detailed, more flexible code and has been validated with considerable experimental data. The development and documentation of the original integrated risk analysis code, the Source Term Code Package (STCP) has been completed. The STCP, a collection of various codes with the MARCH code serving as the cornerstone, will no longer receive developmental or maintenance support.

The development of severe accident phenomenological models and computer codes continues to have a role in achieving the objectives of the severe accident research program. The codes now under NRC's sponsorship and support are SCDAP/RELAP5, MELCOR, and CONTAIN. In addition, several other codes such as VICTORIA, COMMIX, and IFCI are being developed and maintained to perform specific functions that require detailed modeling These codes will be used to benchmarking the system level codes discussed earlier. Also, the core concrete interaction code, i.e., CORCON, will be incorporated into the CONTAIN and MELCOR codes. The relationship of these codes to various severe accident progression phenomena is shown in Figure 2.4.1. Additional discussion on the status of each of these codes is provided below.

2.4.1 SCDAP/RELAP5

2.4.1.1 Research Needs

Since the 1979 Three Mile Island-2 (TMI-2) accident, the NRC has conducted a broad-based research program to develop an understanding of severe accident behavior. As far as practicable, much of this understanding, from initial core uncovery through reactor vessel failure, is contained in the SCDAP/RELAP5 computer code. SCDAP/RELAP5 is designed to model the coupled interactions that occur between the reactor coolant system (RCS), the core, and the fission products during a severe accident.

The objective of this research program is to develop an analytical tool (i.e., the SCDAP/RELAP5 code) that will provide the NRC with the capability to perform detailed analyses of in-vessel core melt progression phenomena during various severe accident conditions and scenarios.

2.4.1.2 Current Research Program

The development of SCDAP/RELAP5 has been based on a combination of factors, including model assessment and validation activities, insights and results of separate effects and integral experiments, and user needs to support resolution of various severe accident issues, and accident management. Since 1984, the code has been used to support the analysis of several major severe accident experiments such as PBF SFD 1-3/1-4, LOFT FP-2, ACRR DF-4, and CORA experiments. The code has been used for severe accident analyses, including natural circulation studies and the analysis of lower plenum debris and lower head heatup. Accident management studies have been performed which include the analysis of strategies or phenomena that minimize direct containment heating.

Systematic assessment of the newest version of the code [i.e., MOD3(8x)] will be performed to define modeling uncertainties in important severe accident phenomena. These uncertainties, along with uncertainties in system thermal-hydraulics from natural circulation experiments and design basis accident (DBA) experiments, will be propagated through plant-and accident-specific conditions to assess the uncertainties in conditions of RCS failure caused by natural circulation, a damaged core, and fission product and hydrogen releases during a severe accident. Some late-phase core-melt-related experiments will be planned and performed as discussed in Section 2.2. Once these experiments have been performed, models to treat late-phase core melt progression that are not adequately modeled in the code will be developed, and systematic assessments will be performed.

Preliminary assessment of earlier versions of SCDAP/RELAP5 that began in 1991 have identified the relative strengths and weaknesses of many models in the code. In general, many predicted early-phase phenomena, such as system temperature, fission product release, and the initial change in the core geometry caused by ballooning and melting of core structures, have been within the experimental uncertainties for the individual experiments. Assessment results have also revealed several areas that will require model improvements. Specific areas include (1) renewed bundle heating, melting, hydrogen production, and fission product release during reflood, (2) initial relocation of molten material as droplets and rivulets, (3) interaction between fuel rod cladding and Inconel grid spacers, and (4) the diversion of flow from damaged fuel assembles. Removing these modeling deficiencies will significantly reduce the calculated phenomenological uncertainties in the early-phase core melt progression. As a result, the end state of in-vessel core melt progression can be more accurately determined. This in turn should contribute to a better estimate of containment loading, fission product release, and perceived overall risk for severe accidents. Thus, efforts devoted to removing these code deficiencies are the main focus of current SCDAP/RELAP5 research activities. Other research activities include (1) incorporation of ORNL-developed BWR model

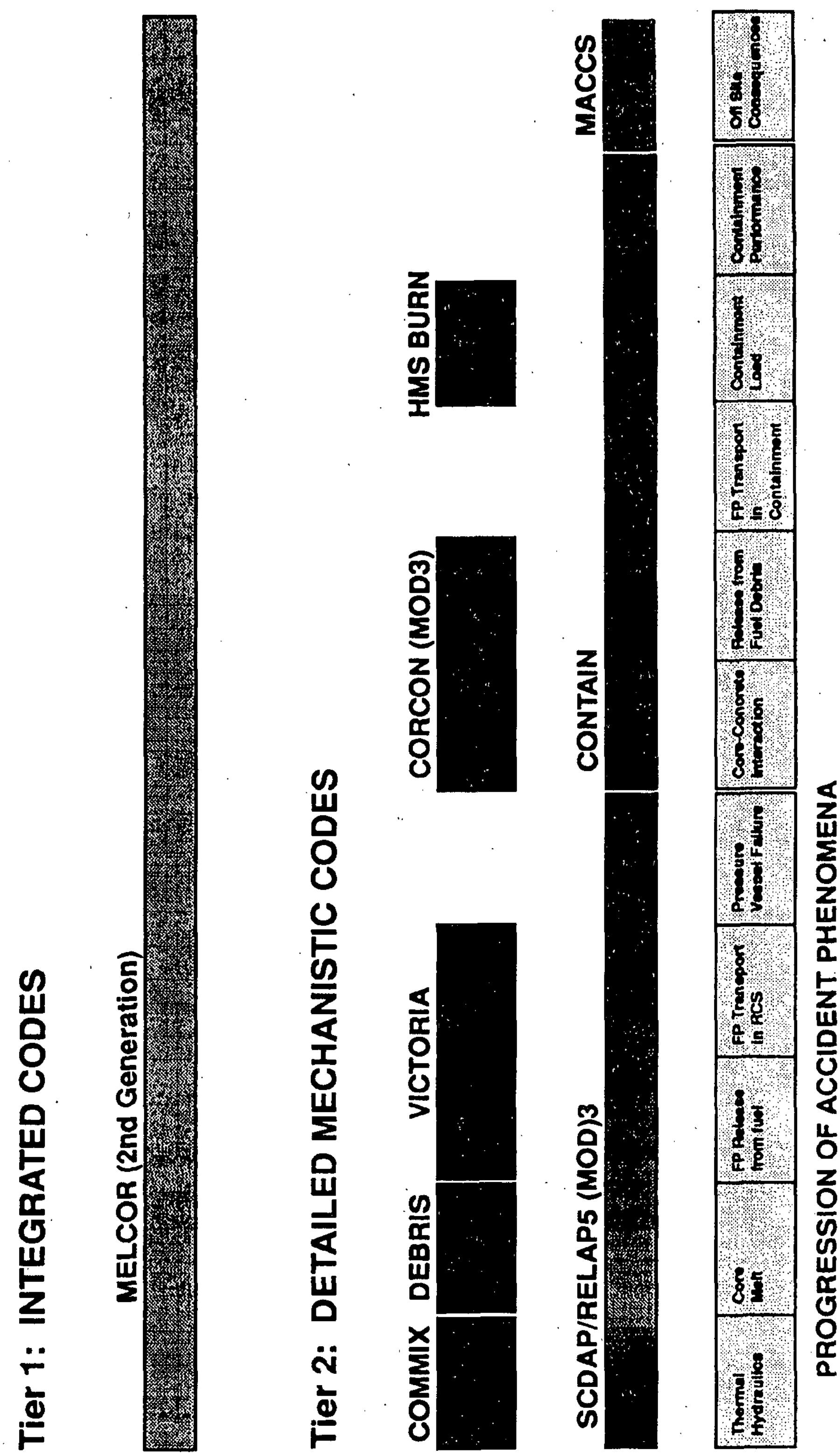

Figure 2.4.1 NRC Severe Accident Codes

improvements for current generation BWR as well as for the advanced SBWR, (2) SCDAP/RELAP5 peer review, and (3) model extension to treat Westinghouse AP600 core design changes, including core structures and materials that differ from current PWR designs. The purposes of the SCDAP/RELAP5 peer review effort are (1) to provide an independent, high-quality review of the code, (2) to help the NRC determine future code development direction, effort, and priority, (3) to provide information concerning code deficiencies and limitations that need to be improved or corrected, and (4) to determine the degree of technical adequacy of the code.

As for the late phase of severe accidents (e.g., from the formation of bundle-size blockages through the growth of molten pools and the relocation of molten material into the lower head), the core melt progression phenomena are still poorly understood because of the lack of experimental data. As the late-phase core melt progression data become available, models will be developed and incorporated into the code. Modeling assessments against experimental results have been made (see Table A.2.1). Additional modeling assessment and validation efforts will continue to be made to ensure that SCDAP/RELAP5 meets the code's design objectives and targeted applications.

2.4.1.3 Anticipated Results

Uncertainties in predicting vessel and RCS failure times and vessel failure modes using SCDAP/RELAP5 will be substantially reduced because of the extensive data base (including late-phase experimental results) and model improvement efforts. The end product of this research program is to provide NRC with a computer code that is capable of performing (1) plant (both PWR and BWR) analysis for the in-vessel core melt progression phenomena for various severe accident scenarios and (2) experimental analysis and support for in-vessel severe accident experiments. Other uses of the code include (1) assessment of the efficacy of accident management strategies, (2) MELCOR benchmarking and assessment, (3) TMI–2 accident evaluation, and (4) support for ALWR design certification as defined in SECY–91–161.

2.4.2. CONTAIN

2.4.2.1 Research Needs

The CONTAIN code is a detailed mechanistic code for the integrated analysis of containment phenomena. It has been developed under NRC sponsorship to provide the capability to predict the physical, chemical, and radiological conditions inside a nuclear reactor containment in the event of a severe reactor accident, as well as fission product releases to the environment in the event of containment failure.

The CONTAIN code models intercell flow, hydrogen and carbon monoxide combustion, heat and mass transfer (radiation, convection, conduction), aerosol behavior, fission product behavior (decay heating, transport), engineered safety systems (sprays, fan coolers, ice condensers), BWR-specific systems (suppression pools and safety relief valve discharge), core-concrete interaction, and simple treatment of direct containment heating. The code provides the capability to analyze a wide variety of LWR plants and accident scenarios.

One of the safety issues that is currently under extensive investigation is direct containment heating (DCH) and pressurization of the reactor containment atmosphere by molten core materials ejected following the lower head failure of the reactor vessel under pressure. DCH involves a large array of complex processes, many of which are only now becoming better understood and for which little or no experimental data previously existed.

Several research activities are ongoing in an attempt to quantify the effects of DCH (See Section 2.1). Historically, the NRC DCH experimental program was guided by calculations and sensitivity studies performed with the CONTAIN code. Although these sensitivity studies and pretest and posttest predictions were useful in identifying important processes, there was concern over the degree of confidence in the code's treatment of the DCH processes to determine the relative importance of these processes in the tests. In FY90, a comprehensive scaling methodology was developed and implemented for the purpose of ensuring a properly focused and technically defensible direct containment heating experimental program. This program will guide the development of models for incorporation into the CONTAIN code.

Another key research need is related to containment analyses for ALWR designs namely AP600 and SBWR plants. Containment designs are being developed for ALWRs that incorporate passive cooling and decay heat removal features for protection against long-term containment overpressure in accident situations. The passive nature of these containment systems poses unique challenges to containment analysis codes for predicting containment response in both design basis and beyond design basis events. Such challenges which require new or improved models include natural circulation air flow in the channel outside the containment shell, behavior of an evaporative flowing water film on the containment shell, stratification of gases within the containment, potentially unique heat transfer and condensate film behavior not adequately represented by existing correlations, and numerical challenges arising from the need to efficiently perform long-term containment response calculations. Similarly, the performance of the passive containment cooling system (PCCS) in the SBWR will be modeled.

2.4.2.2 Current Research Program

In FY92 and FY93, selected DCH models will be incorporated into the CONTAIN code and evaluated against the related experimental data. Then, actual plant cases will be performed with the updated CONTAIN code to determine the impact of DCH on containment pressurization.

Another primary objective of the current research program is to develop and validate models related to the ALWR performance for both design basis accidents and severe accidents. The CONTAIN code will be modified accordingly, and industry experimental programs will be evaluated and used to validate the CONTAIN code.

Other research activities include efforts to update the CONTAIN code consistent with the improvements in modeling already developed under other programs. Currently, only iodine washout by containment sprays is modeled mechanistically in CONTAIN. The TRENDS code developed at ORNL is able to provide mechanistic modeling of iodine chemistry and other fission product physical and chemical processes. Therefore, incorporation of this package into the CONTAIN code would further advance the overall capability to predict fission product release constituents and fission product spatial heating effects. This effort will be completed in FY93. Core-concrete interaction phenomena modeled in CONTAIN are based on the CORCON-MOD2 code. Release of the CORCON-MOD3 code, which includes the improvements accumulated over approximately 6 years, necessitates the update of the corresponding CONTAIN model; this effort will be completed in FY93. Also, Figure 2.4.2 illustrates past and present validation and assessment activities to compare experimental tests against code model predictions.

As a result of incorporating the various selected models in CONTAIN, the code manual will be updated. Also, to ensure the CONTAIN code meets its design objectives and need for regulatory application, a peer review will be undertaken by independent experts. The peer review will begin in CY93, and will be completed in about 12 months after initiation.

2.4.2.3 Anticipated Results

As a result of upgrading the CONTAIN code, actual plant analyses will be able to be performed to determine the impact of DCH on containment loading. Also, the code will be capable of performing ALWR containment analyses for both design basis accidents and severe accidents for the AP600 and SBWR plants. In concert with these efforts, the CONTAIN peer review will provide an overall independent assessment of the code. This assessment will assist in steering future efforts on this program.

2.4.3 MELCOR

2.4.3.1 Research Needs

Research needs for the MELCOR code fall into a number of areas that can generally be divided into development or assessment. The development areas include (1) adding calculational capabilities (or models) into the code, (2) correcting or improving existing calculational routines (or models) in the code, (3) correcting code errors, and (4) improving code documentation. The assessment area includes (1) comparing calculational results with experimental data or output from other code calculations, (2) varying input parameters to characterize the sensitivity of results for specific types of calculations, and (3) determining reasons for differences between the results of calculations and experimental data.

A comprehensive independent review of the MELCOR code was undertaken by a peer review committee. The objectives of this review were to (1) provide an independent assessment of the MELCOR code, (2) determine the technical adequacy of the code, and (3) issue a final report describing the technical findings. The review committee recommended improvements in five areas: (1) MELCOR numerics, (2) missing models, (3) revisions to existing models, (4) expanded assessment, and (5) documentation. They concluded that when these recommendations are satisfied, MELCOR will be technically adequate for PRA applications, although the code may not always be sufficient for some parametric accident management studies. The peer review committee issued its finding in 1992 (report LA-12240).

2.4.3.2 Current Research Program

The current focus of the MELCOR research efforts are on (1) resolving time-step size and machine-type dependencies (numerics problems) as urged by the peer reviewers, (2) correcting inadequacies in some of the phenomenological models that were pointed out by the peer review committee, (3) adding missing models into MELCOR, and (4) expanding the technical assessment program to provide assurances that the code is reliable, or, if deficiencies are found, to identify those deficiencies so that remedial actions may be taken.

In regard to the five areas of recommendations by the peer reviewers the following actions are planned.

2.4.3.2.1 *MELCOR Numerics:* Efforts have been directed toward investigating problems and sensitivities associated with code numerics including time-step and machine-type dependencies, identifying their underlying causes, and eliminating or mitigating them. Efforts to remedy the numerics problems started in FY91 and those problems identified through 1991 have been settled; others are being investigated in FY92 and FY93.

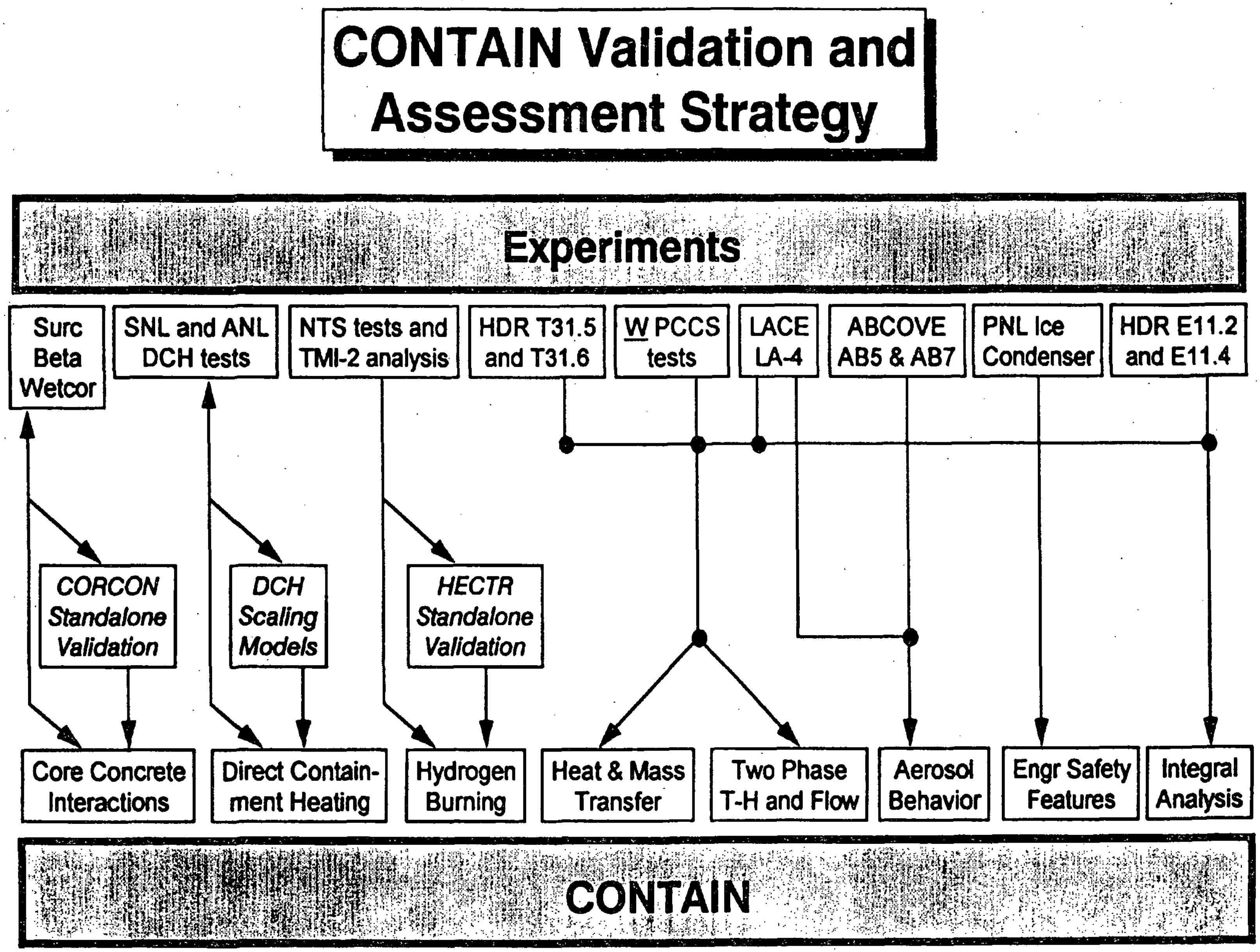

Figure 2.4.2 CONTAIN Code Validation and Assessment

2.4.3.2.2 *Missing Models:* The review committee concluded that models for the following phenomena should be given the highest priority for incorporation in MELCOR:

- PWR primary system natural circulation in components with countercurrent flows,
- high-pressure melt ejection and DCH,
- ice condenser containments,
- nonexplosive interactions between debris and water,
- fission product vapor scrubbing,
- additional reactor coolant system fission product deposition processes, and
- fission product reactions with surfaces.

