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POLICY ISSUE

(Information)

June 21, 1990 SECY-90-224

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: STATUS OF SEVERE ACCIDENT RESEARCH

Purpose: The purpose of this paper is to inform the Commissioners of the progress of research activities undertaken in conjunction with implementation of the "Revised Severe Accident Research Program" (SECY-89-123) (NUREG-1365).

Background: The staff provided, for Commission consideration, a description of its "Revised Severe Accident Research Program" (SARP) in SECY-89-123, April 20, 1989. A description of the SARP was subsequently published as NUREG-1365. The revised SARP is intended to guide the formulation and conduct of severe accident research by addressing, in the near-term, those issues pertinent to the implementation of the "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147), with primary emphasis on issues associated with accident sequences that lead to early containment failure. While the primary focus of the SARP is on those issues which dominate risk for nuclear power plants, the SARP also includes elements of our long-term research plan. Long-term research is directed at improving our understanding of the range of phenomena exhibited by severe accidents and developing improved methods for assessing fission product behavior and their availability for release in the event of containment failure at various phases of severe accident scenarios.

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER

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Summary: The revised SARP identified the five specific near-term areas for near-term research and evaluation:

1. Scaling
2. Depressurization and Direct Containment Heating
3. BWR Mark I Containment Shell Meltthrough
4. Adding Water to a Degraded Core
5. Use and Status of Severe Accident Models (Codes)

The staff's efforts over the past year to develop a severe accident scaling methodology (SASM) have produced a generic approach which has been applied to the issue of direct containment heating (potential for early containment failure). The results of this activity have been used to guide planning for the specific tests related to direct heating phenomena. Activities during the past year to address Item 2 above, depressurization and direct containment heating, have centered around incorporation of the insights derived from the SASM program as well as completing scoping tests to ensure future direct containment heating testing is successful and will provide the information needed for regulatory closure.

In February of this year, the staff distributed for peer review draft NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment." This study represents the first integrated engineering approach to the problem which includes probabilistic treatment of the uncertainties surrounding the major phenomenological issues. This study utilized data from previous and ongoing NRC programs as well as information developed by the industry. A workshop to explore and address the comments of the peer review group is scheduled for July 23-24, 1990.

Most of our progress to resolve issues related to adding water to a degraded core has resulted from our activities in conjunction with the TMI vessel inspection program. Through analysis of vessel samples along with examination of core debris retrieved from the lower head, it is hoped that we might better understand the conditions under which molten core debris can be quenched and retained in the vessel.

Our efforts during the past year related to severe accident analysis codes have focused on documentation and assessment of the existing models. No significant code development was undertaken. In several instances, support for previously developed codes was terminated after concluding that codes were no longer needed or were unnecessarily duplicative.

A more detailed discussion of the progress of research efforts on near-term SARP issues is provided in Enclosure I. A discussion of the ongoing activities related to selected long-term issues identified in the revised SARP is included in Enclosure 2.

Note that integral to the conduct of research under the revised SARP was the need to thoroughly review the planned research activities. In that regard SARP activities have been subjected to close scrutiny and are regularly evaluated by the Nuclear Safety Research Review Committee as well as the ACRS. The staff met with and briefed both the ACRS Severe Accident Subcommittee and the full committee on the revised SARP on April 5-6 and April 18, 1990, respectively. In its letter dated April 24, 1990, from Carlyle Michelson to Chairman Kenneth M. Carr, "Severe Accident Research Program," the ACRS expressed reservations and provided comments on certain elements of the SARP. Although the staff readily accepts critical review and routinely re-assesses the direction of research in order to identify needs and determine if efforts are being directed most efficiently, we disagree with some of the ACRS comments and contend that the revised SARP objectives and structures are appropriate. We are optimistic that it will provide the needed answers for operating LWRs, and also the evolutionary LWRs. The staff response to the ACRS was transmitted in a letter dated June 4, 1990, from James M. Taylor to Carlyle Michelson, "Staff Response to ACRS Letter on the Severe Accident Research Program (SARP)."

As noted, this paper is being provided for information and no Commission decisions are being requested in connection with this paper.

