POLICY ISSUE
(Information)

September 24, 1992

SECY-92-329

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: SEVERE ACCIDENT RESEARCH PROGRAM PLAN UPDATE
(DRAFT NUREG-1365 REV. 1)

Purpose: The purpose of this paper is to inform the Commissioners of the staff's update to the Severe Accident Research Program (SARP) which supports the tasks and objectives discussed in the staff's "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, and more recently the certification reviews of advanced reactors. The principal objectives of this paper are:

1. To identify severe accident issues that have been closed or are near completion.

2. To describe the major objectives and elements of the long term SARP.

3. To describe how the SARP activities relate to the Commission's policy, strategic goals, and other activities associated with closure of severe accident issues for existing plants as well as for evolutionary and advanced light water reactors.

CONTACT:
Farouk Eltawila, RES
492-3525

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

4. To describe how the SARP activities relate to the criteria for containment performance during severe accidents set forth in the draft Advanced Notice for Proposed Rulemaking on Severe Accidents.

Additionally, as a result of recent attention directed at the MELCOR code peer review and its findings (documented in the LANL report LA-12240, March 1992) we have provided separately, as Enclosure I to this paper, a summary of the RES response to the significant conclusions of that review. Disposition of the peer review recommendations along with other MELCOR research activities are part of the SARP nonetheless, and are also described in the overall plan.

Summary: Since issuance of the revised SARP (NUREG-1365) in 1989, significant progress has been achieved in a number of research areas. This progress, together with the evolution of research user needs related to advanced light water reactors, has necessitated update of the SARP reflecting those recent developments. This update to the SARP has been discussed with the ACRS and the NSRRC before submission to the Commission. Unless otherwise directed, the staff plans to continue to implement this update to the SARP.

Background: For the past 12 years, the NRC has sponsored an active research program on severe accidents in light water reactors as part of a multifaceted approach to reactor safety. In August 1985, the Commission issued a Severe Accident Policy Statement (50 FR 32138) in which the Commission concluded that, based on available information, existing plants posed no undue risk to the public health and safety and that there was no present basis for immediate action for any regulatory requirements for these plants related to severe accidents. However, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), the Commission was convinced of the need for both continuing research on severe accidents and a systematic examination of each existing plant to identify any plant-specific vulnerabilities to severe accidents. These systematic examinations are now being accomplished under the Individual Plant Examination (IPE) program.

In 1989, the staff revised the SARP to better focus the research towards resolution of specific severe accident issues of importance (i.e., those phenomena that could result in early containment failure; in particular BWR Mark I containment liner meltthrough, direct containment heating) as well as to improve the effectiveness of the severe accident code development program. It was also revised to support implementation of the staff's plan for closure of severe accident issues in accordance with SECY-88-147. This revision was extensively reviewed both in-house, as well as by our contractors, foreign users of the codes, and oversight committees. This revision was documented in NUREG-1365.

In addition to the Severe Accident Policy Statement mentioned previously, the NRC has also issued other guidance for addressing severe accidents. This guidance can be found in: (1) the Policy Statement on Safety Goals for Operation of Nuclear Power Plants, (2) the Policy Statement on Nuclear Power Plant Standardization, (3) 10 CFR Part 52, and (4) SECY-90-016 and its supplements.

Understanding severe accident phenomena and how they might affect plant performance should a severe accident occur is necessary to allow the NRC to evaluate the extent to which a plant has design features to both prevent severe accidents and to mitigate their consequences. The SARP is structured to provide that understanding.

Discussion: The purpose of this Commission paper is to describe how the long-term SARP is designed to improve our understanding of severe accidents and provide the technical support to the NRC staff to facilitate closure of the severe accident issues described in SECY-88-147 and SECY-90-016. In developing this SARP update, the staff recognized that the overall goal is to achieve an adequate level of understanding of severe accident phenomena and to reduce the uncertainties in predicting these phenomena to the extent practical, and sufficient to enable the staff to make regulatory decisions on severe accident issues. However, the staff also recognized that for some issues it may not be practical to attempt to reduce uncertainties further, and some regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

The SARP plan has produced valuable information, and the level of knowledge has increased such that regulatory closure of some of the severe accident issues can be made. Although regulatory closure can be made, it will be made with the knowledge that uncertainties in the phenomena associated with some of these highly complex issues still remains high. Therefore, the NRC will continue research on these issues. The level of effort will be commensurate with the practical level of uncertainty reduction expected to be achieved. The level of effort will also take into consideration the need to maintain of a level of expertise within the NRC and its research organizations to address severe accident issues that may arise in the future.