Model development activities were initiated in FY91 for three missing models: natural circulation, direct containment heating, and ice condenser containment. Work on the remaining models dealing with fission products is being integrated into the current workscope. The effort to incorporate a model into MELCOR for handling direct containment heating phenomena was based on modeling insights derived from the DCH experimental and analytical programs developed through interactions between the experimental group and the code developers. Incorporation of appropriate DCH modeling into MELCOR was completed in FY92. Completion of the implementation of a relatively fast-running model for calculating natural circulation that is consistent with the MELCOR architecture and execution time requirements is scheduled for FY92.

Three fission product behavior models, identified as missing by the MELCOR peer review committee, are being given consideration for inclusion in the source term calculation, especially for PRA use. These are described as follows: Although MELCOR has a relatively sophisticated model for scrubbing aerosols in a water pool, in some sequences gas temperatures in the primary system may be sufficiently high that volatile fission product vapors, rather than aerosols, would be discharged into pools. MELCOR does not represent the removal of these vapors as they condense to aerosols and attempt to pass these aerosols through the pool. Examples of sequences in which this process is important include low-pressure BWR sequences with discharge of vapor through the safety relief valve lines to the suppression pool or low-pressure PWR containment bypass sequences with discharge to a water pool in the auxiliary building. In such sequences, this phenomenon could dominate the calculation of the source term, especially when other fission product removal mechanisms are weak.

MELCOR's fission product deposition models are adapted from the MAEROS containment model. As such, certain processes that are not generally important in the containment have been neglected. These include impaction and turbulent deposition of aerosols. Experimental data on containment bypass sequences performed for the Electric Power Research Institute, as well as calculations using more comprehensive aerosol deposition models, indicate that the neglect of these processes may result in a significant underestimate of the retention of aerosols in the primary system, especially for low-pressure sequences in which gas velocities are high. Correlations which have been developed to represent these effects are included in the VICTORIA code and consideration will be given to adopting those or simpler versions in the MELCOR code.

Chemical reactions between settled aerosols and vapors and heat sinks in the primary system can greatly affect deposition (chemisorption) and revaporization rates. The reviewers state the lack of explicit modeling in the code applies to all accident sequences and is particularly serious for cesium hydroxide and tellurium compounds. Simple models for these effects which capture the most important effects during accident sequences, as determined by the detailed VICTORIA code, will be considered for MELCOR in the future.

2.4.3.2.3 *Revisions to Existing Models:* The Committee recommended that the following concerns with existing MELCOR models be given the highest priority.

- Condensation is treated independently in the codes hydrodynamic behavior modeling from those calculations of aerosol particle growth and deposition in the radionuclide modeling portion of the code. The validity of this approach should be demonstrated by comparison with more exact models or data.
- Inconsistencies in the treatment of chemical reactions between CORCON and VANESA should be resolved, and improvements should be made to the CORCON/MOD2 phase diagrams.
- The model for condensation in containment (mass transfer) should be revised.
- There seems to be a disagreement between the pool decontamination factors computed with the current pool scrubbing model and those calculations made using other codes. A general feeling is that there may be an implementation error, however, no such error has been located.

The acceptability of using the hydrodynamics package water condensation/ evaporation model for calculations of aerosol particle growth and deposition in the radionuclide package is being addressed.

Improvements in the oxidic and metallic phase diagrams are needed because the effects of eutectic formation on debris layer melting and solidification behavior are important. Current incompatibilities in material properties between the cavity modeling package and other MELCOR packages are being eliminated, and duplicative and inconsistent chemistry between CORCON and VANESA are being eliminated by the implementation of CORCON/MOD3 into MELCOR.

The model for condensation in the presence of noncondensables is being revised to address identified deficiencies, including modeling of the film thermal resistance and high mass transfer effects.

The pool scrubbing model discrepancy will be resolved.

2.4.3.2.4 *Expanded Assessment:* The committee also concluded that the ability of MELCOR to calculate severe accident phenomena has not been demonstrated sufficiently. Such a demonstration should be based on (1) sensitivity studies, (2) benchmarking activities using experimental data, and (3) code-to-code assessments.

A plan for a more comprehensive integral assessment of MELCOR is under development. The technical assessment planning involves a number of related activities, including verification, validation, and quantification of uncertainties. This process involves reviewing models and comparing analytical results to experimental data, including small-scale and full-scale experiments. Only a small portion of the comprehensive plan has been accomplished so far. It is a high priority to have at least some validation results as soon as possible for each of the major phenomena treated by the code. There is a need to develop a standard code package demonstration/test problem for many of the MELCOR code packages. At least one assessment problem for each major package or for each major phenomenological area will be prepared (e.g., the ice condenser model being added to the general HS package). Also, the program is being carefully designed to provide more substantive assessment of the COR and RN packages, which were major areas of concern to the peer reviewers. A graphic description of the assessment program comparing analytical results and experimental data is given in Figure 2.4.3 which relate code phenomena being validated to applicable experiments. Additional assessment activities involve code to code comparisons and analysis and evaluation of full plant transient calculations for various plant and accident sequences.

Consistent with the draft assessment plan, activities concentrate on PRA risk-dominant sequences, such as station blackout, steam generator tube rupture, and V-sequences. The Semiscale test selected for primary systems thermal-hydraulic and heat transfer assessment complement and support these selected demonstration plant analyses. The work in this area is intended to be combined with work on other plant geometries (e.g., B&W PWRs, BWRs) for similar sequences in both experimental analyses (using MIST and FIST test facilities, from Semiscale-sized equivalent facilities for those other geometries) and for full-sequence demonstration analyses for other plant types. Demonstration calculations to analyze B&W and Combustion Engineering PWRs are being run at Brookhaven National Laboratory.

In addition, the NRC is in the process of initiating an international cooperative effort for technical assessment of the MELCOR code, the MELCOR Code Assessment Program (MCAP). The objective is to accelerate the technical assessment, consistent with the peer reviewers comments, by employing the expertise of many of the code users both inside and outside the U.S. Thus, many cases can be run in a shorter time. This will also allow expansion of the user community and at the same time improve the understandings and abilities of the users to run the MELCOR code.

2.4.3.2.5 *Documentation:* The body of existing MELCOR documentation is significant. However, the Committee felt that detailed descriptions of the models and correlations were lacking in some cases, and documentation on the applicability and benchmarking was either inadequate or missing. The Committee also recommended that a process for collecting, documenting, and distributing user guidelines to the MELCOR user community be developed.

Improvements to the MELCOR documentation are planned. However, documentation equivalent to the TRAC "Models and Correlations" code document would be highly resource intensive and somewhat duplicative of information already available in the MELCOR manuals. Nevertheless, in FY93 some resources are directed to upgrading reference manuals to cover the most critical needs. Further, there is a task to develop a practical user guidebook for MELCOR users.

2.4.3.2.6 *Additional Research*

Not all of the code development and assessment work is a result of the peer review. Other research activities include: (1) incorporation of ORNL-developed BWR model improvements, (2) incorporation of an upgrade of the CORSOR fission product release model including use of the Booth model, (3) improving core debris relocation modeling to include spreading, (4) improvement of core melt modeling to allow simple material (eutectic) interactions to be treated on a parametric basis, (5) further expansion of the technical assessment program to involve other DOE laboratories engaged in severe accident research for the NRC, and (6) testing of an input model for AP600.

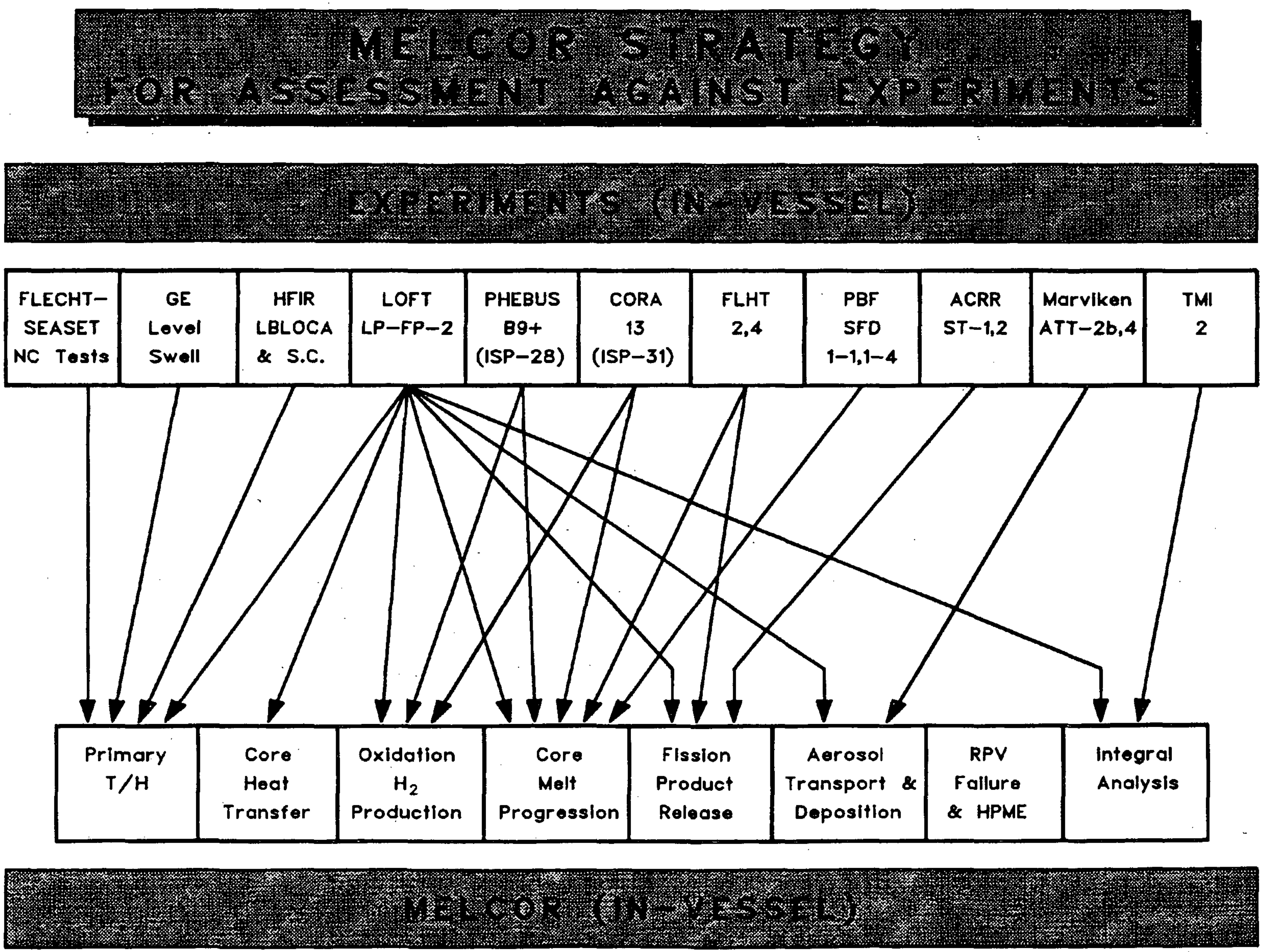

Figure 2.4.3 MELCOR Code Validation and Assessment

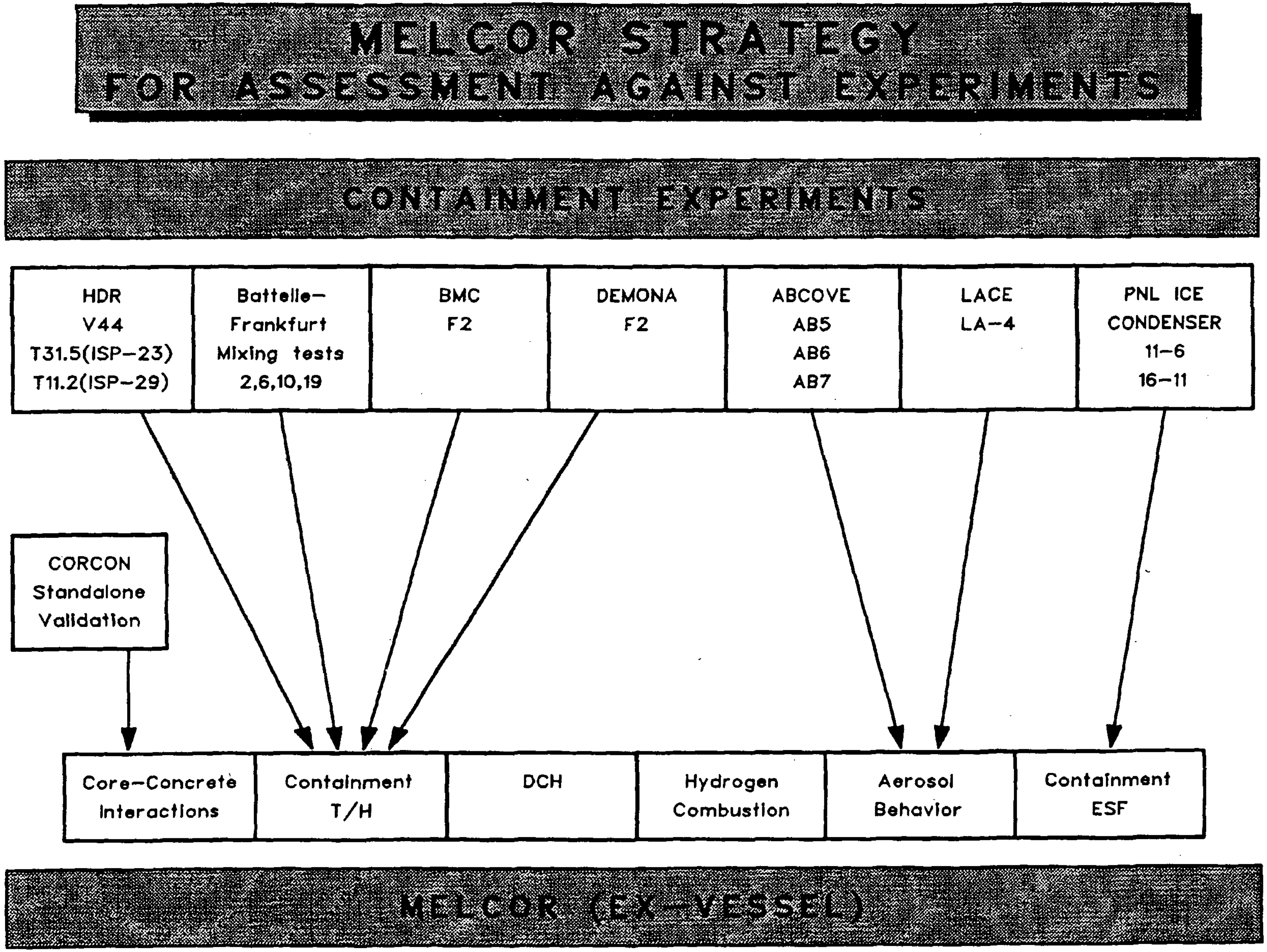

Figure 2.4.3 Continued

Material interaction models for the in-core region will be incorporated into MELCOR. ORNL has arranged the programming of the BWRSAR eutectics model for incorporation into MELCOR. This model is limited to BWR applications and to lower plenum debris bed behavior.

In another important developmental area, work is ongoing and planned for the evolutionary light water reactor (ELWR) and ALWR applications. MELCOR is currently capable of handling evaluations of transients in ELWR primary systems and containments. It is also capable of treating most features of ALWRs. MELCOR will be benchmarked against results of detailed codes to determine its adequacy and areas of development needed for ALWR applications. In FY92, an activity is under way at Brookhaven to develop and test an input model for the Westinghouse AP600 design. Similar activity is under way at ORNL for SBWR. Based on the results of the input model effort and on initial runs with the model, additional recommendations will be developed for further code improvements to integrate MELCOR capabilities for ALWRs.

2.4.3.3 Anticipated Results

Within the context of the current research program, the results of the MELCOR peer review, and the continuing plans for MELCOR development and assessment, the anticipated results of the MELCOR program are:

1. Development of a second-generation integrated severe accident code that should (1) appropriately model phenomena essential to the understanding of severe core damage accidents, (2) provide predictions of the progression and consequences of severe core damage accidents, (3) permit estimates of the uncertainties associated with such predictions, and (4) have a structure that facilitates the incorporation of new or alternative phenomenological models based on the ongoing experimental research program.

2. Provide a code that includes major phenomenological developments from severe accident research in adequate detail to address the phenomenology and also has a practical running time for severe accident analyses.

3. Provide continuing maintenance and user support for the MELCOR code.

Within a few years, all the important models for severe accident analyses will have been added to the MELCOR code and the code will have been assessed for its adequacy.