James M. Taylor
Executive Director
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Enclosures:
1. Near-Term SARP Issue Status
2. Long-Term SARP Issue Status

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Enclosure 1
Near-Term SARP Issue Status

Scaling

Since full-size severe accident experiments are not practical or feasible, analyses of severe accidents are mainly based on specialized models or on computer code calculations that are supported and verified by experimental data obtained from scaled-down test facilities. Implicit to this approach are two premises: 1) that the experimental data from such facilities are applicable and relevant to nuclear power plant accident conditions, and 2) that the specialized models or the computer codes have the capability to scale up processes from small-scale test facilities to full-scale plant conditions. Thus, both premises are concerned with the issue of scaling.

To address the scaling issue, RES initiated the Severe Accident Scaling Methodology development program. The objectives are to:

1) Ensure that models or computer codes used to resolve a severe accident issue have the capability to scale up processes to relevant nuclear power plant conditions.

2) Provide the scaling rationale and similarity criteria for facility design and test specifications.

3) Quantify the effects of scale distortions.

4) Integrate experiments and analyses by means of a methodology that is systematic, comprehensive, auditable, and practical.

5) Provide a proper balance between experiments and analyses to ensure cost-effective and timely resolution of a safety issue.

A technical program group has been formed to provide information to the Severe Accident Scaling Methodology (SASM) program. The group is composed of experts from universities, national laboratories, and industry to ensure inputs from a broad spectrum of technical sources. The staff is making the first application of SASM to the issue of direct containment heating. Our objective is to plan the next and subsequent experiments at the SURTSEY facility at Sandia National Laboratories consistent with the scaling methodology developed under the SASM program. It is anticipated that the application of the SASM to DCH will be completed in July 1990. A final report is expected to be issued at the end of FY 1990.

Depressurization and Direct Containment Heating (DCH)

Conduct of the research under this issue is intended to gather information needed to make an informed decision regarding the likelihood and the consequences of a high pressure melt ejection event.

The staff, over the past year, has continued limited research on DCH. The main activities have been confined to the preparatory planning of experiments incorporating the insights obtained from the SASM development program. It is through this process that we ensure that future tests better represent prototypic conditions associated with core melt phenomena as well as actual plant design characteristics (e.g., containment structures) which might influence dispersal of molten debris.

During FY 1990 the major accomplishments of the direct containment heating (DCH) research focused on the development of an experimental test plan with the supporting scaling analysis as discussed earlier, and the conduct of technology development tests which demonstrated the ability to conduct steam-driven DCH tests and provided information needed for determining the necessary instrumentation of planned future tests.

In December 1989, a DCH workshop was held in Annapolis, Maryland. It was widely attended by representatives from NRC, DOE, international organizations, national laboratories, industry (EPRI, FAI, PLG and various utilities) and universities. The major conclusions of this workshop were: 1) NUREG-1150 results suggest that DCH is significant but not the dominant contributor to risk for the specific plants studied; 2) scaling of initial conditions and boundary conditions are important if results are to be appropriately interpreted and used for reliable full-scale predictions; 3) plant-specific features such as cavity design and lower compartment structures may effect the likelihood for and consequences of DCH and therefore should be appropriately modeled; and 4) a data base to provide information on fragmentation and entrainment/de-entrainment phenomena and the impact of results on DCH is needed. These conclusions reaffirm our position that the SARP objective regarding DCH is appropriate and will provide the needed answers for operating LWRs.

BWR Mark I Containment Shell Meltthrough

For several years the NRC and industry have sought, through testing and analysis, to reduce the uncertainty surrounding the performance of BWR Mark I reactor containments during a severe core damage accident. Existing Mark I plants provide an adequate level of safety, stemming largely from the estimated low likelihood of core melt accidents. However, if core melt and reactor pressure vessel failure are assumed, the performance of the Mark I containment remains clouded by a wide divergence of technical analysis and judgement. Specifically, the Mark I containment has been judged to be vulnerable to two early failure mechanisms with high attendant consequences: early over pressurization of the drywell and drywell shell meltthrough. The NRC staff, with th endorsement of the Commission, is currently addressing the early overpressure failure mode of Mark I containments through its efforts related to installation of a hardened vent path. However, the case of drywell shell melt-through, proceeding from a station blackout event and loss of all core cooling, has not been resolved. The major reason for this is that the nuclear safety research community has not as yet reached agreement on the effectiveness of water to cool core debris and prevent or substantially delay drywell shell failure.