Major SARP Accomplishments:

The issues that have been completed or are near completion are: (1) the assessment of the probability of BWR Mark I containment shell (liner) failure. This program addressed the Mark I liner failure issue from a probabilistic basis and concluded that without water addition to the drywell, liner failure was a near certainty, whereas with water available to the drywell, such as via sprays, the probability of liner failure was extremely low. A final peer review will be conducted at the end of CY92 to determine the appropriateness of the approach and conclusions; the results of the final review will be reflected in the IPE reviews for BWR plants with Mark I containments; (2) the development of a severe accident scaling methodology (SASM) to guide the formulation of experimental programs and analytical methods has been completed and published as NUREG/CR-5809; (3) the completion of experimental and analytical research on fission product release and transport which culminated in the current staff efforts to revise the TID-14844 source terms; (4) the completion of current staff efforts to revise the experimental and the analytical research on core concrete interactions; (5) the completion of the experimental and analytical research to evaluate static or dynamic loads from hydrogen combustion or detonation at low temperature; and (6) the near completion of the TMI-2 vessel investigation program to investigate the condition and properties of material extracted from the TMI-2 reactor pressure vessel lower head, to determine the extent of lower head damage and the structural integrity margin remaining in the pressure vessel. These results will be used to perform scoping analyses of potential reactor vessel failure modes.

Finally, there are two major areas of the SARP for which substantial progress has been made over the past year: direct containment heating and MELCOR code assessment and validation. Using specific insights and general guidance from the SASM application to the direct containment heating issue, the staff has developed a coordinated, peer-reviewed program of prototypic integral tests (at different scales), separate effects testing, and analytical methods development and application to reactor analysis. Counterpart integral testing of reactor designs similar to Zion have just recently (June 1992) been completed at both 1/10th and 1/40th scale at Sandia National Laboratories and Argonne

National Laboratory respectively. Preliminary analysis and comparison of data shows good agreement between results at different scales and also indicates reduced containment loadings as a result of prototypic containment features which act to mitigate the consequences of a high pressure melt ejection. The focus of the DCH research program beyond CY92 will be on application of the general insights from the program to resolve the DCH issue for the other types of cavities and lower subcompartment configurations used in U.S. PWRs or, as necessary identifying the need for testing different designs.

As part of our SARP activities to develop and validate severe accident analysis computer codes, the staff undertook a program to conduct a broad-based, in-depth peer review of the MELCOR code. The peer review, documented in LA-12240, March 1992, was performed to guide both code development and validation activities. The recommendations of the peer review committee have either already been addressed by code modifications completed since the peer review, or are planned to be addressed, as appropriate, through corrective actions and improvements within the next year. Additionally, the staff has recently instituted the MELCOR Cooperative Assessment Program to broaden the code assessment base by including those voluntary efforts of the code user community.

While we have completed the major research activities on the hydrogen and fission product release research programs, residual research efforts on each of these programs are still ongoing.

Hydrogen research to date has been limited to tests involving hydrogen-air-steam mixtures under ambient or relatively low (100°C) temperature conditions. While this is sufficient for most severe accident analysis, conditions can be postulated in which high temperature hydrogen-air-steam mixtures can exist. We have entered into a joint agreement for cooperative research with the Ministry of International Trade and Industry of Japan and the Nuclear Power Engineering Center (NUPEC) to conduct high-temperature, high-speed combustion research at the Brookhaven National Laboratory. This data is intended to supplement the existing data base with information about combustion phenomena at elevated temperatures. By the nature of the cooperative agreement, this is a very cost-beneficial program for the NRC.

With regard to fission product release research, the recent French low power/shutdown PRA indicates that risk during these conditions might be significant (i.e., of the same order of magnitude as full power risk). The NRC-sponsored PRA to study low power/shutdown risk will be completed in January 1993. It is believed that during a postulated severe accident during plant shutdown conditions with the reactor vessel head is removed, air might ingress into the overheated core, and could result in a larger release than under air-starved conditions typical of severe accidents that are postulated to occur at power. This program will be concluded shortly with the completion of air-rich test to cover this portion of the severe accident spectrum.

Major SARP Ongoing Activities

Over the next several years, SARP will focus on two phenomenological issues and on code development, validation and assessment. While we expect to complete the major severe accident experimental programs within the next 2 to 3 years, the development, validation and assessment of the major severe accident codes (SCDAP/RELAP, CONTAIN, and MELCOR) will continue to incorporate the results of experimental data being generated worldwide and to incorporate recommendations from the peer reviews that will periodically be performed to assess the adequacy of these severe accident codes to perform the intended analyses. The major two phenomenological issues are: (1) core melt progression; and (2) fuel-coolant interactions and debris coolability.

The core melt progression research impacts the containment integrity issue. However, as described in the SARP update, our approach to assessing containment integrity is focused on the particular processes relevant to specific containment challenges, assuming these processes can be adequately characterized for the more likely severe accident sequences. The benefits of the core melt progression phenomena research will be related primarily to reduced uncertainty in risk estimates, and improved understanding of the melt progression phenomena. An additional benefit is related to accident management, in particular the ALWR, where a better characterization of the core melt progression is needed to assess the efficacy of the measures proposed to arrest the core melt accident within the vessel.