2.4.4 COMMIX

2.4.4.1 Research Needs

In the past years, research in the area of code development for reactor safety analysis was mainly focused on developing and validating one-dimensional system codes such as SCDAP/RELAP5 and MELCOR. However, a number of phenomena encountered in postulated severe accidents are inherently multidimensional in nature. Typical examples are: natural circulation, flow stratification, countercurrent flow in a pipe, hydrogen distribution and mixing processes, and the effect of noncondensible gas distribution on local condensation and evaporation.

The unique features of the passive containment cooling system in the proposed Westinghouse AP600 plant are specifically designed to prevent damage to the containment during design-basis events or severe accidents. The passive cooling functions are carried out via natural circulation inside or outside the containment. The COMMIX code (originally developed for analysis of transient fluid flow and heat transfer in the reactor coolant system) is being modified so that it can be used for the analysis and understanding of transient fluid flow and heat transfer phenomena in the containment. The research needs for this program are to assess and evaluate the capability of COMMIX for use in analyzing the new and unique features of ALWR plants during design-basis events or severe accidents, and to apply the COMMIX code to perform audit calculations.

2.4.4.2 Current Research Program

The main objectives of this program is to improve and extend COMMIX's capability and then in turn use COMMIX to assess the adequacy of the unique passive containment cooling system designed specifically to prevent damage to the AP600 containment system during design-basis events or severe accidents.

To satisfy the objective above, some code assessment efforts are being carried out, and a number of new models are being developed and implemented into COMMIX. The following is a list of new models, needed for the analysis of the Westinghouse AP600 design:

1. Multicomponent capability to compute the distribution of steam, air, and hydrogen throughout the reactor containment.

2. Liquid film tracking model to compute liquid film thickness, velocity, and temperature on the internal and external containment steel shell to calculate transient heat removal from the AP600 containment. The AP600 steel shell consists of semi-elliptical domes at the top and bottom of a cylinder. The internal liquid film is formed as the result of condensation of steam on the liner wall of the steel

shell. The external liquid film on the steel shell wall is formed from water flooding at the top of the dome. Evaporation of this liquid film then occurs as the buoyancy-driven air stream passes through the annulus outside the steel shell.

3. Heat and mass transfer models which must be validated with the Westinghouse AP600 Passive Containment Cooling System (PCCS) small scale and 1/8-scale test data.

4. Radiation model to account for heat transfer from the steel shell to the air baffle wall in the AP600 design.

The multicomponent capability has been implemented in the COMMIX code, and a limited validation of this capability was carried out using steam blowdown data from a full scale vessel in Germany (i.e., HDR ISP-23). Further validation effort is planned. The development of the liquid film tracking model has been completed. Both the liquid film tracking model, and heat and mass transfer models implemented in COMMIX are being validated with the Westinghouse PCCS small scale data. The validation effort with 1/8-scale test data will soon be carried out. The need to develop a radiation model to treat the radiation from the dry patch of the containment steel shell to the baffle wall will be made in FY93.

2.4.4.3 Anticipated Results

With upgrading of the COMMIX code, it will be capable of performing containment analysis of ALWR for both design-basis events and severe accidents. Also, COMMIX can provide multidimensional information and serve as a benchmarking tool for the CONTAIN lumped parameter system code.

2.4.5 VICTORIA

2.4.5.1 Research Needs

The analyses for NUREG-1150 have shown that bypass accidents, such as steam generator tube ruptures to be the dominant risk accidents for some classes of plants (subatmospheric PWRs and ice condenser PWRs). There is substantial uncertainty in the risk associated with these accidents because the accident analysis tools available for the NUREG-1150 analyses omit models critical to proper calculation of radionuclide retention and revaporization in these sequences. The accident analysis tools appear to underpredict retention in the reactor coolant system. The VICTORIA code was developed to address this uncertainty related to the bypass accident.

Current state-of-the-art models of core degradation indicate that much of the reactor fuel will remain in the vessel and the core region at the time of vessel failure. Tests in Canada (CRNL) indicate that, especially for PWRs, radionuclide release from fuel retained in the vessel in an air environment will be radically different from release during early stages of core degradation. Plant configuration during shutdown situations may also lead to the possibility of air ingress into the core either by natural circulation or from the residual heat removal system. The experimental data has to be obtained for the fuel-release model of VICTORIA before one could estimate the effects on risk associated with the radionuclide releases of these types.

Deposited radionuclides may be resuspended in the reactor coolant system when the system is depressurized either as a natural event of the accident or as a deliberate measure to mitigate the possibility of direct containment heating. VICTORIA has incorporated such a model. Currently, there are no suitable experimental data to validate revaporization model, but the PHEBUS-FP tests may provide some.

2.4.5.2 Current Research Plan

The VICTORIA code is developed under an international collaborating efforts including representatives from Sandia National Laboratories (SNL), Argonne National Laboratory (ANL), Oak Ridge National Laboratory (ORNL), Battelle Columbus Laboratory (BCL), Chalk River Nuclear Laboratory (CRNL) and Winfrith Technology Centre (WTC). SNL is the principal developer of the code while other establishments are providing specific model(s) in their area of specialty. The code and a user's manual was first released in October 1990. An updated version of the code has been completed in May 1992, and the revised user's manual and the code are scheduled for release by the end of FY92. The VICTORIA code can now provide predictions adequate for reactor accident analyses of:

1. Radionuclide release during the early stages of core degradation,

2. Aerosol processes in the reactor coolant system,

3. Vapor deposition in the reactor coolant system,

4. Radionuclide release during later phases of core degradation, especially once the reactor vessel has failed,

5. Resuspension of deposited materials at the time of reactor coolant system depressurization,

6. Radionuclide entrapment and revaporization in ruptured steam generator tube accidents and other bypass accidents.

Distinct classes of experimental data are being used to validate the various aspects of the code:

1. Release During Core Degradation

Separate-effects tests such as the out-of-pile HI and VI under steam and hydrogen conditions tests done at ORNL and the ST–1, 2 in-pile tests have been used for much of the validation of VICTORIA. French out-of-pile tests (PHEBUS–SFD) and in-pile tests (PHEBUS–FP) with irradiated fuel will be examined in the future. More integral tests such as the PBF tests, the LOFT–FP test, and the FLHT tests may be used for validation of the release models once clearer portrayals of thermal-hydraulic processes are available for these tests.

2. Release During Late Phases of Core Degradation

There are no suitable test data to validate models of release once fuel has melted and the core geometry has been lost. Integral experiments to study fission product release from late-phase core melt progression and revaporization are not being planned in the United States. However, the PHEBUS–FP project could provide some of the needed experimental data.

3. Transport in the Reactor Coolant System

Results of the Marviken and LACE tests are being used to validate the aerosol transport models in VICTORIA. Results of the Argonne and Sandia National Laboratory tests as well as tests now under way in the United Kingdom (WTC) are being used to validate models of chemical processes affecting radionuclide transport. More integral validation will be provided by results of the PHEBUS–FP tests.

4. Revaporization of Deposited Radionuclides from the Reactor Coolant System

There are not suitable data to validate the models of revaporization. The PHEBUS-FP tests may provide some.

5. Release and Transport during Shutdown or After Vessel Failure

Separate-effects tests done at CRNL are available to validate some chemical aspects of models of release and transport in the reactor coolant system after vessel failure. In addition separate-effects tests on fission product in the presence of air will be performed at ORNL. There are no integral data on core degradation during this phase of an accident.

In FY92, developmental assessment will be completed and documented. A peer review of the code will be initiated in FY93; after peer review systematic assessment of the code will be carried out. Meanwhile, the code is used for the planning of post-test samples analysis and pre-test calculations for integral test for the PHEBUS–FP project. In addition, VICTORIA is also used for the International Standard Problem (ISP) 34 exercise for the FALCON fission product chemistry experiments in the United Kingdom.

2.4.5.3 Anticipated Results

The end product of this research is to provide NRC with a computer code that models the radionuclide and non-radionuclide materials release, transport, and deposition within the reactor coolant system under severe accident conditions. It will be capable of performing (1) plant (both PWR and BWR) analysis for the in-vessel fission product behavior for various severe accident scenarios, and (2) experimental analysis and support for in-vessel fission product experiments. Specific uses of the code include (1) PHEBUS–FP post-test samples analysis, and pre- and post-test analysis, and (2) FALCON ISP–34 exercise. The code is also being used by CRNL for the Blowdown Test Facility pre-and post-tests analysis and by WTC for assessment of full plant behavior under severe accident conditions.

2.4.6 Integrated Fuel Coolant Interaction

2.4.6.1 Research Needs

In the event of a severe accident leading to core melt, molten fuel materials can come into contact with water, producing a fuel coolant interaction (FCI). FCI's can occur for a variety of in-vessel and ex-vessel conditions, including: reflood of a partially molten core, melt pouring into the lower plenum, and melt pouring out of the vessel lower head into a water-filled reactor cavity under low or high pressure. In these situations non-explosive or explosive FCI's may occur. The mode and resulting energetics of FCI's depend on the complex interaction of various thermal, physical, and chemical processes, including: coarse mixing, particle fine fragmentation, and heat transfer.

The goal of the Integrated Fuel Coolant Interaction (IFCI) effort is to provide a stand-alone code that embodies an integrated models for these processes so that FCI severe accident events can be calculated for full scale plants. Presently the IFCI code has phenomenological models for the pre-explosion mixing phase of FCI's, and includes multi-phase, multi-dimensional, three fluid hydrodynamic equations required to represent non-explosive events and the thermal detonation and expansion phases of explosive FCI's. Mechanistic models for triggers that initiate explosive FCI events are not included; however, there are provisions for representing intentionally imposed triggers, such as those in FCI experiments.

The fundamental physics of FCI's are being experimentally investigated as described in Section 2.3 of this document. This work is largely focused on understanding triggers and how they affect FCI's, and the development of separate effects models. There is a need for a code that will embody the models that result from this work and others as our understanding of FCI's increase. This need stems from the fact that realistic modeling of FCI's requires that the complex interactions between the governing processes be captured as well as the processes themselves. The code is intended to fulfill the need for a tool to quantitatively describe non-explosive FCI's, and explosive events when the triggering mechanism is known. Such a predictive tool is needed to help answer questions regarding the impact of FCI's for present and future nuclear power plants.

2.4.6.2 Current Research Program

FY92 and FY93 efforts will focus on making an IFCI a stand-alone predictive tool for FCI events. Presently IFCI is a module of the inactive MELPROG systems code. The thrust of the present program is to extract IFCI from MELPROG. Only these modules needed to represent FCI's and the interaction of the governing processes of FCI's will be retained. This effort will include testing of the resulting stand-alone code and correction of any models or algorithms required to obtain a robust analysis tool. The code will be documented in a draft NUREG report and peer reviewed in FY93 prior to issuing the final report.

2.4.6.3 Anticipated Results

The anticipated result of this work will be a stand-alone IFCI code that can be used on workstation to analyze non-explosive FCI events, and explosive events when the triggering mechanism is known. An operational report that demonstrates the applicability of the code to full scale plants will be provided along with the peer reviewed documentation report.

2.5 BWR Mark I Containment Liner Failure

2.5.1 Current Research Program

As stated in Appendix A.1, the NUREG/CR–5423 peer reviewers identified three areas that warranted additional research to confirm the appropriateness of the analysis in NUREG/CR–5423. These three areas are liner failure criteria, melt superheat, and melt spreading phenomena. The staff has initiated necessary research in each of these areas as discussed below.

2.5.1.1 Liner Failure Criteria

The NUREG/CR–5423 analysis assumed that the liner would fail when the temperature of the liner reached the melting point of the liner steel. A member of the peer review group suggested that the steel liner could dissolve by chemical interactions with the melt prior to reaching the steel's melting temperature. Further examination of this issue indicated that the dissolution and the data base cited by the peer reviewer are not applicable to this issue. It was also noted by several peer reviewers that the strength of the steel liner would be greatly reduced at elevated temperatures. This, together with pressurization of the drywell following vessel failure might produce a creep-rupture failure of the containment liner prior to reaching its melting temperature. An NRC contractor is currently performing a structural analysis of the liner under these conditions to determine if creep-rupture failure will precede liner failure by melting.

2.5.1.2 Melt Superheat

A question was raised as to whether the NUREG/CR–5423 analysis of melt superheat was based on the best understanding of core-concrete interactions to date. Sandia National Laboratories performed the necessary confirmatory calculations using CORCON MOD3. The results indicate that the melt superheat duration in NUREG/CR–5423 is very conservative (i.e., CORCON MOD3 calculation of superheat duration is one-third of the values used in NUREG/CR–5423). Therefore, the thermal loading of the liner used in NUREG/CR–5423 is deemed appropriate.

2.5.1.3 Melt Spreading Phenomena

A concern was raised that in underwater volcanic lava flows, crusts have been observed to form at the lava-water interface, forming an annulus inside of a lava crust within which molten lava could flow without interacting with the surrounding water. The concern was that this phenomenon, should it occur, could insulate the molten core material flowing out of the pedestal region from the overlying water pool, and it also could preferentially channel the flow to the liner, eliminating the benefits of spreading. This is a complex problem not amenable to simple experimental resolution. The MELTSPREAD–1 code has been used to perform melt spreading sensitivity analyses to address this issue.

2.5.2 Future Plans

The staff plans to incorporate the results of the above-mentioned confirmatory activities into a NUREG report. A final peer review of the NUREG report will then be conducted.

2.6 Hydrogen Combustion and Research

2.6.1 Current Research Program

Appendix A.3 describes the state of knowledge of hydrogen combustion and transport and identifies processes that are not well understood. The NRC has recently sponsored two programs for experimental investigation of issues that heretofore have received little attention. While hydrogen research has considered diffusive flame behavior, flame acceleration, and detonation behavior, the research to date has been limited to tests involving hydrogen-air-steam mixtures under ambient or relatively low (100°C) temperature conditions. In the absence of reliable ignition devices, auto-ignition of jets and plumes released at high temperatures during a severe accident could result in the continuous burning of hydrogen as it is released into the containment. For premixtures of hydrogen-air-steam in containment at elevated temperatures (but below the auto-ignition temperature), flame acceleration and high-speed combustion may be more likely to lead to a transition to detonation. Thus, our current and future experimental research is directed at confirming the treatment of this behavior.

To address the effects of elevated temperatures on flame acceleration and detonation transition, the NRC initiated a high-temperature, high-speed hydrogen combustion program under a joint agreement (signed in June 1991) for a cooperative program with the Ministry of International Trade and Industry of Japan and the Nuclear Power Engineering Center. Under this agreement, a high-temperature high-speed hydrogen combustion research program, extending over 5 years, has been developed. Two combustion vessels will be used for this research program at Brookhaven National Laboratory. The largest of the two vessels will be approximately 30 cm in diameter and 20 m long. High-temperature hydrogen combustion mixtures will be used with a pre-ignition temperature as high as 700°K and an initial pressure of 1 atmosphere. The test gases that will be used will be mixtures of hydrogen, air, oxygen, nitrogen, steam, carbon dioxide, and carbon monoxide. The facility will be able to accommodate testing with and without venting. The smaller of the two vessels will be approximately 10 cm in diameter and 6.7 m long. This vessel will permit use of the apparatus for learning the effects of high temperature on the smoked foils and testing the instrumentation. It will also provide preliminary data to assess the Shepherd/ZND model for calculating cell size.

In the low-speed hydrogen combustion research program, the aspects of diffusion flames scalability and transient high-temperature combustion will be investigated. The results will be used to help resolve outstanding issues in severe accidents, i.e., hydrogen combustion aspects of DCH; high-temperature combustion phenomena, and detonation initiation by high-temperature steam-hydrogen-particle laden jets. These experiments will use a technique of combusting premixed rich and lean mixtures in separate vessels followed by deliberate mixing to initiate combustion. This will enable the creation of much higher temperatures (1000–3000°K) than could be readily obtained by conventional heat exchanger techniques. The natural heat transfer processes within the explosion vessels and the timing of combustion and mixing processes will be used to vary the temperatures in the gases. Dispersing metal or inert dusts within one vessel prior to combustion will allow the creation of particulate-laden atmospheres, which will simulate the process of high-pressure melt ejection during a DCH event. Pressure and temperature measurements and photographic recordings will be used to determine the nature of the combustion phenomena.

Evaluation of hydrogen transport in reactor containments remains a long-term research issue owing to a limited set of experimental data and thus limited validation of existing codes used in such analysis (i.e., CONTAIN, HMS). Current and near-term future activities include a modest program to document the development and assessment of the HMS code. The staff also expects to benefit from DOE-sponsored research to upgrade the HMS code such that the code may be readily applied to different configurations; the earlier version of HMS used in NRC-sponsored research, an analysis of plume mixing and diffusion flame combustion of Mark III reactors, required code modification for different geometries. At the completion of the DOE-related activity and our sponsored activity to document the code, the NRC will be better positioned to apply the code for reactor analysis and validation. With regard to the CONTAIN code that utilizes the traditional control volume technique for containment analysis, validation of the flow and mixing models against HDR project data is continuing with analysis of the international standard problem ISP–29.

As a result of our recent cooperative agreement with Japan in the area of hydrogen research, the NRC now has access to ongoing hydrogen mixing and distribution testing in the Tadotsu facility, a large-scale mixing and distribution test facility that simulates at one-fourth scale a PWR 4-loop reactor containment with compartmentalization. Additional data from hydrogen mixing and combustion testing in the Takasoga facility, which roughly simulates a RESAR SP–90 type design, will also become available during FY92–93. Test results from these facilities will provide a greatly expanded and improved data base for the validation of our analytical tools. It is anticipated that long-term confirmatory research will focus on this newly available data.

An important supplement to our hydrogen research program is that work being carried out under an arrangement with the Russian Academy of Sciences in collaboration with the I.V. Kurchatov Institute. Under this cooperative program, the HMS code, which has been used to predict hydrogen mixing and combustion, will be assessed and compared against Russian experimental data. Under this agreement, the NRC is also provided with the results of Kurchatov hydrogen deflagration test data and the results of a program to develop a scaling relationship for the critical conditions of turbulent jet initiation of a detonation. This research includes elements of experimentation, numerical simulation and theoretical analysis. An improved understanding of these phenomena will enable a more definitive evaluation of the detonation potential to be performed for both U.S. and Russian nuclear power plants.