In an effort to divide this complex problem into smaller parts, the Office of Nuclear Regulatory Research recently sponsored an independent evaluation of this matter by Professor T. Theofanous of the University of California. Professor Theofanous' study, published as draft NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment," provides a comprehensive and integrated approach to the problem within a probabilistic framework.

The basic approach is to divide the problem into the major phenomenological components, to assign probability distributions to those phenomena, to determine the causal relationships, and then to synthesize the variables to estimate the probability of the outcome of interest, that is, liner failure. Using this systematic approach, it has been estimated that, for low-pressure vessel failure scenarios absent water on the drywell floor, drywell shell failure is a virtual certainty, with a probability between 0.63 and 1. However, and most important, the probability of drywell shell failure with water on the drywell floor was estimated at approximately 1×10^{-4} or less. The NUREG-1150 elicitations on this issue reported that for large debris superheats and large metallic content debris with no water on the drywell floor, the containment liner failure was a near certainty. This is consistent with the finding in NUREG/CR-5423. However, for the flooded drywell case, the NUREG-1150 elicitations indicate that the aggregate failure probability distribution is much higher than estimated in NUREG/CR-5423, although there was clear division of opinion among experts with four out of the six experts supporting a very low probability of melt through while two consider it highly likely even in the presence of water. The important factors that contribute to this low failure probability for NUREG/CR-5423 are: 1) melt superheats are initially low and short-lived once the melt interacts with coolant and concrete, and 2) the liner, acting as a heat transfer fin, cools the material as it reaches the liner and prevents melting. An extremely low likelihood of early drywell failure when there is water available has significant ramifications on consequence analysis, plant evaluations, and accident management considerations. If this study withstands the scrutiny of a broad peer review, we believe the results of this evaluation could lead to a resolution of the Mark I liner issue.

In February 1990, we distributed the report to some 18 distinguished members of the nuclear safety research community, a diverse group of individuals with considerable experience in this area. The peer group was canvassed for their opinions on four items of interest: 1) the technical merits of the analysis and the quantification of inputs, 2) appropriateness of the methodology, 3) interpretation of research cited in the report, and 4) the existence of other relevant work that may have been overlooked or neglected.

The staff is planning a workshop for July 23-24, 1990, at which individual peer review group members can discuss their comments. Through this process the staff and the report's authors will develop a better understanding of specific comments and the underlying bases. Following this workshop the final report will be issued along with a discussion and response to the comments and recommendations of the peer reviewers. We currently anticipate the final report will be issued in October 1990, and at that time, the staff will make its recommendation to the Commission.

Adding Water to a Degraded Core

The study of the effects of water addition to a degraded core is a central issue in the near-term focus of the SARP. This issue, unlike the issues related to early containment failure, may be viewed as investigation of phenomena pertinent to prevention of vessel failure, versus mitigation of a severe accident following vessel breach. In the development of the SARP the characterization of this issue generally inferred that we were emphasizing research that would allow us to better define the negative attendant consequences of water addition, e.g., hydrogen and steam generation, and containment pressurization, so that reactor operators could be more cognizant of concomitant effects of their actions. While we continue to believe it is important for reactor operators to have the best available information related to their actions, our attention over the past year has also been directed at this issue from another perspective.

The related question that also needs to be explored is that if melting proceeds, what are the circumstances in which a grossly degraded core can be quenched and retained within the vessel? A light-water reactor severe accident with failure to quench assumes core heat up, degradation, and meltdown. The accident proceeds in the core region until the accumulated core melt material leaves the core region and enters the water in the lower head of the pressure vessel. There may be several outcomes of the melt-water-structure interactions in the lower head,

1) the melt pour does not quench and fails the vessel promptly,

2) the melt pour is partially quenched, but the vessel fails later from vessel wall heat-up from the deposited melt layer,

3) the melt pour is fully quenched, reheats in 2 to 3 hours, and then fails the vessel, or

4) the melt pour is quenched and the vessel remains intact

These possible outcomes pose very different considerations for accident management and subsequent containment loadings. The issue is whether the lower head fails, and if it does, the timing of the failure and the characteristics of the corium discharge into the containment, in terms of the corium mass rate of flow, composition, and temperature.

Over the past year the staff has undertaken a number of activities to address the issue of water addition and core recovery. As noted, staff activities are now focused on establishing conditions for quenching debris in the lower head and establishing integral effects of water addition on the plant.