The fuel-coolant interactions and debris coolability research is focused to address issues of interest applicable to the evolutionary LWR including the AP600 and SBWR. The AP600 design incorporates a flooded cavity to reduce the likelihood of vessel failure, and, should the vessel fail, to reduce the likelihood of core concrete interactions. The SBWR employs a passive containment flooding system to flood the drywell to a level above the top of the active fuel and is intended also to flood debris in the cavity following vessel failure to prevent core concrete interactions.

In addition, the information from this SARP is useful to the NRC in preparation of the Part 50 rule change addressing severe accident criteria applicable to future LWRs. This rule change is the second phase of the effort to decouple siting from plant design. Information on the following technical areas is expected to be useful for this rule change: hydrogen control, melt progression, core debris ex-vessel spreading and coolability, direct containment heating, fuel coolant interaction, and containment bypass. The research program is focused to provide the information needed to assess particular plant features incorporated to prevent or mitigate the consequences of severe accidents.

<u>Coordination</u>: We have discussed the SARP update with the Severe Accident Subcommittee of the Advisory Committee on Reactor Safeguards on May 27, 1992. Due to the extensive scope of the SARP update, a second ACRS Severe Accident Subcommittee meeting was held on June 25, 1992, to cover the remaining issues of the SARP update that were not covered during the May 27, 1992, meeting due to time limitations. In its letter to Chairman Selin dated August 18, 1992, the Committee found the updated SARP Plan described in draft NUREG 1365, Revision 1, a noticeable improvement over previous plans that they have reviewed. The Committee also found the plan to be well written, the goals and objectives of the individual projects are more clearly stated than they have seen in the past, and the descriptions of the proposed research are generally clear and specific. Enclosure II to this SECY paper addresses the specific recommendations in the August 18, 1992, ACRS letter.

We have also discussed the SARP update with the Severe Accident Subcommittee of the Nuclear Safety Research Review Committee (NSRRC) on June 2, 1992. In its letter dated August 11, 1992, from David Morrison, Chairman, NSRRC to Eric Beckjord, Director, Office of Nuclear Regulatory Research (Enclosure III), the Committee agreed with the programs underway and endorsed the staff's use of peer review to reach consensus and resolution of technical issue. The NSRRC Committee also was impressed with the Office of Nuclear Regulatory Research management of the Severe Accident Research Program as reflected in the revisions to NUREG-1365, the use of and response to peer reviews, and the extensive international cooperative programs.

The report also has benefitted from reviews by researchers doing work on severe accidents at the DOE National Laboratories, from industry representatives, from members of the academic research community and the Office of Nuclear Reactor Regulation.

Resource Commitment: In preparing this SARP update, the staff assumed a relatively constant level of funding for the next four years as projected in the Five Year Plan. While it is expected that some issues will be closed, the funds originally allocated to these issues will be redirected to accelerate closure of other issues or to address new issues identified as a result of the staff review of ALWR design-specific features to preclude or mitigate severe accidents. It should be noted that the staff is leveraging its resources by participating in international programs such as the FARO fuel-coolant interaction research being conducted in collaboration with the Commission of European Communities (CEC) at the Joint Research Center in Ispra, Italy, the PHEBUS fission product release and transport project sponsored by Commissariat a l'Energie Atomique (CEA) and CEC, the TMI-VIP program sponsored by the OECD Nuclear Energy Agency, and the NUPEC-NRC bilateral agreement on high-temperature hydrogen combustion research. The NRC is also deriving substantial benefits from our cooperative agreement on severe accident research with the Russian Academy of Sciences, in collaboration with the Russian Science Center (I.V. Kurchatov Institute). Specific research being performed in Russia under this agreement addresses the issues of hydrogen combustion and distribution, in-vessel core melt retention and core concrete interactions. These programs effectively utilize the considerable scientific experience and capabilities as well as the physical facilities of the Kurchatov Institute to supplement our severe accident data base. This international collaboration, in addition to obtaining data at modest costs without spending the large capital costs associated with building and operating these facilities (e.g., $150M U.S. for the PHEBUS project), will enable the staff to improve its understanding and reduce residual uncertainties associated with severe accident issues and maintain the level of expertise needed to address issues that may arise in the future.

The staff intends to advise the Commission periodically on the SARP progress toward achieving closure of severe accident issues and the status of ongoing programs as part of the semiannual update that is prepared to inform the Commission of the status of the implementation plan for closure of severe accident issues (SECY-88-147).