2.7 Source Term

2.7.1 Status

At the present time, the NRC is pursuing several regulatory initiatives to incorporate insights from updated severe accident source terms. Updated source term insights arising from the technical update of TID–14844 are expected to be made available for voluntary use by existing licensees. A revision of 10 CFR Part 50 to incorporate updated source term and severe accident insights will then be undertaken, with a proposed rule for comment expected to be issued by early CY93. Although regulatory positions arising from updated source term insights remain to be developed, some preliminary implications can be seen at this time. It is clear that updated source term insights indicate the need for consideration of nuclides (e.g., cesium) in addition to iodine and the noble gases. In addition, revised insights on iodine chemistry call into question the need for high-efficiency charcoal absorbers (assuming that the pH is controlled, post accident). The iodine chemistry can, in turn, impact such important plant systems as fission product cleanup systems, control room habitability, and allowable containment leak rate. Finally, and most importantly, the above discussion and all recent risk studies have shown the importance of maintaining containment integrity under severe accident conditions in order to ensure low risk. This strongly suggests that the appearance of a severe accident source term within containment and challenges associated with such releases should be more closely linked with the temperatures, pressures, and containment loads, rather than an arbitrary linkage with a single sequence such as a large-break loss-of-coolant accident. The disconnect in present practice is not so much that temperatures and pressures came from only one sequence, but that they came from a sequence that was terminated without perceptible damage to the core, since the analysis that gave those temperatures and pressures was required to show that the peak clad temperature remained below 2200°F. The revised source term, is believed to be consistent with the source term expected to release into the containment resulted from core melt under low pressure severe accident sequences. It also provides the basis for the evaluation of the effectiveness of containment mitigation features under severe accident conditions. The revised source term is not intended to address by-pass accident source term, since the release is direct to the environment. However, analytical tool like VICTORIA has been developed specifically (section 2.4.4.) to address source term for by-pass accidents.

Although additional physics and chemistry research can be performed to reduce uncertainties in source term phenomena, it is important to consider the need for such research and the potential that this research could significantly improve our risk perspective on severe accidents. Hence, future research is oriented to assess the NRC severe accident codes and to address residual source term issues related to plant configurations during shutdown situations which would lead to the possibility of air ingress into the core either by natural circulation or from the residual heat removal system. The air will interact exothermically with the cladding remaining on the fuel, producing high temperature in the fuel. Vapor of ruthenium and molybdenum will be produced because of the strongly oxidized conditions. A test will be conducted at ORNL at these conditions in FY93 if the ongoing risk study sponsored by NRC indicated that accidents during shutdown conditions are major contributor to the core damage frequency. The NRC's participation in the PHEBUS–FP project is to obtain integral effects experimental data to validate NRC severe accident codes.

2.7.2 PHEBUS–FP

2.7.2.1 Objective

The objective of the PHEBUS-FP Project is to perform integral effects experiments in an in-pile test facility, under sufficient prototypical conditions, on the processes governing the transport, retention, and chemistry of fission products under LWR severe accident conditions. The processes to be investigated are those taking place in the core region, in the reactor coolant system, and in the containment building. The experiments will also study the degradation of high burnup fuel, typical of the later phases of the accident.

The Commissariat a l'Energie Atomique (CEA) of France and the Commission of the European Communities agreed to undertake the PHEBUS–FP Project in close collaboration, using the experimental facilities available at the Research Centre of Cadarache, France. NRC entered into an agreement with CEA; under this agreement fission product generation; transport and deposition generated in the PHEBUS–FP program will be made available to the NRC.

2.7.2.2 Facility Layout and Testing

PHEBUS–FP is a loop-type test reactor with a low-enriched driver core of 20 to 40 MW power, using fuel rod elements. Core cooling and moderation is achieved by demineralized light water, and light water and graphite are used as reflectors.

A cluster of 20 fuel rods, 1 m long, in a PWR configuration, is inserted in a test train and located in the central hole of the driver core of the PHEBUS–FP reactor.

Before a test, the test fuel from the BR3 reactor (a Belgium reactor that use 1 m-long fuel rods) is re-irradiated in the PHEBUS–FP in-pile section for 2 weeks using the existing pressurized water loop in order to generate a sufficient inventory of short- and medium-lived fission products. The loop is then slowly blown down with simultaneous reduction of the reactor power, with the in-pile section isolated from the loop. After these steps, testing may begin. During the test phase, the in-pile section is connected to a circuit and vessel that simulate the primary circuit and containment building of a PWR.

During the test phase, the in-pile fuel bundle is heated by fission power from the driver-core at a rate typical of a severe accident up to temperatures at which the fuel is damaged. The test bundle is pushed to conditions in which fission product release takes place, and control rods and structural materials are vaporized, producing sufficient quantities of aerosols. The fuel bundle will be damaged to the extent necessary not only to release fission products and aerosols, but also to study the mechanical behavior of the fuel during extensive degradation.

The released fission products and aerosols are swept by a flow of steam and H2 into the circuit that simulates the primary cooling system up to the point of pipe break. Then the flow enters a vessel that simulates the containment building.

2.7.2.3 Instrumentation

Instrumentation is used for process control, safety, and interpretation of the experimental results (scientific analysis). During the design of the PHEBUS–FP facility, a large effort was devoted to the instrumentation for scientific analysis. The success of the PHEBUS–FP experiments depends largely on the capability for measuring the parameters of interest.

The fuel bundle region will be instrumented with temperature, pressure, and flow sensors. The primary RCS circuit will be provided with two main instrumented sections: one in front of the large components (steam generator, pressurizer, etc.) and one just before entering the containment vessel. The main measurements of interest are (1) thermal-hydraulic conditions, (2) fluid composition, especially the fission product content, (3) aerosol concentration and granulometry, (4) deposits on the pipe wall, and (5) composition of the gas phases, including H_2. These will also be measured in the containment vessel at several positions in order to characterize their spatial distribution. In addition, the sump water will be monitored with respect to dose rate, isotope concentrations, etc. An extensive program for post-test examination, to back up and complement the on-line measurements, is being put in place at Cadarache with collaboration from several qualified laboratories of the European Community.

2.7.2.4 Test Matrix

The main objective of PHEBUS–FP is to obtain integral effects experimental data on fission product transport in the RCS and in the containment. This implies studies on retention and revaporization of fission products in the RCS. Revaporization could be enhanced either by decay heating or by a sudden steam spike in the circuit. The formation of gaseous species in the containment is also of a great importance. Studies in PHEBUS–FP are expected to yield information related to volatile iodine coming from radiolysis of the sump or organic iodine from paints in the containment.

The current test matrix consists of six tests. The intent of each test is to capture the key processes and phenomena associated with a particular severe accident sequence (e.g., V sequence) and not to simulate a sequence in details. The first test is scheduled for 1993, and subsequent tests are scheduled at yearly intervals.

2.7.3 Other Research Activities

In addition to the PHEBUS–FP project, the following research activities are planned.

- In FY92, complete the analysis of the ORNL VI-6 fission product release test and a technical report on the analysis and interpretation of the fission product release experiments performed at ORNL. Beyond FY92, additional tests (VI-7) will be conducted to investigate fission product release at high temperature for (1) fuel exposed to air under shutdown conditions or residual fuel remaining in the reactor core after vessel bottom melt-through, and (2) high burnup (>60 MWd/kg) and low-power-density ALWR fuel.

- In FY92, complete model development and documentation for the VICTORIA code, and validate the code against test data. In FY93, complete a peer review of the VICTORIA code. Further code improvement or development will depend on the outcome of the peer review. The code will also be used for PHEBUS–FP pre- and post-test analysis and for the International Standard Problem exercise on the FALCON project in the Winfrith Technology Center (WTC) U.K. Benchmarking of the CONTAIN code will also be carried out with the PHEBUS–FP data.

- A small experimental program will be continued at Battelle to provide the capability to utilize mass spectrometry under various oxidation states (reducing and oxidizing) at different temperatures (1000°K to 2700°K) to obtain chemical speciation and thermochemical data for a combination of materials (fuel, reactor structural materials, etc.).

Appendix B.1 provides a more detailed discussion on source term issues.

3 RESEARCH PLAN FOR ADVANCED LIGHT WATER REACTORS

3.1 Introduction

The role of severe accidents, despite their exceedingly low probability, has been recognized for consideration in advanced light water reactors (ALWR). In fact, the improved ALWR design features that reduce the likelihood and consequences of severe accidents, and the excellent safety record that continues to accumulate from operating plants, leaves only such exceedingly low-probability, high-consequence, events to be of concern. The objective of the ALWR severe accident research program is to examine the issues in depth and develop the necessary tools to address these issues. While it is recognized that the Westinghouse AP600 and the GE SBWR are sufficiently different from existing LWRs that different severe accident issues or variations on existing issues may exist, much of the phenomenological understanding of accident progression, containment loading, and source term analysis developed in relation to existing reactors will also apply to ALWRs. The readiness and applicability of the severe accident codes (e.g., SCDAP/RELAP5, MELCOR and CONTAIN) to ALWR plant-specific designs and phenomena must be addressed. Important phenomena addressed in accident progression analyses include timing of core melt and vessel breach, in-vessel hydrogen production, fission product transport and behavior, fuel-coolant interactions, release of fuel from the vessel, core-concrete interactions, hydrogen burns, and containment loading. We will assess the state of knowledge for phenomena that arise in ALWR designs, that have not had importance for severe accidents in current plants. Containment heat transfer, mixing, and hydrogen distribution in the passive containment are examples. As discussed in Section 2.4 of this report, NRC's severe accident codes will be modified to be capable of analyzing ALWR designs, and input decks will be developed specifically for the AP600 and SBWR designs.

This plan is based on an assessment of the applicability of current knowledge of severe accident phenomena to the ALWR designs. The plan identifies studies needed for specific ALWR designs, where the reviews for current reactors may not be sufficient. We do not foresee a need of new experimental research beyond what has been accomplished or is underway for current reactors.

3.2 In-Vessel Severe Accident Phenomena

3.2.1 Core Melt Progression and Reactor Vessel Lower Head Failure

3.2.1.1 Research Approach

Assess the methodologies used to develop or evaluate the effects of phenomena resulting from ALWR fuel and core designs on severe accidents, and the effect of design features (i.e., cavity flooding) to prevent lower head failure.

3.2.1.2 Discussion

The ALWR fuel design typically involves a lower power density than current reactor designs. For example, the AP600 plans to employ a power density of 73 to 79 kw/liter compared to 100 to 110 kw/liter for current PWRs. A lower power density results in an increase in the mass of the active core for a given power level, and correspondingly, the mass of zircaloy as well. This lower power density provides additional thermal margin during transients, and there is the potential to form a relatively thick protective oxide layer that could delay the onset of fuel melting. Nevertheless, if the core does melt, there is the potential for greater H_2 generation owing to the greater amount of zircaloy present. The ability to evaluate the effect of this additional non-condensable gas should be assessed, including but not limited to its effect on events involving steam generator tube rupture and events with assumed credit for natural circulation in the primary system. Although there is no reason to expect any new or additional phenomena to be important to in-vessel severe accident scenarios as a result of this fuel design, natural circulation effects may need to be modeled more accurately in our analytical codes such as MELCOR and SCDAP/RELAP5. Hence, adequate assessment and documentation of these codes are vital to provide the confidence in using them for ALWR analyses. Other than the lower power density, there appear to be no major differences in the core design between the present generation reactors and the ALWRs. However, some differences in the core structures and materials do exist (e.g., stainless steel reflector rods at core periphery, flow distribution grid, extended burnup fuels, and burnable poison rods in AP600). The severe accident code SCDAP/RELAP5 will be modified to account for these design differences. Our current understanding, with the knowledge that will be acquired under the revised severe accident research plan, is adequate to address core melt progression for ALWRs.

The AP600 design includes the capability to flood the reactor cavity during a postulated severe accident in order to cool the reactor vessel with an external water pool prior to core debris penetration of the vessel. The SBWR design also includes a system to flood the cavity. Analyses of potential lower head vessel failure for ALWR designs will need to consider the impact of accident management strategies for flooding the cavity.

3.2.2 Fission Product Transport and Behavior

The existing data base on fission product release should be applicable to the new low power density fuel used in advanced reactor designs. When the current assessment and development plan for the VICTORIA code is completed, the VICTORIA code will be adequate to address the fission product transport and behavior in the reactor coolant system for ALWRs. Since ALWRs rely heavily on natural circulation in the primary system for heat removal during a severe accident, the effects of natural circulation on fission product transport and behavior could be assessed with codes such as SCDAP/RELAP5 and MELCOR, provided the assessment discussed in Section 3.2.1 above revealed no major deficiency.

3.3 Ex-Vessel Severe Accident Phenomena

3.3.1 Research Approach

Assess the methodologies that are used to evaluate design features incorporated in ALWRs to mitigate the effects of core-concrete interactions, high-pressure melt ejection and DCH, and hydrogen combustion, including but not limited to methodologies that evaluate cavities designed to mitigate the effects of DCH and methodologies that evaluate cavity floors designed to provide sufficient area to allow coolable debris geometry.

3.3.2 Discussion

Within this context, the assessment is on the methodologies used to evaluate design features incorporated in the ALWRs to mitigate the effects of core-concrete interaction, high-pressure melt ejection and DCH, and hydrogen combustion. It includes, but is not limited to, methodologies that evaluate cavities designed to mitigate the effects of DCH and methodologies that evaluate cavity floors designed to provide sufficient area to allow a coolable geometry.

Another area for assessment is evaluating the new containment cooling concepts, including passive natural circulation cooling using air and containment dome cooling using water sprays.

3.3.2.1 Molten Core-Concrete Interactions

The available knowledge for predicting molten core-concrete interactions (MCCI) appears to be both applicable and adequate for ALWR accident analyses. The assessment of the criterion that core debris be spread over an area of 0.02 m^2/Mw thermal to assure coolability needs to be evaluated. This is part of the current severe accident research program (i.e., ACE consortium MACE tests at ANL and the WETCOR tests at SNL).

The SBWR cavity design was indicated to be similar to the ABWR design. In this design, molten debris in the lower cavity is allowed to spread to a shallow bed and then be quenched by water from the suppression pool. The applicability of the MACE data to the SBWR design should be evaluated. It is recognized that there are differences between the experiments and the SBWR design (e.g., debris depth and composition).

3.3.2.2 Direct Containment Heating

The AP600 has incorporated an automatic depressurization system for the RCS and eliminated instrument penetrations in the bottom head of the reactor pressure vessel. These design changes, however, do not completely eliminate the DCH threat because repressurization of the RCS during core degradation, particularly as a result of core debris-coolant interaction, is still possible. Current research on lower head failure mechanisms and DCH phenomena are expected to provide information that would allow the NRC to provide a preliminary assessment of the mitigative features of the ALWR designs. However, final conclusions on the retentive capabilities of a specific cavity design to preclude DCH may require further attention depending upon the credit required for such features.

For the SBWR, the low primary system pressure, the inerted containment atmosphere, and the passive depressurization system should significantly reduce the threat of a high-pressure melt ejection. Again, the methodology being developed in the current research program is adequate for application to SBWRs.

3.3.2.3 Hydrogen

3.3.2.3.1 *Research Approach.* The methodologies that are used to evaluate acceptability of the vendors' proposed hydrogen concentration criteria will be assessed, including evaluation of the maximum allowable H_2 concentration and the hydrogen control features proposed within the containments.

3.3.2.3.2 *Discussion.* For AP600, the current design criteria proposed by Westinghouse for hydrogen is to have a containment volume large enough that the bulk average hydrogen concentration could not exceed about 13%, assuming good mixing and reaction of 100% of the active cladding. Local accumulation of hydrogen will be controlled by dc-powered igniters. Westinghouse's choice of 13% as a maximum allowable hydrogen concentration is most likely based on existing correlations, which indicate that this is the highest practical concentration for which stable detonations could be ruled out. However, a concentration of 13% is too high to avoid combustion altogether. If we assume the containment volume is limited to about 100 °C by steam condensation on the containment dome, incomplete combustion can occur for hydrogen concentrations as low as 4%. Flame balls can grow, rise to the ceiling, and quench, leaving a stratified region with hot combustion products along the ceiling and unburned hydrogen in air below. Rapid, almost complete, combustion can occur for hydrogen levels above about 8%, which would leave little unburned hydrogen. For a containment atmosphere containing some water vapor, stable detonations are highly unlikely for 13% hydrogen and below. Accelerated flames with damaging overpressures could occur inside this containment (~10 percent or more H_2) in cluttered regions that promote flame acceleration, but when the flame transits into the region of open containment volume, it may decelerate again. Some residual shocks may strike the containment, but with far less energy than for a full-containment detonation. The NRC position taken on ALWRs is that the containment design should limit the hydrogen concentration to no greater than 10%, and that containment-wide hydrogen control should be provided to preclude the formation of local detonable mixtures and lessen the accumulation of hydrogen on a global basis.

Although the above criterion goes a long way toward mitigating the hydrogen problem, two problems remain to be addressed. They are (1) loads from low-pressure combustion, and (2) local detonation. It appears that low-pressure combustion does not generate overpressures adequate to threaten the containment. With 13% hydrogen, overpressures of 4 to 7 bars may be expected, depending on the initial temperature and steam concentration. Presumably, the containment can withstand this overpressure.

To address the local detonation problem, Westinghouse is depending on dc-powered hydrogen igniters to burn the hydrogen early. If the igniters are successful at triggering combustion before the hydrogen concentration builds up to a locally detonable level, even local detonations can be avoided. This depends on the number and reliability of the igniters as well as the strategic placement of these igniters. While the reliability of the igniters could be addressed, the number and placement of them requires the prediction of hydrogen concentration, stratification, and distribution in the containment. Because the passive designs lack active mixing capability, assessment of containment mixing becomes a more challenging task. The existing codes used for containment analysis utilize a control volume approach, and the constituents within each volume are assumed to be well mixed. This may be inadequate to address the issue on hand. Special flow field models, e.g., HMS may be used to address this issue.

Current design criteria proposed by GE would inert the containment region with nitrogen. Thus for the SBWR, all the hydrogen generation issues appear to have been dealt with by assuming that inerting the containment environment will take care of any problems. One unresolved issue for SBWR is the treatment of noncondensable gases in the isolation condenser and the passive containment cooling system (PCCS). Additional work may be found necessary to assess the hydrogen effect on the isolation condenser and PCCS performance.