The staff concluded that in order to address the issue of concomitant effects it was necessary to evaluate proposals from a wide cross section of researchers. Thus the staff opted to solicit proposals under a Broad Agency Announcement. The deadline for proposals under the announcement was March 15, 1990. The staff has completed review of the proposals and is now

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On January 31, 1990, NRC contractors entered the TMI-2 containment and began the difficult task of cutting out the specimens. By March 1, the end of the 30-day time window for work in the containment, a total of 15 metal samples had been successfully removed from the pressure vessel wall. In addition, 14 damaged penetration nozzles were removed from the bottom of the vessel and 2 instrument guide tubes were removed from the core support assembly. The nozzles and guide tubes, which were removed by a mechanical cutting procedure, contained frozen core debris and provided unique opportunities for examination.

By examination of these samples, along with analysis of previously molten core debris retrieved from the lower head it will be possible to construct models for predicting the thermal response of the RPV lower head and penetrations. With such a tool it may then be possible to predict with greater assurance the behavior of the RPV for a variety of accident sequences.

While the information gleaned from the post mortem examination of the TMI-2 reactor is valuable, it is recognized that TMI-2 represents essentially one data point for confirming any models that might be applied to reactor analysis. In order to obtain more information on both lower head quenching processes as well as water addition to cool material above the support plate the NRC over the past year has negotiated with EPRI and the Joint Research Center (Ispra) to conduct a joint research program. This program will experimentally investigate quenching phenomena by pouring molten UO_2-Z_rO_2 mixtures into a deep water pool in an enclosed vessel. The staff and EPRI through their contractors have recently completed preliminary pre-test calculations to assist in the final design modification of the facility apparatus. This research will also address Long-Term Issue 4, "Fuel-Coolant Interactions."

Use and Status of Severe Accident Models (Codes)

As noted last year in the discussion of the SARP, the staff intended to concentrate its near-term efforts on severe accident analyses codes on the documentation and review of those codes already developed under the SARP. This emphasis on code review and documentation (as opposed to continuing code development) reflects recognition that at the onset of severe accident research it was necessary in certain situations to pursue analytical model development on parallel tracks. The necessity of this approach is derived from the fact that success in model development is not guaranteed. Thus it is prudent under specific circumstances to develop alternative options. At this juncture however, severe accident analysis has matured to the point that, at a fundamental level at least, it is possible to evaluate various codes to examine if the codes are adequate, require improvement, or may be archived.

As a result of ongoing code assessments the staff, during the course of this past year, has terminated explicit code support and maintenance for three severe accident analysis codes, HECTR, MELPROG and BWRSAR. The HECTR code was developed to perform calculations of hydrogen mixing and combustion within reactor containments. It has been concluded that other systems type codes, specifically MELCOR and CONTAIN, possess sufficient capabilities for assessing hydrogen challenges from severe accidents. Similarly, support for MELPROG has been terminated on the basis that core melt progression phenomena are more appropriately analyzed using SCDAP/RELAP. Lastly, after a thorough review, the staff has withdrawn developmental support for BWRSAR code, which was originally developed to model the details and specific nature of core melt progression in a boiling water reactor (BWR). BWR specific models developed under BWRSAR sponsorship will be subsumed into MELCOR or SCDAP/RELAP if appropriate.

During this past year the staff has completed development and documentation of our original integrated risk analysis code, the Source Term Code Package. The STCP, a collection of various codes coupled together, with the MARCH code serving as the cornerstone, will no longer receive developmental or maintenance support.

The principal tool for NRC sponsored analysis of severe accidents is the MELCOR code. MELCOR is an integrated systems risk analysis code intended to address the spectrum of severe accident phenomena including: reactor coolant system thermal-hydraulic response, core melt progression, in-vessel and ex-vessel fission product release and transport, engineered safety system performance, and containment loading. MELCOR may be used to model both BWRs and PWRs. MELCOR has been especially designed to facilitate sensitivity and uncertainty analyses. Its current uses include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications.

This past year has seen the widespread distribution of the MELCOR code. Initial distribution was made to ten domestic institutions, principally other national laboratories and a few universities, and to one foreign institution. Later releases through the fall of 1989 were made to other domestic institutions and an additional 14 foreign institutions participating as partners in the Severe Accident Research Program.