James M. Taylor
Executive Director
for Operations

Enclosure I: MELCOR Review

Enclosure II: Staff's Response to ACRS
Letter dated August 18, 1992, from
David Ward to Chairman Selin

Enclosure III: NSRRC Letter dated
August 11, 1992, from David Morrison
to Eric Beckjord

Attachment 1:
Provides Severe
Accident Research
Program Update (Draft NUREG-1365 Rev 1)
(Commissioners, SECY, OGC only)

DISTRIBUTION:
Commissioners
OGC
OCAA
OIG
IP
OPP
REGIONAL OFFICES
EDO
ACRS
SECY

ENCLOSURE I

MELCOR REVIEW

(PEER REVIEW COMMENTS AND STAFF ACTION)

In mid 1989 the MELCOR code was, for the first time, made available to the general research community. It was with our full understanding of the incomplete state of the code that it was distributed. The code release agreement required to be signed by prospective recipients explicitly stated that the code was not complete. It is for precisely this reason that the code was not placed in the public domain via the National Energy Software Center. Prior to this time the code was made available only to NRC contractors. In 1989, however, we determined that a broader release of the code, albeit restricted, would be beneficial to the NRC for a variety of reasons.

Expanding the user base of a code accelerates identification of code problems in that the number and type of analyses performed expands proportionately. Feedback from a wider user community provides a mechanism for code assessment without investment of staff resources. Critical review of existing models and identification of important missing models is also promoted through the process. In point of fact much of what is accomplished through peer review can potentially be achieved through code user interactions.

It should be pointed out that for FY89 virtually all code development was halted while we sought to identify exactly what was needed to be done to improve the code. The code developer (SNL) and other NRC contractors had provided recommendations for code improvements but we were also interested specifically in comments from a broader community.

The feedback from the user community was quite valuable and was passed along to the peer review group. This crossflow of information between the user community, NRC and the peer review committee in our view contributed to the thoroughness of the code review.

Notwithstanding our electing to more widely disseminate the MELCOR code, we submitted the code to an extensive formal peer review in mid 1990. The reasons for this decision include consideration of the extremely broad scope of MELCOR calculations - from RCS thermal-hydraulics to offsite fission product releases. As a result of this broad scope of the code's calculational capability, a multi-disciplined review team is essential for such an undertaking. It is RES policy to submit all major new codes to such a review. In part, this practice, at least for MELCOR, reflects recognition of the large uncertainties surrounding severe accident phenomena. Uncertainty over phenomenological behavior and differences of opinion between researchers ultimately find their way into the discussion of code modeling. For example, industry has postulated that corium may be cooled ex-vessel by providing a specified spreading area (EPRI ALWR debris coolability criterion). The MELCOR code was criticized by the peer review group for not having a model which would simulate the behavior postulated by EPRI, when in fact there was virtually no data until very recently (April 1992) which might support such a model. In other instances work was already underway to address shortcomings corroborated by the peer review (e.g., incorporation of models for DCH, natural circulation, and ice condenser performance). It was our objective and intent to use the peer review assessment to guide the future development of MELCOR.

As identified in the SARP (Section 2.4) the NRC has adopted a tiered strategy of code development wherein MELCOR represents the top or first tier, i.e., MELCOR is intended as a PRA code, its models are in many cases simplified with parametric capability. This is a deliberate approach that was taken that reflects the uncertainty of severe accident phenomena. Modelling approaches for more classical calculations are also often based on simplified approaches in order to make the code faster running or less unwieldy. It is not critical at this stage that a full plant code be capable of calculating in a mechanistic manner the behavior associated with severe accident phenomena; it is more important that the code have the capability to calculate a range of outcomes for phenomena which dominate plant response. In areas or circumstances where more detailed calculations are of interest, the more mechanistic codes such as VICTORIA (for fission product release, transport and deposition), IFCI (for fuel coolant interactions) or CONTAIN (for containment loads and fission product behavior) may be used. It is with this underlying precept that the staff is evaluating the peer review comments and developing plans for remedial actions.
The following is a summary of the disposition of those major peer review recommendations.

MELCOR Numerics: The Committee concluded that code numerics were a primary source of concern regarding the technical adequacy of the code, and that correction of the MELCOR numerics problems should be considered to be a high-priority activity.

This issue was first brought to our attention early in 1991, by our contractor at BNL and our international users at HSK in Switzerland. This information was subsequently brought to the attention of the MELCOR peer review committee. We agree with the recommendation of the peer review and in fact we redirected SNL in the summer of 1991, to devote resources to fix the numerics problems.

At this point we believe we have corrected the significant numerics problems identified through the end of 1991. As we continue to benchmark and apply the code, other issues will arise and indeed some new numerics problems have been identified which we are correcting.

Models Missing from MELCOR Version 1.8.1: The Committee concluded that models for the following phenomena, not then currently modeled, should be given high priority. The NRC and the code developers were already aware of each of these needed models and had already initiated actions on some, and the committee was aware of that. The committee's thorough review of the needs in these areas provided valuable corroboration of the importance of placing this work on high priority. We agree with the first three items and they were already planned for FY91 (which the committee learned about in their review), the remaining issues are planned in FY92 and FY93.