3.3.2.4 Containment Cooling

3.3.2.4.1 *Research Approach.* The methodologies that are used to evaluate new containment cooling concepts will be assessed, including natural circulation cooling using air and containment dome cooling using water sprays.

3.3.2.4.2 *Discussion.* For certain classes of accidents, the AP600 is expected to exhibit performance characteristics that are different from existing PWRs. These phenomena include,

(1) Natural circulation within the containment,

(2) Hydrogen mixing and development of high local concentrations,

(3) Natural circulation cooling of the containment, and

(4) Water cooling of the external shell of the containment.

For the external shell, flow patterns and correct heat transfer correlations are needed for water cooling of the external shell. Inside the containment, the modeling of natural circulation and mixing processes over large volumes will pose a challenge to existing codes. For the SBWR, the behavior of the suppression pool can be readily modeled. Like the AP600, for certain accidents the SBWR may pose new challenges in modeling natural circulation and mixing processes for existing containment analysis codes.

Natural circulation flow and related mixing processes are key issues for both the AP600 and the SBWR designs. For the AP600, this will primarily be a containment issue. For the SBWR, it will be an issue for the passive core cooling system and the containment long-term heat removal system, since they are closely related.

Existing NRC severe accident codes CONTAIN and MELCOR are control volume (i.e., lumped-parameter) codes that do not take into account finite gas velocities or momentum convection in the control volumes, or cells. Momentum is considered in the junctions, or flow paths, between cells, but this momentum is considered to be completely dissipated in the downstream cell. The atmospheres in the control volumes are consequently assumed to be stagnant. Such modeling is appropriate for large control volumes connected by flow paths whose cross-sectional area is small compared to the cross-sectional area of the control volume itself. In certain situations, however, the neglect of velocities and momentum convection within control volumes is not justified, and the distribution of flows can be adversely affected by this assumption.

Since the ability of the control volume codes to calculate the distribution (i.e., stratification) of hydrogen is limited in some cases, and since the principal NRC severe accident analysis codes are all control volume codes, a better understanding of the limitations of these codes and possibly improved modeling capabilities based on that understanding could be important with respect to resolving severe accident issues.

It is possible to modify lumped-parameters codes by implementing velocities within the control volumes. The COMMIX code (see section 2.4.4) is being modified to evaluate ALWR designs for mixing characteristics.

In addition to the general issues of mixing and natural circulation, the ability to evaluate ALWR designs will strongly depend on understanding natural convection and steam condensation processes, particularly in the presence of large quantities of noncondensables. The vendors have proposed testing to address the question.

3.3.2.5 Fission Product Transport and Behavior

The existing knowledge on source terms is applicable to the AP600 and the SBWR. Clearly, source term predictions will be driven by containment behavior, particularly aerosol transport, deposition, and suppression pool scrubbing. The CONTAIN code will be adequate to address the source term in the containment. Given that the issues related to natural circulation and mixing processes are adequately addressed, the existing methods for source term analysis are adequate.

3.4 Summary

Westinghouse and GE have not presented the NRC with details of their severe accident research program or the extent to which vendors' research will satisfactorily address issues related to severe accidents. The NRC is planning to hold meetings with the vendors to understand their research programs. We will then develop a program to resolve all severe accident licensing issues. In consultation with NRR, RES will identify the research program that will be carried out by NRC. We do not foresee a need for new severe accident phenomena experimental programs. However, NRC will develop analytical capabilities to independently assess the vendors' analysis. To accomplish this, we plan to:

1. Complete the research delineated in this updated report.
2. Perform assessment or modification and documentation for NRC codes (MELCOR, SCDAP/RELAP5, CONTAIN, COMMIX, VICTORIA) so that they could be used to perform analyses to audit licensee calculations.

APPENDIX A
SEVERE ACCIDENT ISSUES, STATUS, AND PROGRESS TO DATE

Introduction

During the past few years, the severe accident research program has focused on generating information that can narrow the broad range of uncertainties that have been identified (e.g., NUREG–0956, "Reassessment of the Technical Bases for Estimating Source Terms, 1986," and the Kouts report, NUREG/CR–4883, "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants, 1987") as limiting the ability to accurately calculate source terms and to provide an improved degree of assurance in estimating risks from severe accidents. The original eight areas of uncertainty in NUREG–0956 were:

1. Natural circulation in the reactor coolant system,
2. Core melt progression and hydrogen generation,
3. Steam explosions,
4. High pressure melt ejection,
5. Core-concrete interactions,
6. Hydrogen combustion,
7. Iodine chemical form, and
8. Fission product revaporization.

In the "Revised Severe Accident Research Program Plan," NUREG-1365, published in August 1989, similar areas of research were combined (e.g., iodine chemical form and fission product reevaporization), whereas other areas that are either important to accident scenarios that might lead to early containment failure (e.g., Mark I containment shell melt-through and direct containment heating) or are important to assessment of accident management strategies (e.g., adding water to degraded core) were presented separately, and research needs were identified. Also in NUREG-1365, research plans addressing the issue of scaling of severe accident experiments were presented.

Appendices A.1–A.5 provide summaries of the severe accident issues listed in Table A.1, along with their current status, progress to date, and future plans. Appendices B.1–B.3 provide the technical bases for closure of the issues related to source terms, core-concrete interactions, and scaling methodology. Note that Items 4 and 8 are not "issues" in the same sense as the other issues that represent technical phenomena associated with severe accidents; however, they are major programmatic areas in the severe accident research plan, and therefore, are listed and discussed as separate items.

Table A.1 Severe Accident Research Program Issues

	Appendix
1. Mark I Containment Shell Melt-through	A.1
2. Core Melt Progression and Hydrogen Generation	A.2
3. Hydrogen Transport and Combustion	A.3
4. TMI–2 Vessel Inspection Program	A.4
5. Fuel-Coolant Interactions and Debris Coolability	A.5
6. Source Terms	B.1
7. Core-Concrete Interactions	B.2
8. Severe Accident Scaling Methodology	B.3

A.1 Mark I Containment Shell Melt-through (Liner Failure)

A.1.1 Background

An accident sequence leading to early containment failure has been postulated for BWR Mark I containments. This sequence involves the direct attack of the containment steel liner by molten core material following vessel failure. In SECY–89–017, "Mark I Containment Performance Improvement Program," the staff addressed the issue of severe accident challenges to the Mark I containment and proposed a balanced approach utilizing accident management and mitigation as the optimum way to reduce overall risk in BWR plants with these containments.

SECY–89–017 stated, "there is a growing consensus that water in the containment (from an alternate supply to the drywell sprays) may help mitigate risk by fission product scrubbing and possibly by preventing or delaying containment shell melt by core debris. Research is continuing in order to confirm and help quantify these initial conclusions."

NRC research over the past several years has addressed key phenomena associated with the liner meltthrough issue, such as melt conditions at the time of vessel failure; melt spreading characteristics; thermal-hydraulic characteristics of molten core-concrete interactions both with and without an overlying water pool; heat transfer characteristics at the interface of the molten core, overlying water pool, and liner; and fission product attenuation in the presence of an overlying water pool.

Integration of the research information derived from these programs into an assessment of the conditional probability of liner failure both with and without an overlying water pool in the drywell, given a core melt accident that proceeds to vessel failure, was completed. A description of this methodology and its conclusion is provided in NUREG/CR-5423. In summary, by developing probability distributions for important parameters that factor into the analysis from data (where available), computer analyses, or other insights, developing causal relationships between phenomena, and convoluting these distribution functions and causal relationships, the authors of NUREG/CR-5423 obtained estimates of the likelihood of liner failure both with and without a water pool overlying the molten corium in the drywell. When water was assumed to overlie the molten core material as it spreads on the drywell floor toward the containment liner, it was concluded that the liner failure would be physically unreasonable. In the absence of water, however, the same conservative approach led to the conclusion that failure would be certain.

A.1.2 Status

NUREG/CR-5423 has been subjected to an extensive and thorough peer review. Nineteen experts knowledgeable in the subject matter were asked to review the analyses. In addition to providing detailed written comments, the peer reviewers were given the opportunity to discuss their comments with the authors in a public workshop that was also attended by members of the Advisory Committee on Reactor Safeguards and the Nuclear Safety Research Review Committee.

The main conclusions of the peer review and the workshop were:

1 The methodology employed in NUREG/CR-5423 was considered basically sound—no major deficiencies or problems were identified that would invalidate the results.

2 The sensitivity study performed in NUREG/CR-5423 showed there was no single process or parameter that had a controlling influence on the overall failure probability.

3 There was also a general consensus by the peer review group that three areas warranted additional research to confirm the appropriateness of the analysis in NUREG/CR-5423. These three areas are liner failure criteria, melt superheat, and melt spreading phenomena. The staff's plan to conduct the necessary research in each of these areas is discussed in Section 2.5.

For each of the three areas, a small (three or four) group of experts was convened, many of whom served on the larger peer review panel, to assist the NRC in addressing the concerns. In addition, in order to ensure that the initial melt quantity and composition were appropriate, a meeting was held to assess the adequacy of the report's quantification of the initial melt conditions at vessel failure. Additional analyses were performed using the APRIL-MOD3 code in which the heat-up, collapse, and melting of the BWR steam separator or dryer were explicitly modeled. The basic conclusion was that the boundary conditions used in NUREG/CR-5423 seem to be adequate.

NUREG/CR-5423 identified three scenarios that could challenge the containment integrity. Scenario I is based on an initial sudden massive core slump (50% of the core) leading to localized lower head failure. Scenario II is based on initially quenched debris and a subsequent localized lower head failure owing to water depletion and remelting of the debris in the lower plenum. Scenario III is similar to Scenario II, except that the debris heats up the lower head uniformly, resulting in weakening and eventual creep failure. Detailed analyses were performed for Scenarios I and II. For the NUREG/CR-5423 peer review, the peer reviewers were informed that the evaluation of Scenario III would await the completion of the NRC lower head failure analysis program at Idaho National Engineering Laboratory. This program addresses the likelihood of creep rupture of the lower head as well as other potential failure modes of the lower head. The INEL program is now complete, and draft NUREG/CR-5642 has been issued. The preliminary results indicate that for a depressurized reactor vessel, global vessel failure is not likely to occur. Therefore, the NRC is not planning to perform any analyses to qualify the potential for containment shell meltthrough resulting from accidents that follow Scenario III.

While there are residual issues related to the uncertainty of analysis in NUREG/CR-5423 in predicting liner integrity in the presence of water, it is generally recognized that the presence of water will sharply attenuate the magnitude of aerosol production and radionuclide release. Aerosol production is affected by water because aerosol-laden gases produced by the core-debris attack on concrete must sparge through the water. A recently completed NRC-sponsored study investigated the detailed processes involved in aerosol trapping by water pools and developed a simple model of water effects on aerosol generation during core-debris interaction with concrete. The model provides the probability distribution functions for decontamination factor (a measure of aerosol trapping) based on the uncertainty analysis using the Monte Carlo simulation technique. Bounding case calculations using this simple model indicate that water has a profound mitigative effect on aerosol production and radionuclide release.

A.2 Core Melt Progression

A great deal of information has been obtained on the processes involved in the early phase of melt progression in core uncovery accidents that extends through core degradation and metallic (but not ceramic) material melting and relocation. This information has come from integral tests in the PBF, ACRR, NRU, NSRR, and Phebus test reactors, from the LOFT FP-2 test, from tests in the CORA ex-reactor fuel-damage test facility, and from separate-effects experiments on significant phenomena. These tests have provided core degradation information on fuel failure, Zircaloy oxidation by steam with attendant hydrogen generation, Zircaloy-clad melting and relocation, and the effects of PWR Ag–In–Cd control rods, BWR B_4C control blades, high burnup fuels, and the reflooding of severely damaged cores.Most of the available information on late-phase melt progression has come from the post accident examination of the TMI–2 reactor. Despite the core reflooding that successfully terminated the TMI–2 accident, the general late-phase melt progression phenomenology of that accident, although not the detailed behavior, appears to be applicable to unrecovered as well as to recovered accidents and possibly to some BWR accidents as well. The integral experiments that have provided most of the current information base on melt progression and an outline of the information obtained from these experiments are given in Table A.2.1.

The results of these integral tests and the TMI–2 core examination have provided a consistent picture of melt progression. This is illustrated in the end-state configuration of the TMI–2 core, which is shown schematically in Figure A.2.1. Figure A.2.1 shows the development of a debris-supporting metallic blockage above the water level in the lower portion of the core during coolant boildown. This blockage is produced by the relocation and freezing of metallic melt, mostly from unoxidized Zircaloy cladding. Fission-product decay heating produces a growing pool of mostly ceramic UO_2 fuel and oxidized zircaloy above the metallic core blockage. The growing pool melts downward and radially outward through the supporting metallic blockage and the secondary ceramic crust that surrounds the pool. During pool growth, the crust system melts and reforms, relocating downward and outward. It appears that meltthrough occurs when the crusts do not reform to continue containment of the melt pool. At TMI–2 with a reflooded core, pool melt-through was out the side of the core. The mass and other characteristics of the ceramic melt that drains from the core into the lower plenum in blocked-core accident sequences are determined by the threshold and the location of the meltthrough of the supporting crust. The TMI–2 configuration illustrates essentially the general melt progression phenomenology thought to apply to both recovered and unrecovered blocked-core accident sequences in PWRs, and also to any BWR accidents that have blocked core sequences. Differences in specific phenomenological behavior from TMI–2, for unrecovered accidents are possible, however, particularly with regard to the question of sideways or downward meltthrough from the core.

The similarity of the results of the many integral tests on core degradation and early phase (metallic) melt progression show that the overall behavior in this regime is reproducible and is not strongly stochiastic in nature. Such assurance is not available for the late phase melt progression, however, with Figure A.2.1 the TMI–2 core examination providing nearly all the available information. The processes of melt pool growth in a particular ceramic debris and melt through of the supporting crust system, however, do not appear to be stochastic processes.

Metallic melt relocation leaves behind free-standing cracked UO_2 fuel pellets and ZrO_2 oxidized cladding shards that have melting points, including eutectics, in the range from 2800°K to 3100°K. During late-phase melt progression in unrecovered blocked core accidents and in very severe recovered blocked core accidents such as TMI–2, a mostly ceramic melt pool forms and grows in the mostly ceramic debris bed. Thus, the metallic and the ceramic debris with melting points that differ by 600°K or more become separated in space, and, as the TMI–2 core examination shows, they behave quite differently as core heat-up and melt progression continue. The metallic and the ceramic materials need to be treated separately in accident analysis codes if melt progression is to be represented realistically. This is not done in the older simplified codes that treat the core melt as a single fictitious "corium" fluid with a unique (high metallic) composition and a relatively low melting point (usually 2550°K).

In BWR accidents with automatic depressurization, primary system blowdown lowers the water level below the core and the BWR core plate. Under these conditions, heat-up occurs in a dry core with very low steam flow. The contribution of zirconium oxidation to the core heat-up is small under these conditions, and a large fraction of the Zircaloy cladding melts over a short period of time. It has been hypothesized that, under these conditions, the metallic melt (including eutectic alloys) drains from the core and the BWR core plate into the water-filled lower plenum instead of freezing to form a blocked core similar to that at TMI–2.

The reflooding to the top of the core in the TMI–2 accident, however, did not prevent continued core melting because the water did not penetrate into the hot molten pool region of the core. The accident was only successfully terminated when hot ceramic core melt that constitute

Table A.2.1 Sources of Current Integral Experimental Information on Melt Progression

Experiments	Key Information
PBF Severe Fuel Damage Tests SFD-ST, 1–1, 1–3, 1–4	Integral information base on core degradation and melt progression
	Control rod and high burnup fuel effects
NRU Full-Length Tests FLHT 1, 2, 4, 5	Data on length effects and the absence of a cut-off to hydrogen generation
ACRR Damaged Fuel Tests DF–1, –2, –3, –4	Separate-effects data on core degradation and melt progression
	Basic BWR information from DF–4
CORA Ex-reactor Tests and Related Experiments	Basic information base on material-interaction effects and metallic melt relocation
	Core degradation information for BWR and PWR geometries, including reflood effects, using electrically heated, simulated fuel rod bundles of up to 57 rods
Phebus SFD Tests	Core degradation and early phase melt progression phenomena
NSRR Reactivity Initiated Accident (RIA) Tests	Fuel failure thresholds in reactivity initiated accidents
LOFT FP–2 Large-Bundle (101-rod) Test	Unique results on metallic melt relocation and the absence of a cutoff to hydrogen generation with a large flow-bypass area
	Significant results on the effects of reflood on core degradation and hydrogen generation
	Unique test results with fission- product decay heating
ACRR Late Phase (Ceramic Melt) Tests DC–1, DC–2, MP–1	Dry UO_2 debris-bed thermal characterization and melting behavior
	Dynamics of pool and crust growth in particulate ceramic debris beds
TMI–2 Core Examination	Major source of significant information on late-phase melt progression
	Results applicable to basic phenomenology for both recovered & unrecovered accidents