A significant amount of documentation has already been generated for MELCOR, especially during FY90. This documentation is organized into three categories for each MELCOR module: Users' Guides, providing instructions for building MELCOR input decks; Reference Manuals, describing the phenomenological models in detail; and Programmers' Guides, giving information related to the FORTRAN implementation of the models. In addition, a summary document describing MELCOR 1.8.0 has been completed, with publication expected by September.

A recent highlight of MELCOR user support activities has been the development and presentation of a users' workshop for the Severe Accident Research Program Partners. The first workshop, held in December 1989 was highly successful in its overall goal of quickly teaching new users how to run the code.
Plans for additional validation and verification of MELCOR have been formulated and are summarized in a report entitled, "MELCOR Quality Control

and Technical Assessment -- Preliminary Assessment Recommendations and Data Compilation." The ultimate objective of this strategy will be to validate important models, quantify their uncertainty, and develop recommendations for user input, nodalization and model parameters.

Consideration is being given to some further model development and improvements to the code as they may benefit NRC objectives, as the code becomes completed in the next few years. The contractor staff will identify their recommendations of need for improvements in the code considering: the adequacy of the code without the improvements, the benefit to the user of each improvement, the anticipated cost in terms of laboratory resources to accomplish the improvement, and the proposed priority for each improvement relative to others.

Most importantly, to assure that the code development has accomplished the objectives set forth by the NRC, an extensive peer review is being undertaken for RES by independent expert reviewers. The review will involve about one year to complete and is beginning in the summer of 1990. The reviewers will compare the capabilities of the code with requirements, consistent with the objectives of the Severe Accident Research Program and the guidance provided in SECY-88-147. This work is applicable also to L1, "Modeling Severe Accidents."

Enclosure 2

Long-Term SARP Issue Status

This enclosure discusses selected long-term issues, specifically L6, "Fission Product Behavior and Transport," and L2, "In-Vessel Core Melt Progression and Hydrogen Generation." Issues L1, "Modeling Severe Accidents," and L4, "Fuel-Coolant Interactions," have been discussed in Enclosure 1. The remaining long-term issues L3, "Hydrogen Transport and Combustion," L5, "Molten Core-Concrete Interactions," and L7, "Fundamental Data Needs," will be included in an upcoming revision to NUREG-1365 to be issued in the fall 1990.

Fission Product Behavior and Transport

In FY90, the VICTORIA code Mod1 was completed, including a draft of the user's manual. The VICTORIA code provides modelling of the fission product release and transport behavior for the reactor system. This code will be used to support the PHEBUS-FP program, an international experimental program regarding fission product behavior and transport.

The TRENDS code provides comprehensive modelling of iodine behavior for the containment. It determines distribution of the chemical forms of iodine as a function of time after an accident, and the amount released from the containment.

In response to a February 13, 1990 (M900109) Commission request for early completion of research regarding the chemical form of iodine (and other areas in TID-14844), the TREND code is being used to quantify the iodine chemical form and behavior (inside a containment) during accident sequences that do not breach the reactor vessel. The result will provide technical bases for staff proposals regarding changes to existing regulatory guidance on iodine chemical form delineated in Regulatory Guides 1.3 and 1.4. Under our current practice, the chemical composition of iodine is assumed to be constant over time. This influences the design of important fission product cleanup systems such as sprays and filter systems. Current understanding is that iodine is released into the containment primarily in the form of cesium iodide (possibly with some hydrogen iodide). After release, iodine could undergo a large variety of chemical reactions in the atmosphere. To estimate this iodine behavior, the TRENDS code is used to perform parametric studies by varying certain conditions including the pH of the sump and water pools in the containment. Completion of this effort is expected by the end of 1990.

Analysis of the third out-of-pile ORNL VI-3 test was completed, and the results are being published as a NUREG/CR report. This test is part of an experimental test matrix investigating parametric effects on fission product release. These tests have provided a substantial experimental data base regarding in-vessel radionuclide release under severe accident conditions. The results are being used to improve modelling of fuel fission product release in source term analysis codes. These improvements will lead to a better determination of the transport and retention of fission products within the reactor coolant system, transport to and within the containment, and subsequent estimation of the radiological consequence of severe accidents.