- o PWR primary system natural circulation in components with countercurrent flows: We agree with the need for natural circulation capability and a FY92 task addresses development of the capability to calculate in-vessel natural circulation. The existing hydrodynamics package in MELCOR can handle the primary system natural circulation and it was thought at the time of scheduling the FY91 development work that it could also handle the steam generator natural circulation. An assessment task had been scheduled to compare MELCOR results with the FLECHT-SEASET test results which would demonstrate that capability. However, difficulty developed with fluid crossover at the top of the steam generator tubes. Subsequently, a correction was scheduled for FY92 to rectify the problem. That work is now nearly complete and will be finished this fiscal year.

- o High-pressure melt ejection and direct containment heating: We agreed with the committee and had already scheduled the incorporation of a model for DCH in FY91-92. This work was completed in April 1992.

- o Ice condenser: We agreed with the need for this capability and had already scheduled the work for FY91. It was completed in FY91 and is now scheduled for technical assessment in FY92 using data from a PNL ice condenser test series.

- o Non-explosive interactions between debris and water: We agree with the need for providing heat transfer from debris to water as the core debris relocates from the core region to form a debris bed in the lower plenum of the reactor vessel. As such we have scheduled work which will be completed this fiscal year. We also agree that it would be desirable to model the heat transfer from a debris bed formed below the reactor vessel and the contractor has been asked to consider this in FY92, but no specific model has been decided upon at this time and resources are not yet identified to add this capability. Under the ACE consortium, an analysis group is being formed to develop models for debris coolability.

- o Fission product vapor scrubbing: We agree that it would be desirable to add fission product vapor scrubbing capability into MELCOR and this is anticipated for FY93. This capability might be in the form of a user specified DF that is supported by experimental data.

- o Additional reactor coolant system fission product deposition processes, and

- o Fission product reactions with surfaces: There may be some enhancement to the accuracy of the calculated MELCOR source terms by inclusion of certain models to handle these effects and the contractor has been asked to make a recommendation to the NRC on this in FY92. Depending on the recommendation and the anticipated measure of improvement to MELCOR, a decision will be made in late FY92 as to the resolution of this concern. Consideration is being given to parametric treatment of the phenomena.

Existing MELCOR Models Needing Revision: The Committee's review of the MELCOR phenomenological packages identified individual models that were of concern. The Committee considered which models should receive priority attention using as screening criteria: (1) the time of containment failure, and (2) the magnitude of the source term. The Committee recommended the following items be given the highest priority among models needing upgrading. The first three items were already planned for the FY91 or FY92 workscope (which the committee learned about in their review) and the last item should be addressed within a year.

- o An evaluation should be made to determine whether the water condensation/evaporation model used in the Hydrodynamic Behavior Package is implemented adequately: Concurrent with the recommendation to evaluate this model, an assessment task was being pursued at SNL using the data from the LACE LA4 experiments. That assessment did in fact cover the behavior of aerosols in a condensing environment and there were a series of sensitivity studies performed with the code to compare with the data. The implementation of the model of concern to the peer reviewers was thus checked-out and the performance of MELCOR was confirmed. The results of this work, however, were not reflected in the peer review report. The report on the assessment (SAND-91-1532) was printed and released in September 1991.

- o Inconsistencies in treatment of chemical reactions between CORCON and VANESA should be resolved, and improvements should be made to the CORCON/MOD2 phase diagrams: We agree with the peer reviewers and have scheduled an upgrade to the MELCOR CORCON (and VANESA) modeling which will involve implementation of the proposed CORCON MOD3. This will provide improvements to the phase diagrams and remove the inconsistencies in chemical modeling.

- o The model for condensation in containment (mass transfer) should be revised: We agree to consider revising the model for condensation in the presence of noncondensibles. However, we are still investigating the need for this revision, including following-up discussions with the peer reviewers to assure we have a detailed understanding of the perceived model shortcomings and the potential for improvement. Assuming the proposed improvements will have a significant enhancement for the MELCOR capabilities, the model will be implemented in FY93.

- o The pool scrubbing model should be improved: According to one set of calculations done at BNL, the pool scrubbing model in MELCOR calculates decontamination factors that are quite low when compared with the results of an earlier code, even though the MELCOR model was largely derived from the previously available model. As stated in the peer review report, there may be an implementation error in MELCOR, although any such error has not been located. We anticipate that an additional assessment effort on this question as a part of our expanded MELCOR assessment will provide further clarification on the decontamination factors. When this has been completed (likely within the next year) we can better address whether changes need to be made in MELCOR.

Need for Expanded MELCOR Assessment: The Committee had concluded that the ability of MELCOR to calculate severe accident phenomena was not sufficiently demonstrated. Such a demonstration would be based on a documented collection of (1) sensitivity studies, (2) benchmarking activities using experimental data, and (3) code-to-code assessments. The Committee concluded that an expanded assessment program should be pursued at a high priority. The NRC agrees with the need for code assessment and validation but has limited the number of assessment calculations until the numerics issue is resolved in order not to waste efforts in using the uncorrected code.