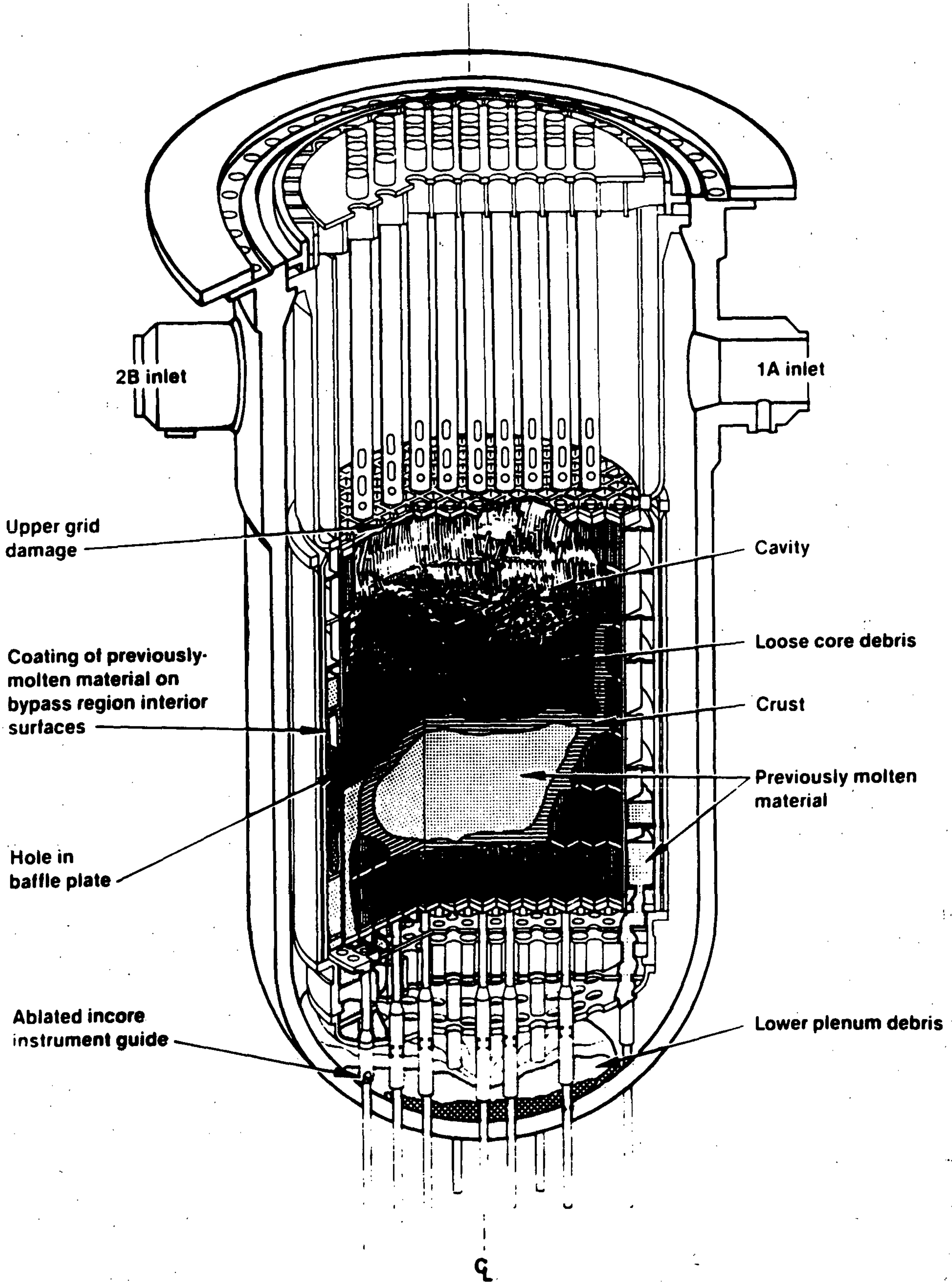

Figure A.2.1 TMI–2 Core End-State Configuration

Table A.2.2 Core Melt Progression: Status of Current Understanding

- Early (Metallic Melt) Phase, Reasonably Well Understood Phenomena:
  - Clad ballooning
  - Intact-core-geometry oxidation heating and hydrogen generation.
  - UO_2 liquefaction (dissolution) by molten Zircaloy.
  - Eutectic material interactions among UO_2, ZrO_2, Zr, and control materials and their oxides. Rate limitations are less well understood.
  - Opening up the compartmentalized BWR core early in a BWR accident by the eutectic interaction of control blade material with zircaloy channel box walls.
  - Molten zircaloy relocation is a noncoherent, noncoplanar, rivulet-flow process, not a film flow process. The melt forms an incomplete blockage that does not cut off steam flow and hydrogen generation.
- Late (Ceramic Melt) Phase, General Understanding:
  - Information primarily from the TMI–2 core examination.
  - Results are also generally applicable to PWR unrecovered accidents.
  - Ceramic melt pool growth and meltthrough from a blocked core.
  - Reflooding probably stopped downward pool and crust relocation to give side melt-through at TMI–2.
  - Limited melt mass released from core (20% at TMI–2).
  - Low metal content in ceramic melt pool.
  - During Reflood: much hydrogen generation and strong heating of uncovered core from Zircaloy oxidation by reflood steam (LOFT FP–2 and CORA), is not well understood.

about 20% of the core mass drained from the melt pool into the water-filled lower plenum (an additional heat sink). The melt was cooled by the lower plenum water and did not fail the vessel lower head. The core reflooding, however, did stop the previous downward relocations of the metallic crust (by melting and refreezing) and may have been the cause of the melt-through of the melt pool out the side, rather than the bottom of the core. This resulted in the drainage of only about half the melt pool rather than the entire pool.

A major finding from all the integral tests and also from the TMI–2 core examination is that the unoxidized Zircaloy, the control rod materials, and their eutectics melt before or during the rapid temperature transient from steam oxidation of the core zircaloy and at temperatures ranging from as low as 1200°K for the eutectics up to 2200°K for the Zircaloy. Molten metal, which includes some dissolved UO_2 fuel, relocates downward by gravity to refreeze and form a porous partial core blockage, at least in PWR accidents with water in the bottom of the core.

The question of metallic melt drainage or core blockage is a major branch point for in-vessel core melt progression, and it has a large effect upon the characteristics of the melt released from the core into the lower plenum. In the core blockage case, a large mass of mostly ceramic melt at about 300°K drains rapidly into the water-filled lower plenum, as happened at TMI–2. In the drainage case, layers of quenched melt are formed under the lower plenum water in the order of their time of melting and drainage from the core, with the low- melting metals at the bottom and the ceramics at the top. These differences also have a major effect on the vessel failure process and on the characteristics of the melt released into the containment upon vessel failure. In the drainage case, the released melt has a lower temperature and a high metal content.

In the TMI–2 accident, the melt released on melt-through from the blocked core into the water-filled lower plenum by the ceramic melt pool contained only about 20% of the core mass and had a very low metal content. The low metal content is important because metal provides the potential for oxidation heating and hydrogen generation after vessel melt-through. A more quantitative understanding of the key processes involved in determining the melt-through threshold and location of failure are needed, however, in order to generalize the application of these relatively benign results. Acquiring this information is a major objective of the current melt progression research. A second major objective is determining whether metallic, and later ceramic, melt drainage from the core without core blockage may occur in BWR accidents in which the blowdown from automatic depressurization (ADS) lowers the water level below the core, and core heat up occurs in a "dry core" with very low steam flow.

A summary of the current state of phenomenological understanding of melt progression is given in Table A.2.2. There is reasonable understanding of the significant phenomena involved in fuel damage and early phase (metallic melts) melt progression, except for the processes of metallic melt relocation and blockage formation. For the late phase (ceramic melts), there is, in contrast, only a general understanding that is based mostly on the TMI-2 core examination and analysis. Significant information on reflood effects has also been obtained.

During a severe accident, hydrogen and heat are generated during the exothermic chemical reaction (oxidation) of steam with core Zircaloy (and possibly some stainless steel). The integral severe fuel damage (SFD) tests that have been performed have provided data on Zircaloy oxidation and hydrogen generation during severe accidents in addition to the data on core degradation and melt progression that were discussed earlier. These tests were conducted over a wide range of conditions in both BWR and PWR core geometries. In addition, separate-effects experiments on Zircaloy oxidation in steam have furnished basic reaction rate data and correlations. The rates of Zircaloy oxidation and hydrogen generation are well known as long as intact core geometry is maintained. At lower temperatures (below about 1700°K), the reaction rate is limited by oxygen diffusion through the growing protective ZrO_2 layer formed by cladding oxidation, and the rate is limited by the availability of steam and Zircaloy at higher temperatures. The rate limit from diffusion is well described by a parabolic rate law with an Arrhenius (exponential) temperature dependence. Correlations of the experimental data on oxidation rates are incorporated into the MELCOR and SCDAP/RELAP5 codes as well as into other severe accident codes.

Oxidation continues as the intact geometry is lost through relocation of molten unoxidized core Zircaloy downward to cooler regions of the core, but the oxidation rate becomes less well known. This is because of a lack of knowledge of the actual geometry, the thickness of new protective oxide layers, and the locally available steam flow. In all the integral severe fuel damage tests, substantial oxidation rates continued after the start of molten zircaloy relocation so long as supplies of steam and unoxidized high temperature Zircaloy were available. In tests with BWR core geometry, eutectic interactions among the control blade melt of stainless steel and B_4C and the Zircaloy channel box walls failed the walls and opened up the geometry to the cross-flow of steam and continuing hydrogen generation as in a PWR. During late-phase melt progression in unrecovered accidents, oxidation and hydrogen generation cannot be significant because of the low metal content and the compacted geometry of the hotter region of the core around the growing ceramic melt pool.

The severely damaged fuel bundle was reflooded in the LOFT FP-2 test, and most of the oxidation and hydrogen generation in the test was produced by steam during reflooding. A quantitative understanding of the rates involved, however, does not yet exist. When Zircaloy-containing melt drops into lower plenum water, there can also be a small amount of oxidation and hydrogen generation from the unoxidized Zircaloy in the melt, particularly if a steam explosion occurs.

As core melt material relocates into the lower head of the reactor vessel, the major concern of severe accident analysis becomes the mode and timing of lower head failure. The research program in this area includes the analysis and examination of samples from the TMI-2 lower head. Failure mode analyses have been conducted to examine failure by the ejection of a vessel penetration, failure of a penetration outside the vessel shell, and global and local creep- rupture failure of the vessel shell. The failure mechanisms have been evaluated for debris and thermal-hydraulic conditions estimated for current BWR and PWR designs. Results of these analyses are reported in draft NUREG/CR-5642, "Light Water Reactor Lower Head Failure Analysis," that was issued in December 1991.

Typical severe accident scenarios have been used for each of the reactor types to develop lower head failure maps. The failure maps show, as a function of system pressure and vessel inner wall temperature, the mechanisms by which a vessel is most likely to fail. The studies have shown that important parameters include not only the system pressure and vessel temperature, but also the effective flow area and wall thickness of the penetrations and the size of the annular gap between a penetration tube and the vessel wall.

Under conditions that result in low heat-up of the vessel wall, the most likely mode of failure is ejection of a penetration because of failure of the vessel seal weld. In this case, the friction between the penetration tube and vessel opening is so low that tube ejection can occur without tube rupture or damage. As the temperature of the vessel wall rises sufficiently to restrain the penetration tube by friction, failure may occur by melt material penetration of the vessel through the tube, with subsequent melt-through or rupture of the tube outside the vessel boundary. Two modes of melt penetration have been studied for this case: conduction-limited penetration that is governed by a freezing annular shell and bulk freezing penetration. Conduction-limited penetration has been found to give the greater melt penetration in all cases. Molten debris was also found to be more likely to penetrate through a tube with a large effective diameter, such as a BWR instrument tube or a BWR drain nozzle, rather than through smaller diameter PWR instrument tubes.

Failure of the vessel lower head by creep rupture may be possible when the system pressure and vessel temperature are sufficiently high. The failure mode can be either a global failure of the hemispherical head, circumferentially around the vessel below the debris surface, or local bulging and membrane rupture of the shell. Vessel failure analyses to date have used assumed debris conditions to drive simple thermal analyses. Dimensionless groups derived from these analyses may now be used in further detailed analyses with severe accident codes such as SCDAP/RELAP5. An analysis for the local bulging case has also been developed in which localized vessel wall heating may occur as a result of jet impingement of molten core material on the vessel wall.

No additional research is planned on hydrogen generation during in-vessel melt progression. However, analyses that are under way of oxidation and hydrogen generation from reflooding in the LOFT FP–2 test will be completed.

Plans for lower head failure analysis include a peer review in FY92 of draft NUREG/CR–5642. The results of the peer review will be incorporated into a final NUREG report.

There are some significant uncertainties related to core melt progression phenomena, primarily in the late (ceramic melt) phase. Future plans to address these uncertainties are discussed in Section 2.2.

A.3 Hydrogen Combustion and Transport

A.3.1 Background

The safety significance of hydrogen combustion during a severe accident for non- inerted containments is that the concomitant energy release manifested as pressurization and heating of the containment atmosphere could pose a threat to containment integrity or to the survival or functioning of essential safety equipment. When hydrogen combustion alone is insufficient to threaten containment integrity, combustion may still represent a significant contribution to containment loadings when considered conjunctively with direct containment heating or steam pressurization. Hydrogen combustion can also impact the source term by altering fission product chemistry and resuspending fission products in aerosol form. Hydrogen transport is a safety issue for operating reactors primarily insofar as the mixing of hydrogen determines the nature of subsequent combustion. In the event hydrogen released into the containment during a severe accident accumulates without igniting but mixes rapidly throughout the entire volume, the global concentrations in most instances will remain below the limits for detonation. If mixing does not occur because of stratification or pocketing in enclosed areas, those richer mixtures that occur, at least locally, present a greater likelihood for flame acceleration and detonation. For advanced reactor designs without active mixing systems that rely on passive containment heat removal from the containment atmosphere to the containment shell (AP600) or to an external isolation condenser (SBWR), the transport of hydrogen within the containment may influence the overall heat removal capability of those related safety features.

A.3.2 Status

Research conducted world-wide over the past 12 years has extensively investigated a number of issues related to hydrogen combustion and transport during severe reactor accidents. Much of the work, performed to experimentally investigate the design and evaluation basis for reactor containment performance, focused on global deflagrations of premixed volumes of hydrogen, air and steam. Since containment analysis generally presumed global deflagrations were the mode of combustion (this assumption could be contrived to produce appropriately conservative loadings), research results were instrumental in establishing the dominant parameters (flame speed and combustion completeness) influencing the peak pressure from volumetric deflagrations. Diffusive burning of hydrogen has also been the subject of experimental research conducted by both by the NRC and the industry. In a cooperative hydrogen research program conducted at the DOE Nevada Test Site (NTS), hydrogen and hydrogen-steam jets and plumes in ratios intended to represent severe accident blowdowns were ignited by thermal igniters to study diffusive burning behavior. Diffusion flame research has also been carried out at Sandia National Laboratories and at other research facilities, including the large scale facility at Factory Mutual Research Corporation, used to investigate hydrogen mixing and combustion in a Mark III containment.

Recognizing that combustion modes include supersonic as well as subsonic flame propagation, the NRC has also sponsored considerable research on the detonation of hydrogen-air and hydrogen-air-steam mixtures. The international reactor safety research community has also contributed to the data base on the detonability of various gaseous mixtures. This research clearly identified the sensitivity of scale in establishing the limits for detonability of mixtures as well as providing insights to the mechanisms for flame acceleration and transition to detonation. The range of detonable concentrations for hydrogen- air mixtures has been shown experimentally to be much wider than the classic limits of approximately 18 to 60% established in small-scale testing at ambient temperatures. The range of detonable concentrations has been shown experimentally in our research to be as wide as 11.6 to 74.9% hydrogen at 20°C and 9.4 to 76.9% hydrogen at 100°C with the limits depending on scale, geometry, and temperature. The view has been that steam greatly reduced the mixture's detonability, but there is analytical evidence that increasing temperature may make steam mitigation less efficient. In conjunction with hydrogen combustion research, the research community has also experimentally explored the issue of hydrogen transport and mixing. While the NRC has not exclusively sponsored significant experimental research on hydrogen mixing, our joint research agreement with Germany has provided mixing test data in the complex geometries of the Battelle-Frankfurt and HDR test facilities. Industry data from the HEDL facility and the FMRC one-fourth scale facility has supplemented that data; although this data was specifically related to ice condenser and Mark III containments under certain accident conditions. To complement the experimental or phenomenological research, the NRC has sponsored the development and application of computer codes for hydrogen mixing and combustion analysis, most notably the HECTR and HMS codes. The HECTR code, which employs control volume modeling of the containment, and the HMS code, a finite difference code, were developed to provide varying levels of resolution of the containment volume for mixing analysis. HECTR combustion modeling has been subsumed into the more general CONTAIN and MELCOR codes.

A.4 TMI-2 Vessel Investigation Project

A.4.1 Background

Most current knowledge of in-vessel severe accident behavior has come from experiments such as the series of severe fuel damage tests performed in the Power Burst Facility, ACRR, NRU, LOFT FP-2, Phebus test reactor, CORA ex-reactor experiments, and from the extensive post accident core examination of the Three Mile Island Unit 2 (TMI-2) reactor, performed by the Department of Energy (DOE). In 1988, the NRC, in cooperation with 10 foreign countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency (NEA), undertook a follow-on program to the DOE TMI-2 examinations. The objectives of this program, called the TMI-2 Vessel Investigation Project (VIP), are to (1) investigate the condition and properties of materials extracted from the lower head of the TMI-2 reactor pressure vessel, (2) determine the extent of damage to the lower head by chemical and thermal attack, and (3) determine the margin of structural integrity that remained in the pressure vessel.

A.4.2 Status

Under the VIP program, 15 reactor vessel steel specimens, 14 incore nozzles, and 2 incore guide tubes were successfully extracted from the lower head over a 30-day period ending March 1, 1990. The vessel steel samples then were decontaminated, sectioned, and distributed to the United States and seven other participating countries for mechanical and metallographic examinations. Also, the nozzles and guide tubes were cut and distributed for examination to INEL, ANL, and the CEA in Saclay, France.

Since the extraction of the test specimens in 1990, substantial metallographic examinations of the vessel steel samples have been completed, including microstructural examinations and hardness measurements. Results of these examinations have provided preliminary estimates of temperature histories of the lower head samples. These results show that the maximum inner surface temperature of at least four samples reached 1050–1100°C for about 30 minutes. These samples were extracted from the lower head from a small region, approximately 2 to 3 feet in diameter. The examinations also indicated that the temperature 2 inches into the wall was about 100°C lower than the inner surface temperature. (The lower head thickness was 5 inches.) Mechanical testing is under way to determine tensile and creep properties of the vessel steel at high temperatures (up to about 1100°C).

Metallographic and scanning electron microscope (SEM) examinations of the instrument tube nozzles are being performed to assess the nozzles' axial temperature profile and interaction of the Inconel 600 nozzle material with molten core materials. Examinations of previously molten thermocouples in the nozzles may be an important indicator for determining the axial temperature profile in the nozzles.

In September 1991, the VIP Management Board decided to amend the original agreement extending the VIP program from September 30, 1991, to March 31, 1993. The objectives of the amended program are to (1) perform more detailed testing and examination of the in-core instrument tube nozzle penetrations and the in- core instrument guide tubes that were extracted from the lower head, (2) perform additional analyses of potential reactor vessel failure modes based on data from sample examinations, and (3) assess the margin-to-failure of the lower head of the reactor vessel.