In-vessel Core-Melt Progression and Hydrogen Generation

In-vessel core-melt progression concerns the state of the reactor core from the start of core uncovery to reactor vessel failure. Melt progression determines the mass, composition, and temperature of the melt released upon vessel failure that may threaten the integrity of the containment. Melt progression includes the generation of hydrogen and attendant core heating from steam oxidation of the zircaloy cladding, and provides the core conditions that determine fission-product release and aerosol generation.

Significant new information on in-vessel core-melt progression has been obtained in the year since the revised SARP was issued, both from the NRC research program and from international cooperative research programs in which NRC participates. Currently, NRC has experimental programs underway in the Annular Core Research Reactor (ACRR) and in the Canadian National Research Universal (NRU) reactor as well as ex-reactor experiments. Considerable information is also being obtained: from the post-test examination (sectioning) and analysis of the results of previously completed tests in these facilities and in the Power Burst Facility (PBF); from results of the international OECD LOFT FP-2 test (Organization for Economic Cooperation and Development, Loss of Fluid Test, Fission Product-2); and from analysis of the results of the TMI-2 core examination. Important information, particularly for BWRs, is also being obtained from the German CORA ex-reactor fuel-damage test facility.

Post-test examination (sectioning) of the full 12-foot length fuel bundles from NRC tests in the Canadian NRU test reactor showed that partial blockages of frozen zircaloy melt formed, melted out, and reformed lower down in a sequential process as boildown of the bundle progressed. In two of these tests, which were held at high temperature for 30 and 60 minutes, respectively, 68% and 80% of the uncovered zircaloy in the bundle was oxidized by steam to produce hydrogen. An analysis was performed on the effects of metallic melt relocation and blockage formation on hydrogen generation in the four Severe Fuel Damage tests in the PBF test reactor as well as in the three high temperature NRU tests. These experiments showed that zircaloy oxidation and hydrogen generation are not cut-off near the onset of zircaloy melting as has been sometimes hypothesized. A NUREG/CR report is being issued on these results and corollary analysis. In these tests, moreover, over 80% of the total hydrogen was generated after the hypothesized cut-off was reached. Results of the joint international OECD LOFT FP-2 test, which had a large by-pass flow area around the 121-rod test fuel bundle, have provided significant additional evidence that such a cut-off does not exist.

New results have been obtained on melt progression in quite complex BWR core geometries. BWR cores are compartmentized by zircaloy channel boxes that contain the fuel rods and have cruciform boron-cardide control blades in half of the intersecting corners between the boxes. These results were obtained from the post-test examination of the test fuel bundle from the BWR Damaged Fuel Test DF-4 in the Annular Core Research Reactor (ACRR), and from the German CORA ex-reactor severe fuel damage test facility in our cooperative international research program. These results have shown that, in BWR core-uncovery accidents, intact core geometry starts to degrade at the relatively low temperature of 1250°C, which is about 200°C below the stainless steel melting point. These eutectic interactions occur at relatively low temperatures prior to the rapid heating of the core by steam oxidation of the core zircaloy and prior to zircaloy melting.

Significant results on reflooding a hot, severely damaged core have been obtained in the international OECD LOFT FP-2 test and also in the German CORA facility. During core uncovery and boildown, steam oxidation of the core zircaloy with attendant hydrogen generation produce a steam-starved hydrogen environment in the hot upper region of the core. The copious steam supply from core reflooding produces rapid oxidation, hydrogen generation, and heating of this core debris. In the large-bundle LOFT FP-2 test, most of the hydrogen generation and most of the fuel-bundle damage occurred during bundle reflooding before the highly damaged test fuel-bundle eventually became quenched. Previous CORA tests had given indications of this behavior.

The examination of the TMI-2 core has provided most of our current understanding of the late core-melt progression phase, which involves ceramic (fuel) material melting and relocation. Experiments have been started in the ACRR test reactor to expand this information base, particularly regarding the key phenomena that determine the melt mass, composition (metal/ceramic ratio), and temperature at vessel failure. The initial experiment has been performed regarding the characteristics of a growing melt pool in a ceramic rubble bed and its attack upon a supporting metallic crust across the lower fuel-rod stubs (the TMI-2 configuration). Temperature measurement and post-test examination (sectioning) of the test capsule are providing data for comparison with models of the processes involved.