In addition, we are 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.

Documentation: The peer review committee found the body of existing MELCOR documentation represented a significant and positive accomplishment. However, the Committee was concerned about several aspects of the documentation. With respect to the code reference manual, it determined that the level of detail was less than desired and recommended that careful consideration be given to producing a MELCOR-equivalent of the "Models and Correlations" document prepared for the various NRC-sponsored thermal-hydraulics codes. With respect to the code user guides, the Committee recommended that a structured and ongoing process of collecting, documenting, and distributing practical user guidelines to the MELCOR user community be developed and executed.

While it appears that some improvements to the MELCOR documentation are warranted, a balance must be achieved between comprehensive, but unwieldy, documentation that is suited for critical review of the code and limited documentation suitable for general users. A new set of documentation equivalent to the TRAC code "Models and Correlations" document would be highly resource intensive and somewhat duplicative of information already available in the manuals. Nevertheless, some resources are directed, primarily in FY93, to upgrade reference manuals to cover the most critical needs. Further, development of a practical user guidebook for MELCOR will involve a small multi-year effort as insights from user applications are assembled and as they become available on a periodic basis. We expect to have input to this from the MCAP.

ENCLOSURE II

Response to ACRS Comments
Letter to Chairman Selin dated August 18, 1992,
"Subject: Severe Accident Research Program Plan"

Mark I Liner Failure

The ACRS raised a valid concern about the ex-vessel steam explosion in the event that water is present on the containment floor prior to vessel failure and debris relocation. Our preliminary investigation of this issue indicates that steam-explosion-induced containment failure is not possible. We intend to discuss this and provide the basis for this conclusion in the final closure report of the Mark I liner failure issue.

Regarding the comment on plausible later containment failure as first identified by Dr. S. Hodge of ORNL, we would like to point out that the NUREG/CR-5423 document considers two possible melt release scenarios that have been judged by the peer reviewers to provide a reasonably conservative envelope of melt release, zirconium content, and melt superheat for the Mark I liner analysis. Dr. Hodge's letter indicates that in the long term (beyond 12 hours after vessel breach), the rest of the core and structural debris will be drained from the reactor vessel, and debris height will rise above the water level leading ultimately to failure of the liner on a long term basis (i.e., duration longer than considered in the NUREG/CR-5423 document). We consider those conditions to be beyond the scope of the study which was to examine the likelihood of early containment failure by liner thermal failure. All other failure modes, e.g., overpressurization by noncondensable gas generation or steam generation, are to be addressed by utilities in the conduct of the IPEs (SECY-090-023). Finally, we fully concur with the ACRS conclusion that further work on the Mark I liner failure issue is not warranted.

Chemical Form of Iodine Released to Containment

The ACRS questioned how the results of the recently completed work on the chemical form of iodine released into the containment (NUREG/CR-5752) will influence calculated risk of existing plants or how the information will be used in the review of the IPEs. The ORNL information that the major form (95%) of iodine released into containment will be CsI, a particulate, has already been included in the severe accident analyses used by licensees in evaluating the backend, or level 2, of the IPEs. These analyses have included those performed with MAAP, and STCP (through analyses for NUREG-1150). The ORNL information that the remaining 5% of the iodine will be HI and I_2 is not included in any code. However, the search for containment failure vulnerabilities by the IPE process is not expected to be influenced by the omission from the codes of this ORNL level of volatile content, because the effect on the release of iodine is small (Section 4.13 of NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms", July 1986). Therefore, the presence of the remaining 5% both volatile and particulate forms of iodine is not likely to change the estimated level of risk at current plants. It also may not allow elimination of equipment such as charcoal filters (whose design was based on 95% volatile iodine), but might allow reduction in some such equipment. This will be considered in the context of 10 CFR Part 50 changes.

Direct Containment Heating

The ACRS noted that in many of the PWR PRAs, including two of those treated in NUREG-1150, containment by-pass is the risk-dominant failure mode, and DCH issue resolution will not have a significant effect on the estimated risk or on risk uncertainty for these plants. While we agree with the ACRS comments, it is worth mentioning that the by-pass accident is risk dominant not because it is frequency dominant, but because the consequences are calculated to be high. The reason the consequences are high is that the codes that were available, when the NUREG-1150 study was undertaken, for analyzing by-pass sequences had a conservative description of the aerosol behavior that takes place as aerosols are transported through the break in the by-pass sequences, be it through the check valves in the V-sequence or the steam generator tubes in SGTR accidents. When more realistic aerosol physics are taken into account in these analyses, the magnitude of the radionuclide release will be reduced. Since the issuance of NUREG-1150, NRC has developed the VICTORIA code to obtain a more realistic estimate of the radionuclide releases associated with by-pass accidents. Most likely, the results from such analyses may show that the risk associated with by-pass accidents is smaller. Absent identification of the dominant accident sequences for all operating plants, technical resolution of DCH is needed; and as the ACRS indicated, resolution of the DCH issue appears to be within reach.