A.4.3 Future Plans

Results of the ongoing TMI–2 lower head examinations are expected to provide additional information on the physical properties of the specimens, temperature distributions in the instrument nozzles, and interactions between the molten core material and the vessel. These results then will be used to perform scoping analyses of potential reactor vessel failure modes, such as penetration tube failures and global or local failure of the reactor vessel lower head. More detailed analyses of the most likely failure mechanisms will be performed to estimate the margin-to-failure of the lower head. A final project report, integrating the results of all the sample examinations and analyses, will be issued at the completion of this program in June 1993. June 24, 1992

A.5 Fuel-Coolant Interactions (FCI) and Debris Coolability

A.5.1 Background

One of the fundamental phenomena in the course of a severe accident is the interaction of degraded core materials with coolant. In the absence of coolant, the core materials overheat, melt, and relocate. The opportunity for fuel- coolant interactions arises not only as a natural consequence of this relocation process (into areas occupied by coolant), but also as a consequence of accident management actions that add water to regions previously depleted of coolant. Considering the variety of reactor geometries, meltdown scenarios, and timing of coolant addition, a broad range of resultant phenomena is possible.

The intensity of the interactions and resulting material state and configuration depend on: (1) the geometry of the region, (2) the masses of core material and coolant involved, (3) the thermodynamic state of the materials, and (4) the rates of pouring of molten core materials into a pool of coolant or the rate of flooding a degraded or molten core. For example, at one extreme, a slow pour of molten core debris in a large, deep pool of water can lead to complete quenching and formation of a coarse particle debris at the pool bottom. At the other extreme, a rapid, massive release could yield a highly energetic explosion with significant mechanical consequences on the system and its surrounding structures. Initially, the subject of FCIs emphasized the phenomenon of a vapor-explosion-induced missile as a possible mode of containment failure. With increased emphasis on accident management, interest in FCIs broadened to include core degradation regimes that can be quenched by flooding with water. For such scenarios, it is important to identify potential circumstances in which FCIs can lead to coolable configurations or can significantly alter the core melt progression scenario.

Certain fundamental design differences between PWRs and BWRs require different areas of emphasis. During the later phase of core melt progression in PWRs, the potential exists to accumulate large quantities of melt in the core region "crucible." Because of the largely open lower plenum geometry, in-vessel FCIs can span the whole range from energetic to relatively benign. For BWRs, on the other hand, if a metallic or ceramic blockage does not occur, large melt accumulations in the core region could be excluded. (The research effort on core blockage or melt drainage in BWR severe accidents is discussed in Section 2.2.2.1 of this report.) In addition, the lower plenum is crowded by the control rod drives. Therefore, small energetic FCIs are more likely, and the emphasis for BWRs is on coolability.

During the ex-vessel stage of a severe accident, water may be added to the reactor cavity as a deliberate measure to mitigate the consequences of core debris-concrete interactions. Water may be present in the cavity as a result of natural processes such as the condensation of steam, flow from containment sprays, or discharge from accumulators in the reactor coolant system. For all containment geometries, various FCI concerns exist that span the whole range from energetic interactions to benign coolability.

A.5.2 Status

There are three specific issues addressed in this section:

1. FCI energetics,
2. Fuel-melt quenching in water pools, and
3. Adding water to degraded core (in-vessel and ex-vessel).

A.5.2.1 Status of FCI Energetics

This topic is of primary relevance to in-vessel interactions in PWRs. The first quantification of the potential for alpha-mode containment failure was offered in WASH-1400. Additional quantification of this failure mode can be found in the expert opinion efforts, i.e., NUREG-1116, "Steam Explosion Review Group, A review of Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosion, 1985," and NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, 1990." It is generally agreed that the probability of alpha-mode containment failure is negligible, although the need to improve the quantification basis for this conclusion is generally acknowledged. NUREG/CR–5030 presents a probabilistic framework that arrives at the same conclusions. This report highlights "premixing" as the process of primary importance in limiting the magnitude of energetic FCIs.

The issue of premixing arises because there is a possibility that core relocation into the lower plenum can occur in large massive pours in PWRs. The geometry of the PWR diffuser plates leads to the breakup of the melt into several smaller jets. Because of extensive steaming, large premixtures, which are expected to develop in the above circumstances, would be largely void of water, and therefore, the magnitude of energetics is limited.

A.5.2.2 Status of Fuel-Melt Quenching

Fuel-melt quenching in severe accident evaluations has several important aspects. For the in-vessel core melt portion of severe accident sequences in current generation PWRs and BWRs, as well as their ALWR counterparts, quenching considerations are fundamental in predicting the mode and timing of lower head failure and the resultant impact on direct containment heating (DCH). In accident management, increasingly more consideration is given to the possibility of flooding the cavity region external to the lower head of the reactor vessel in order to prevent vessel failure and retain core debris in the vessel. For the ex-vessel portion of severe accident sequences, quenching plays an important part in determining the ex-vessel relocation behavior of the melt, and hence, long-term coolability.

The fundamental difficulty in addressing the role of quenching in severe accident scenarios relates to uncertainties in (1) the flow characteristics of the pour (i.e., size and number of melt jets, pour rates) and (2) the composition of the melt (i.e., metallic vs. oxidic components) and its temperature (superheat). Conservative evaluations of such uncertainties and of the quenching process itself, particularly on assessment of containment integrity, are possible. However, a better understanding of quenching will contribute to the depth of understanding such that better judgments, especially on new plant designs, can be made in the future.

A.5.2.3 Status of Adding Water to Degraded Core

The reflooding mode refers to two different configurations:

1. Degraded core configuration in which water is supplied from above or below the core, or

2. A relatively shallow molten corium pool (at the bottom of the lower plenum or on the concrete basemat) with water on top of the molten pool.

The phenomena and technical issues for these configurations are quite different and are addressed separately below.

A.5.2.3.1 Adding Water to a Degraded Core (In-vessel)

A review of past experiments suggests that a few tests have been performed with water addition following bundle degradation, mainly, CORA, PBF, and the LOFT-FP2 test. Also, simulant tests of a coolant added to fuel debris have been performed at BNL, ANL, and UCLA. Although limited in scope, these tests address water addition during core degradation. The major variables affecting the consequences of water addition are the rate and character of water addition and the state of the fuel at the time water is added. Reference states for the fuel are:

1. Initial heat up and early core degradation

   This first stage of a severe accident sequence is defined as beginning with the onset of core uncovery and continuing up to the clad melting (not eutectic dissolution) and relocation. Coolant is still easily accessible to the degraded portion of the core because of efficient lateral flows through the open PWR lattice. Key modeling difficulties include vapor chimney effects and permeability.

2. Advanced core degradation (core rubble, melt, and relocation)

   This stage of the severe accident sequence is defined as beginning with the first significant fuel melting and relocation and extending through core debris relocation onto the lower plenum. The key events in the late stage are the formation of a molten pool, pool growth, and breakthrough of the crust supporting the molten pool. The behavior following water addition during this stage is governed by crust stability, which is dominated by the heat transfer properties of the (porous) medium surrounding the crusts and of the crusts themselves, and the heat load distribution on the inner surface of these crusts by natural convection of the molten core material within the crust.

A.5.2.3.2 Adding Water to Degraded Core (Ex-vessel)

The continued progression of a severe accident can lead to the expulsion of reactor core debris into the reactor cavity. Debris in the reactor cavity can interact with the structural concrete of the containment and even, in some cases, directly with the pressure boundary of the containment.

The configuration of core debris that is expelled from the reactor vessel does not ensure that the mere presence of water will result in cooling the debris sufficiently to eliminate core-concrete interactions. The core debris must be fragmented into coolable rubble, or it must spread over a large enough area that heat extracted by overlying water will cool the debris.

At this time, a technically justified criterion for debris coolability during the ex-vessel phases of severe reactor accidents cannot be defined. Issues that must be resolved to define such a criterion can be broadly categorized as issues of debris configuration and issues of heat removal from core debris by water.

The NRC is participating in the MACE test program, which is examining the effects of water on uranium-dioxide-rich melts interacting with concrete. In the past, the NRC has sponsored tests of high temperature oxidic and metallic melts with water and concrete (the WETCOR tests) to determine limits of coolability.

APPENDIX B
CLOSURE OF SEVERE ACCIDENT ISSUES

B.1 Source Term

B.1.1 Background

Radionuclide releases to the environment, that is, the type, quantity, timing and energy characteristics of the release of radioactive material from reactor accidents ("source terms") are deeply embedded in the regulatory policy and practices of the NRC. For almost 30 years the NRC's reactor site criteria (10 CFR Part 100) have required that for licensing purposes, an accidental fission product release from the core into the containment be postulated to occur and that its radiological consequences be evaluated assuming that the containment remains intact but leaks at its maximum allowable leak rate.

Evaluation of the consequences is used to assess both plant mitigation features such as fission product cleanup systems and the suitability of the site. The characteristics of the containment "source term," which must be distinguished from a release to the environment, are described in Regulatory Guides 1.3 and 1.4, but are derived from the 1962 report TID–14844 (Ref. 1). The source term consists of 100% of the core inventory of noble gases and 50% of the iodines (half of which are assumed to deposit on interior surfaces very rapidly). Regulatory Guides 1.3 and 1.4 also specify that the source term is instantaneously available for release and has significantly affected containment isolation valve closure times. The guides also specify that the iodine is predominantly (91%) in elemental (I_2) form.

In addition to plant mitigation features and site suitability, the regulatory applications of this release also establish (1) the post-accident radiation environment for which safety-related equipment should be qualified, (2) post-accident habitability requirements for the control room, and (3) post-accident sampling systems and accessibility.

In contrast to a specified source term for design basis accidents, severe accident source terms first arose in probabilistic risk assessments (e.g., Reactor Safety Study, WASH-1400, Ref. 2) in examining accident sequences that involved core melt and possible containments failure. Severe accident source terms represent mechanistically determined "best estimate" releases to the environment, including estimates of failures of containment integrity. This is very different from the combination of the nonmechanistic release to containment postulated by TID–14844 coupled with the assumption of very limited containment leakage used for Part 100 citing calculations for design basis accidents. The worst severe accident source terms resulting from containment failure (especially early failures, i.e., within a few hours from onset of an accident) or containment bypass can lead to consequences that are much greater than those associated with a TID–14844 release into containment when the containment is assumed to be leaking at its maximum leak rate for its design conditions. Indeed, some of the most severe source terms arise from some containment bypass events, such as "event V" and multiple steam generator tube ruptures.

Source term estimates under severe accident conditions began to be of great interest shortly after the Three Mile Island 2 (TMI–2) accident. The objective of a major NRC research effort on source terms that began about 1981 is to obtain a better understanding of fission-product transport and release mechanisms in LWRs under severe accident conditions. This research effort, which involved a number of national laboratories as well as nuclear industry groups, has resulted in the development and application of several new computer codes to examine core melt phenomena and associated source terms. Work has also included significant review efforts by peer reviewers, foreign partners in NRC research programs, industry groups, and the general public. Current risk assessment methods, including the latest research efforts on severe accident source terms, are reflected in NUREG–1150 (Ref. 3), which provides an assessment of severe accident risk for five U.S. nuclear power plants. Finally, the occurrence of the accident at Unit 4 of the Chernobyl reactor in the Soviet Union on April 26, 1986, and the large accidental release of fission products resulting from it has provided further impetus to understand severe accident source terms as well as to prevent such occurrences.

B.1.2 Status

Since shortly after the accident at TMI–2, the NRC has sponsored numerous experimental and analytical research projects on fission product release and transport. Early experiments and analytical work tended to focus on release from fuel material under high temperatures and severe accident environments. Later, experimental data on the behavior of aerosols in the RCS and the containment were obtained. These data were used to develop aerosol deposition and transport models to analyze fission product behavior in the reactor coolant system and the containment. Currently, fully integrated models are being assembled into individual codes, the VICTORIA code (Ref. 4) and the CONTAIN code, (Ref. 5), for the analysis of in-vessel and ex-vessel source terms, respectively.

The issue of revaporization of deposited radioactive materials in the RCS is of particular concern because of the instability of the deposited radioactive material at high temperatures induced by decay of nuclides. Recent severe accident calculations have predicted that, for some sequences, natural circulation of gases through the reactor core may also heat structures in the RCS to substantially higher temperatures than had been previously predicted. The high temperatures may also result in the failure of RCS piping at certain locations. This failure of RCS piping is of some significance since current analyses of core degradation indicate substantial fractions of the core could be retained within the RPV after vessel failure. Evidence from TMI–2 suggests that as much as half the core material may have stayed within the original confines after the rest of the core had melted and drained into the lower plenum. This remaining fuel in the core region could be exposed to air once the plenum has been breached. Air will react exothermically with the cladding remaining on the fuel, producing high temperatures in this fuel. Once the cladding has been oxidized, vapors of radionuclides not usually considered highly volatile, notably ruthenium and molybdenum, will be produced because of the strongly oxidized conditions. The VICTORIA code has incorporated a model to address revaporization caused by an increase in temperature of the RCS. The VICTORIA code also models other important severe accident phenomena such as the release and transport of fission products, condensation of vapors, aerosols behavior, and chemical reactions in the RCS.

In low-pressure accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity where it interacts with the concrete. At high temperatures (approximately 1,300–1,500°C), concrete decomposes, and the ablation products commonly include water vapor and carbon dioxide as well as the refractory oxides CaO and SiO_2. The liquefied oxidic components of the concrete mix with the uranium oxide fuel and metallic oxides of the debris. Typically, the core debris is initially all or partially molten; gases released at the debris-concrete interface bubble through the debris pool reducing some low-vapor-pressure oxides such as La_2O_3 to high-vapor-pressure forms such as LaO. These more volatile forms then vaporize into the bubble volume, thus releasing fission product species that were not released in the vessel. Aerosols are formed when the bubbles exit the upper surface and fragment. Among the factors that influence the magnitude of the ex-vessel releases are the composition and temperatures of the core debris. Concrete composition also has a major impact on the amount of aerosols entrained into the containment atmosphere. Limestone concrete produces larger gas flows and is more oxidizing than basaltic concrete. An extensive experimental data base has been obtained on core-concrete interactions with no overlying water pool. If an overlying water pool exists, a considerable amount of the aerosols may be scrubbed and kept out of the containment atmosphere. Research effort is under way to obtain experimental data for this case. Core-concrete interactions and associated radionuclide releases are modeled by the stand-alone code, CORCON-MOD3 (Ref. 6). These models will also be incorporated into the CONTAIN code to allow comprehensive evaluation of containment behavior during severe accidents. In addition, the CONTAIN code models the thermal-hydraulics (pressure, temperatures, etc.), aerosols behavior, fission products behavior and transport, and hydrogen behavior in the containment. Codes such as VICTORIA, CONTAIN,and CORCON-MOD3 incorporate mostly mechanistic models for severe accident phenomena, thus allowing the examination of complex source term issues in a detailed and systematic fashion. On the other hand, the MELCOR code (Ref. 7) was developed as an integral tool for analysis of fission product transport in the RCS and in the containment. MELCOR employs simpler models for severe accident phenomena in order to facilitate the fast running time requirements for the repetitious calculations used in PRA analysis.

With respect to the potential release of iodine from suppression pools and reactor cavity water, the ACE program and the ORNL iodine chemistry research have provided extensive experimental data to address this issue. At ORNL, the research included iodine partition coefficient tests, hydrolysis chemical kinetics tests, radiolysis chemical kinetics tests, hydrogen burn/iodine chemistry tests, and TRENDS models development. These efforts were further enhanced by the ACE program, which included hygroscopic aerosol/iodine chemistry tests, and hydrogen burn/iodine chemistry tests. Many iodine chemistry models were developed from these data bases. These models are now being incorporated into the CONTAIN code. Further validation of the CONTAIN code could be done using the PHEBUS-FP data.

B.1.3 Source Term Uncertainties and Present Research Efforts

With respect to source term uncertainties, NUREG–1150 has identified specific source term issues as contributors to uncertainty in risk estimates. For fission product release from fuel and retention in the RCS, a key question is, "How significant is fission product release during the late stages of core melt vs. the early phase?"

The present theoretical model for the late-stage rubble bed (significant relocation and melting of ceramics) assumes that release is governed by gas-phase mass transport. For the molten pool, the main mechanism for fission product release is governed by diffusion and surface convection. Both of these theoretical models need experimental data for validation. The predominant sources of uncertainty in these theoretical model are:

- geometry of the core debris,
- magnitude of gas fluxes through the debris, and

- thermo-chemical properties of the high-temperature vapor species that vaporize from the debris.

The VICTORIA code incorporates models to address these questions. Uncertainties in our current understanding of the evolution of accidents are handled in a parametric fashion.

Integral experiments to study fission-product release from late-phase core melt progression and revaporization are not being planned in the United States. However, the PHEBUS–FP project could provide some of the needed experimental data.

For late-phase revaporization of fission product from the RCS, the key questions are:

1. After the reactor vessel has been breached and air ingress occurs, what are the release rates from the fuel remaining in the vessel? Likewise, for shutdown accidents, what are the release rates from the fuel exposed to air?

2. What chemical forms are important during the transport and retention of aerosols and vapors?

As mentioned earlier, if the vessel is penetrated, air from the containment atmosphere will circulate over retained fuel. The air will react exothermically with the cladding remaining on the fuel, producing high temperatures in this fuel. Once the cladding has been oxidized, vapors of radionuclides (notably ruthenium and molybdenum, radioactive species not usually thought to make major contributions to severe accident source terms) will be produced because of the strongly oxidized conditions. Tests performed at Chalk River Laboratory in Canada with uncladded and cladded fuel, and at ORNL with fuel fragments under highly oxidized conditions, showed that a large VI–7 fraction of the ruthenium, tellurium, and molybdenum was released. The predicted release rates are dependent on a highly oxidized condition for the fuel. However, it is necessary to conduct tests for radionuclide releases for typical LWR fuel, as opposed to the thinner cladding used in the CANDU fuel. Plant configurations during shutdown situations may lead to the possibility of air ingress into the core either by natural circulation or from the residual heat removal system. That air will react exothermically with the cladding on the fuel, producing high temperatures in the fuel, resulting in massive vaporization of ruthenium, tellurium, and molybdenum. A test (VI–7) will be conducted at ORNL for fission-product release at high temperature in an air environment, the experimental data could be readily incorporated into the current VICTORIA fuel-release model.