Hydrogen

The Committee has some concerns regarding the effects of hydrogen detonations on AP600 containment and other issues related to hydrogen distribution and control. It should be noted that our presentation was focused on the research program that is being carried out by the Office of Nuclear Regulatory Research and did not address any plant-specific analysis, particularly the AP600 where the staff review had not been formally initiated. The following section presents our response to specific ACRS concerns on hydrogen.

Detonation Loading

The problem of structural loading due to detonations has to be considered in view of the very low likelihood associated with these events. Detonation of the entire hydrogen content of the containment structure is a very unlikely event since it is very difficult to initiate a detonation in the hydrogen-air-steam mixtures expected within the main containment volume under severe accident conditions. Since the consequences of detonation may be severe, and detailed plant specific loading and structural analysis are often required, our research has emphasized the formulation of criteria for initiating and sustaining detonations so measures can be taken to eliminate them rather than analyzing their consequences.

The NRC is aware of the possibility of shock and detonation reflections within the containment, and previous NRC-sponsored studies (NUREG/CR-3719, NUREG/CR-1762) on detonation propagation have included the effects of shock and detonation diffraction within the complex geometry of an ice-condenser plant and a generic BWR design. Although the detonations were initiated with a simple spherical or planar shape, the resulting wave pattern naturally evolves into a complex shape due to the interaction of wave with the containment. These two-dimensional computations have demonstrated that wave reflection and interaction processes can generate pressures up to 80 bars. However, these pressures are very localized in space, very short in duration, and depend strongly on the initiation location and geometry of the containment. Clearly, characterizing localized loads is a formidable problem for generic severe accident analysis.

Given the intrinsic uncertainties in accident analyses, the benefits of three-dimensional computations are dubious. Even more important than a highly refined gas-dynamic analysis is a detailed structural analysis. In order to understand the consequences of these transient and localized loads, detailed structural response computations must be carried out in addition to the gas-dynamic simulation of the detonation itself. Such structural response computations have not been done in our evaluations for either existing or proposed plants to our knowledge.

No specific detonation studies have been carried out on the AP600 configuration, but the current NRC research program on hydrogen combustion is examining the generic issue of detonation and detonation loads on structures. If accident sequences that are associated with a significant detonation hazard are identified, then plant-specific analyses could be carried out.

Plant-Specific Issues

The distribution of hydrogen, the location of igniters, the effectiveness of igniters in various atmospheres, and the role of passive cooling systems in producing localized concentrations of hydrogen are all plant-specific issues that will be examined through the certification review process. Extensive prior work on deliberate ignition systems (EPRI NP-3878, NUREG/CR-4138, NUREG/CR-2486, NUREG/CR-3468, NUREG/CR-3273) has demonstrated that under the conditions of severe accidents in the AP600, igniters will be effective unless the mixture is inert. As mentioned subsequently, the only exception is the possibility of a local explosion due to flame acceleration or transient jet initiation.

Enhancement of the hydrogen concentration by the heat removal system is certainly an issue and should be examined as part of the review process. A similar issue was identified and studied for the Sequoyah ice-condenser plant (NUREG/CR-1762). The effect of hydrogen concentration on heat removal is also an issue that will be examined as part of the review process.

How Likely is a Detonation?

The likelihood of a detonation depends strongly on the mixture composition, temperature, and pressure. Further, the source and type of ignition and the degree of confinement by the compartment or containment play important roles. These issues have been extensively examined in the last decade (NUREG/CR-5525, NUREG/CR-5275, NUREG/CR-4961, NUREG/CR-4905).

For typical severe accident conditions with a uniform overall hydrogen concentration of approximately 10%, a detonation will be very difficult to initiate or propagate as a result of steam dilution even in the absence of igniters. The only credible mechanism for initiation is through transition to detonation caused by a hot turbulent jet or an accelerating flame. However, even this mechanism is unlikely to result in a detonation in a lean, steam-diluted atmosphere. A local region of the containment that is rich in hydrogen and/or with a low concentration of steam might undergo a detonation-like event (local explosion). The possibility of the existence of such regions will be a subject of the certification review. The current NRC research program is examining the issue of local explosions and the initiation of detonations by jet flames.