The chemical form assumed for iodine affects late-phase revaporization. If cesium iodide or other iodides decompose to produce HI or I2, there will be little or no retained iodine in the RCS to revaporize after containment failure. If CsI is stable, substantial amounts will be retained temporarily in the RCS and will be able to revaporize, creating an iodine source term after containment failure. Tests at Winfrith and SNL have shown that CsI will react with boron oxides vaporized into the RCS atmosphere to yield cesium borate, I, and HI. Tests at SNL have shown thermal instability of CsI in the RCS environment. These tests have not clarified whether I or HI produced by reactions of CsI can subsequently react to form other iodides such as NiI_2(g,c). Analyses done in the U.K. suggest that high vapor fractions of iodine (CsOH and CsI) at the time of RCS failure could yield aerosols in containment that do not settle rapidly and are only slightly affected by containment sprays. Hence, a source of airborne fission product would be available for leakage out of the containment.

Currently, there are no suitable experimental data to validate revaporization models, but the PHEBUS–FP tests could provide some. It is likely that continued examination of the chemical form of I or HI produced by reaction of CsI could alter the perceptions of risk by altering the predicted amount of suspended radioactivity in containment at the time of containment failure. This could be accomplished by sensitivity analysis using VICTORIA.

It is also important to utilize risk perspectives regarding the uncertainties in late-stage core melt and late-phase revaporization. Generally, risk is increased by the early release of fission products into containment and early containment failure or bypass. Hence, additional fission products released later in an accident phase will denote lesser releases at an earlier stage. Modeling sensitivity can be made in risk assessments that can test uncertainties and their implications.

A key question regarding the ex-vessel source term is, "What effect does hydrogen combustion have on aerosols suspended in the containment atmosphere?"

Aerosol materials containing CsI must be dehydrated and vaporized before chemical interactions of cesium iodide can be expected. Tests conducted at ORNL, as part of the ACE program, found that vaporized CsI was unstable in hydrogen flames with the iodine redistributing as iodide, I_2, and iodate. An excess of metal cations (Cs) reduced the extent of I_2 formation. Oxidation to I_2 was consistent with thermal decomposition of CsI, but iodate production was related to the nonequilibrium OH and O radical concentrations found in hydrogen/air flames. Data are believed to be adequate to address this question. However, the presence of other aerosol species in the containment atmosphere, such as aerosols produced by core-concrete interactions, could affect the stability of CsI during hydrogen combustion events, even though it is a small contribution. For instance, silica could trap cesium to form cesium silicate so that it cannot recombine with iodine. Other basic species could react with iodine produced in the combustion to reform iodides:

$$Na_2O(c) + 2HI(g) \rightarrow 2\ NaI(c) + H_2O.$$

Currently, the TRENDS model (Ref. 8) treats this in a conservative fashion (i.e., it assumes that I_2 will be formed in a hydrogen burn). Such a model has been developed for inclusion in the CONTAIN code.

In conclusion, although additional physics and chemistry research to reduce uncertainties in source term phenomena can be performed, it is important to consider the need for such research and the potential that this research could significantly improve our risk perspective associated with severe accidents. NUREG–1150 indicates that uncertainties in overall estimation of risk are largely driven by uncertainties in containment performance, primarily those associated with estimation of containment loads, estimation of containment performance at load levels beyond the design basis, and estimation of the probability and location of containment bypass.

B.1.4 Regulatory Applications/Implications

B.1.4.1 Development of Updated Source Term

Design basis accident source terms have been used in the United States for licensing purposes in three distinct ways, namely:

1. For siting evaluations as required by 10 CFR Part 100,

2. For defining the radiological environmental conditions for certain plant systems, and

3. For assessing the effectiveness of plant mitigation systems.

The NRC is presently preparing an update of the source term contained in Regulatory Guides 1.3 and 1.4, making use of current severe accident research insights. This effort is expected to result in changes in data for the timing of the release, the composition and magnitude of fission product releases into the containment, and the chemical form of the iodine fission products. A draft of an updated report replacing TID–14844 is expected to be issued for comment by the first half or CY 1992.

Rather than an instantaneous release into the containment, the revised formulation is expected to be stated as a series of fission product releases into the containment, each one associated with a particular stage of an accident or group of accidents. Hence, the revised formulation is expected to begin with the release of coolant activity, followed by the release of activity in the fuel gap, the release of fission products associated with gross fuel degradation prior to reactor vessel failure, and finally, release of fission products from core-concrete interactions.

Additional nuclides other than the noble gases and iodine are expected to be released. For example, preliminary indications are that the fraction of core inventory of cesium released into the containment is generally comparable to that of iodine. In addition, some tellurium and smaller fractions of the remaining nuclides are also expected for release.

A recent study on iodine chemical form and behavior on entering the containment from the RCS, and the subsequent revolatilization of iodine from water pools in containment, has been completed at ORNL. ORNL examined a group of severe accident sequences used in NUREG–1150. These accident scenarios were for both high- and low-pressure sequences that are risk significant. For the RCS, the analysis considered the chemical kinetics of 20 reactions of iodine with water, hydrogen, and cesium, and determined the temperature and time when chemical equilibrium was established. Once chemical equilibrium was established, an analysis determined the iodine chemical forms present. In most calculations, iodine was released from the RCS into the containment as cesium iodide (CsI) with very small amounts of I or HI. Since the ORNL's study considered a limited set of reactions related to CsI, iodine in the form other than CsI could be released from the RCS. In order for the RCS to release iodine in the form other than CsI, a significant fraction of the Cs has to be removed within the RCS. However, for the accident conditions analyzed, only a small fraction of Cs was removed within the RCS. If desired, more sophisticated treatment of CsI reactions could be undertaken with VICTORIA to confirm the ORNL's findings. The ORNL results indicate that the iodine entering containment is at least 95% CsI, 5% as I and HI, with not less than 1% as either I or HI. This is in contrast to the iodine chemical form specified by the TID source term, which is predominantly (91%) elemental.

The ORNL iodine research and the ACE program have provided data that address revolatilization of iodine from water pools. A comprehensive model to estimate the revolatilization of iodine from water pools was developed by ORNL. Once iodine enters containment, it dissolves in water pools or plates out on wet surfaces as I^-. Subsequently, the iodine behavior within the containment depends upon time and the ph of the water solutions. If the ph is maintained at a value of 7 or greater, the amount of iodine in solution that converts to I_2 and organic iodine later in the accident sequence will be very low. If ph is not controlled, radiation levels in water pools are sufficient to convert much of the dissolved iodine to elemental iodine for release into the containment atmosphere. Experimental data on the revolatilization of iodine from water pools resulted from the evolution of Ph and irradiation under severe accident conditions are expected from the

PHEBUS-FP project (section 2.7.2). These data will provide confirmatory assessment of models for iodine revolatilization from water pools.

B.1.4.2 Regulatory Implications

At the present time, the NRC is pursuing several regulatory initiatives to incorporate insights from updated severe accident source terms. A revision of the NRC's reactor site criteria (10 CFR Part 100) is being carried out in parallel with an interim revision of 10 CFR Part 50. The reactor site criteria will be revised to remove source term and dose calculations and to add requirements in Part 100 for exclusion area size and population density based on those from Regulatory Guide 4.7. At the same time, Appendix A to Part 100, containing site seismic criteria, is also being revised to reflect the latest understanding. Source term and dose criteria will continue to be important for plant design; consequently, an interim revision of 10 CFR Part 50 will be carried out in parallel and will contain the present source term (i.e., that from TID–14844 and Regulatory Guides 1.3 and 1.4). These proposed rule changes are expected to be issued for comment by early CY93.

Updated source term insights arising from the technical update of TID–14844 are expected to be made available for voluntary use by existing licensees. A final revision of 10 CFR Part 50 to incorporate updated source term and severe accident insights will then be undertaken, with a proposed rule for comment expected to be issued by early 1993.

Although regulatory positions arising from updated source term insights remain to be developed, some preliminary implications can be seen at this time. It is clear that updated source term insights indicate the need for consideration of nuclides (e.g., cesium) in addition to iodine and the noble gases. In addition, revised insights on iodine chemistry call into question the need for high-efficiency charcoal absorbers (assuming that the Ph is controlled postaccident). These can, in turn, impact such important plant systems as fission product cleanup systems, control room habitability, and allowable containment leak rate.

Finally, and most importantly, the above discussion and all recent risk studies have shown the importance of maintaining containment integrity under severe accident conditions in order to assure low risk. This strongly suggests that the appearance of a severe accident source term within containment should be more closely linked with the temperatures, pressures, and containment loads and challenges associated with such releases, rather than an arbitrary linkage with a single sequence such as a large-break loss-of-coolant-accident.

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B.2 Core-Concrete Interaction

B.2.1 Background

Core-concrete interactions would occur during a severe accident only after penetration of the reactor vessel and flow of the core debris onto the concrete basemat. Decomposition of the concrete from this interaction results in the release of steam and carbon dioxide, which may be partially reduced to the combustible gases hydrogen and carbon monoxide. As the gases pass through the hot molten debris, they sparge small but potentially important quantities of radioactive elements from the debris. These radioactive aerosols can be released to the containment, thereby adding to the accident source term. The major areas of concern associated with core-concrete interactions during a severe accident are the complete penetration of the basemat and the generation of radioactive aerosols and combustible gases. Another related concern is the overheating of important structures inside the containment.

The NRC has conducted an extensive program of analytic and experimental research to obtain improved understanding of core-concrete interactions. The analytical research focused on the development of models for studying phenomenological aspects of core-concrete interactions such as heat and mass transfer, while the experimental research focused on conducting scaled-down experiments simulating prototypic reactor accident scenarios. These studies have recognized the variety of concretes used in nuclear power plants in the U.S. and the widely diverse accident scenarios that lead to core-concrete interactions. The effort to understand core-concrete interactions was also broadened to include a reassessment of the models used to predict radionuclide release.

The efforts to resolve the severe accident issues associated with core-concrete interactions have culminated in the development of the CORCON computer code. The predictions of the early versions (CORCON-MOD1 and CORCON-MOD2) of this computer code have been validated against many large-scale tests of the interactions of both metallic and oxidic core debris and with concrete.

B.2.2 Status

The experimental data base on core-concrete interactions is extensive. This data base is predominantly the result of research sponsored by the NRC and research sponsored in Germany at the Kernforschungszentrum in Karlsruhe. Data on the interactions of uranium dioxide melts with concrete have been obtained from the ACE program sponsored by the Electric Power Research Institute, in which the NRC is a partner. Additional data on core-concrete interactions may come from the ALPHA program in Japan and studies under way in Russia.

The first consideration regarding the range of conditions that could affect core-concrete interactions during a severe accident is the type of concrete. The data base now available for core-concrete interactions includes tests with both of the general classes of concretes in use in U.S. commercial nuclear power plants (i.e., siliceous and calcareous concretes).

Another important consideration deals with the composition of the core debris itself, particularly the content of metallic zirconium in the mixture. The oxidation of metallic zirconium in the core debris can elevate the temperature of the debris during the interaction with concrete, which would result in an increase in the release of normally refractory fission product elements to the containment atmosphere. Also, the presence of metallic zirconium produces a chemically reducing environment that can increase the release of certain key elements.

Many of the early tests of high-temperature melts interacting with concrete were of an exploratory nature that were undertaken to identify phenomena or to develop experimental techniques. More recent tests have been conducted to explicitly validate the models of core-concrete interactions. Such tests have been heavily instrumented to obtain heat balances and data on gas generation, gas composition, and aerosol production.

The available data base spans a broad range of conditions as well as concrete types. The data base includes tests with both oxidic and metallic debris. There are some limitations to the data base (i.e., the data base is of a generic nature and may not address specific issues that arise at particular nuclear power plants). However, model predictions do not indicate that there are substantive uncertainties or issues affecting core-concrete interactions. Some discrepancy has been traced to the material phase relationship used in the CORCON code. Measurements of the melting properties of UO_2–ZrO_2 concrete mixtures are being sponsored by the NRC, and improved models are being incorporated in the most recent version of the code CORCON–MOD3.

With regard to the radionuclide release during core-concrete interactions, an effort has been made in the development of CORCON-MOD3 to include a detailed mechanistic model of aerosol generation and radionuclide release (the VANESA model). The model is based on the assumption that aerosols are produced by the mechanical entrainment of melt when gas bubbles burst at the surface of the melt and by vaporization of volatile melt constituents into gases sparging through the melt. Predictions of the total aerosol generation rate compared to results of the BETA test show that the model predictions are well within the expected accuracy limits of the available thermodynamic data base.

B.2.3 Future Plans

Refinements to the CORCON–MOD3 code are being made with regard to phase relations and models of non-ideal solutions. An effort to compare the code predictions to test results will be completed. Part of the validation of the CORCON MOD3 code will be carried out under an arrangement with the I.V. Kurchatov Institute. The validation will include comparison with data from the BETA (KfK), SURC (SNL) and ACE program. Once validation efforts are finished, the development of CORCON–MOD3 will be complete as a stand-alone model of core-concrete interactions. CORCON–MOD3 will then be incorporated into the CONTAIN and the MELCOR system level codes for severe accidents.

B.3 Closure of Severe Accidents Scaling Methodology

B.3.1 Background

In many areas of severe accident research, the experimental investigation has evolved or progressed to the point that experimental programs seek to resolve very specific issues of uncertainty for particular geometries and reactor plant configurations. The aim of these programs is often to produce results that can be characterized as directly applicable to reactor behavior, or at the least suitable for the development of models that allow for extrapolation to severe accident reactor analysis.

As part of the Revised Severe Accident Research Program Plan that was published as NUREG–1365 in August 1989, the NRC identified initiation of a severe accident scaling methodology (SASM) development program as a programmatic element. Experimental investigation of severe accident phenomena poses a serious challenge owing to the fact that many processes involve a complex synergy of fluid flow across varied flow regimes, combined heat transfer modes, high temperature material interactions, and chemical reactions. Development and application of an SASM, representing a structured methodology that is systematic, comprehensive and scrutable, provides the confidence that scaled experiments faithfully reproduce the phenomena that will occur in a nuclear power plant. Further, application of an SASM provides the basis for application of analytical models validated against smaller scale experimental facilities to full scale.

B.3.2 Status

To address the scaling problem, the NRC implemented a SASM development program at the Brookhaven National Laboratory (BNL). A technical program group (TPG) was formed by the contractor to guide the development of the SASM and to demonstrate its efficacy by applying the methodology to the direct containment heating problem.

The results of the TPG activities culminated in the documentation of the SASM and its application in NUREG–5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," published in November 1991.

The SASM consists of a number of steps that are grouped in three key elements:

Element 1: Specification of Experimental Requirements, in which the experimental objectives are defined in terms of the technical issues.

Element 2: Evaluation and Specification for Experiments and Testing, in which the experimental objectives are reflected in terms of scaling rationales that are necessary to ensure that both separate-effects tests and integral-effect test data are applicable to full-scale reactors, and that the tests include the phenomena important to the specified accident scenario.

Element 3: Data Acquisition and Documentation, in which the data base appropriate to issue resolution is established and documented for subsequent use.

To perform scaling analyses that satisfy the objectives of SASM, a hierarchically based, two-tiered scaling methodology was developed. The complex physicochemical processes that characterize severe accident scenarios and their associated synergetic effects mandate a hierarchical approach to the problem in order to make it tractable. The two-tiered scaling methodology involves a top-down system scaling tier and a bottom-up or process scaling tier. The top-down system scaling analysis provides the basic conservation equations as well as the scaling rationale and similarity groups to be preserved in experimental design. Additionally, it is through the system scaling analysis that quantification of the effects of any distortions is achieved. The bottom-up process scaling tier focuses on the models for the important processes that drive the system response.

In the application of the SASM to the DCH issue, RPV conditions were examined to evaluate the initial and boundary conditions appropriate for experimental simulation. Scaled model laws for RPV discharge phenomena and for reactor cavity phenomena were then derived using the two-tiered scaling methodology. The SASM evaluation of initial and boundary conditions for DCH in a PWR for a station blackout scenario with failure of a lower head penetration served as the basis for the DCH integral-effects tests initiated in September 1991.

The scaling of important processes indicates that debris dispersion or entrainment of corium is a strong function of the gas velocity after conditions for the onset of entrainment are satisfied. Expressed in terms of pressure, entrainment is a function of the pressure ratio (reactor vessel/reactor cavity) to the 4.6 power as well as a function of gas and fluid properties. The practical effect of these dependencies, at least as analyzed for the Zion and Surry plants, is the very rapid increase of entrainment fractions from low initial reactor pressures up to effectively complete dispersal at reactor pressures of approximately 400 psi.

B.3.3 Future Plans

NUREG–5809 was issued as a draft for public comment in November 1991. The authors of the report acknowledged that the SASM application to DCH was not intended to represent complete technical resolution of the scaling questions, as some of the scaling relationships derived are based on specific assumptions that require experimental confirmation and additional analysis. Resolution of those technical issues as they affect both scaling of experimental facilities and operation, as well as modeling development for reactor analysis, will be pursued under the DCH research program described in section 2.1. The discipline of a SASM will be applied to other experimental programs in the SARP as necessary. The staff also intends to incorporate comments received on NUREG-5809 into a final version of the report, tentatively scheduled to be issued by December 1992.

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In August 1989, the staff published NUREG–1365, "Revised Severe Accident Research Program Plan." Since 1989, significant progress has been made in severe accident research to warrant an update to NUREG–1365.

The report describes recent major accomplishments in understanding the underlying phenomena that can occur during a severe accident. These include Mark I liner failure, severe accident scaling methodology, source term issues, core-concrete interactions, hydrogen transport and combustion, TMI–2 Vessel Investigation Project, and direct containment heating. The report also describes the major planned activities under the SARP over the next several years. These activities will focus on two phenomenological issues (core melt progression, and fuel-coolant interactions and debris coolability) that have significant uncertainties that impact our understanding and ability to predict severe accident phenomena and their effect on containment performance. The SARP will also focus on severe accident code development, assessment and validation. As the staff completes the research on severe accident issues that relate to current generation reactors, continued research will focus on efforts to independently evaluate the capability of new advanced light water reactor designs to withstand severe accidents.

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