Fuel-Coolant Interactions

On the issue of fuel-coolant interactions (FCI), the Committee questioned whether the information produced from the small programs supported by the NRC will resolve the issue. We are aware that previous attempts to resolve this highly complex issue were tinged with controversy, but we are optimistic that the NRC FCI research program, will address the fundamental aspects of the FCI issue and will result in a regulatory closure of this issue. Our approach is consistent with the recommendation of the Steam Explosion Review Group (NUREG-1116), which indicated that the conditional probability of α-mode failure is small and recommended additional work on the amount of fuel-coolant mixing and the explosion yield. The model that will be used for closure of the FCI issue is characterized by the precept that, except for the quantity of the melt and core support plate failure area, all parameters are related, and the sequence of events is represented as combinations of causal relationships. Each causal relationship can be dealt with on its own, allowing experts to continue refining respective portions independently of each other. Four efforts are currently underway to refine understanding of these causal relationships. The first [at the University of California, Santa Barbara (UCSB)] is to examine the interface transfer laws in three-phase systems, with one phase in film boiling. The second effort, also at UCSB, is to examine the integral aspects of the premixing process with emphasis on the performance of the three-fluid modeling approach. The third effort, at the University of Wisconsin, addresses the thermal-energy-to-mechanical-energy conversion to ensure that no major uncertainties due to scale-up are ignored. Finally, the fourth effort conducted at the FARO facility in Ispra is to study the fuel break-up phenomena during premixing and to quantify the steam and hydrogen generation rates. Results of the first effort have been reported at the 1992 National Heat Transfer Conference, San Diego; the second effort is near completion and will be published soon. The th[illegible] effort is ongoing and preliminary results will become available in early 1993. For the fourth effort, the FARO experiment, to be carried out in early 1993, will provide data adequate for that purpose. The recently completed NRC program at UCSB (the second effort) is already revealing some promising results (e.g., the key hypothesis in previous quantification of steam-explosion-induced containment failure was the depletion of the liquid coolant in the explosion zone). This water depletion phenomena seems to be firmly established based on the results from UCSB. We intend to have a peer review of the new information prior to pursuing the need for additional major research in this area.

In-Vessel Core-Melt Progression

The ACRS suggested "that the models that result from [in-vessel core-melt progression research] should be taken as representing only one possible severe accident progression. Future severe accidents, if they occur, may take as unexpected a course as those few that we have experienced. Thus predictions of their course and consequences with models based on limited past experience may be misleading." The ACRS also asked "how typical is the TMI-2 accident, even for a PWR, and how well is it understood?"

The melt-progression research is directed toward application over a broad range of severe accident conditions. A major part of the research is to determine the range of applicability of the blocked core-melt-progression phenomenology of the TMI-2 accident, which appears to be large, and to provide a technical basis for the more general application of this phenomenology. Apparently these considerations were not made clear in the melt-progression presentation at the Severe Accident Subcommittee meeting.

The core conditions found in the TMI-2 core examination are consistent with the behavior observed in essentially all the integral core-degradation tests that had been performed in the tests in PBF and ACRR, in the LOFT FP-2 test, and in the German CORA ex-reactor tests. These results were all for the "wet core" conditions of coolant boildown, and not for the "dry core" conditions resulting from the depressurization blowdown in U.S. BWRs. This observed behavior was the melting, relocation, and freezing of the unoxidized metallic zircaloy and control rod materials in the process of producing a core blockage across the fuel rod stubs.

The development of this blocked core sequence in essentially all of the wet core integral tests indicates that this sequence has general applicability in unrecovered as well as reflooded accidents. However, there are two major questions on the range and the details of the applicability of these phenomenological results of TMI-2 to severe accidents in general. The first question is related to dry core conditions, following the actuation of the automatic depressurization system. It has been hypothesized that for BWRs, metallic melt will drain to the lower head rather than form a core blockage as happened in TMI-2. However, whether or not this actually occurs is uncertain. The second question relates to the threshold and the location of meltthrough of the pool-supporting crust system in a blocked core sequence. Two major experiments in the melt-progression research and associated modeling program are directed at obtaining phenomenological information on these two questions for application to the broad range of severe accident sequences (unrecovered and reflooded), in addition to the TMI-2 accident sequence itself.

The ACRS also commented that it "believed that additional fundamental separate effects experiments are needed to better define the crusting behavior and the thermal hydraulics associated with molten pool conditions." As discussed in the SARP update, we will thoroughly peer review the melt progression program in light of the results that will be obtained this fall from the ongoing experimental program. Based on the results of the peer review, we will determine whether or not additional experiments are needed to reduce the uncertainties in late phase melt progression any further. The ACRS recommendations will be presented to the peer reviewers.

Lower Head Failure Analysis

The ACRS commented that "lower head failure analysis (NUREG/CR-5642) of the TMI-2 vessel should be of considerable value if it can be shown that what happened there has general applicability." As a point of clarification, the objective of the work reported in NUREG/CR-5642 is to present generic models of reactor vessel lower head failure analyses for the full range of reactor vessel designs presently used in the U.S. commercial nuclear power plants and for a wide range of corium melt compositions and characteristics. These models are being applied in the TMI-2 Vessel Investigation Project (VIP) to analyze the TMI-2 reactor vessel margin to failure. A separate report on the TMI-2 analysis will be issued at the completion of the VIP.

The ACRS suggested that further attention be given to "the uncertainties or the contributors to uncertainty in the results of the lower head failure analysis." Several peer reviewers of draft NUREG/CR-5642 also felt that additional work was needed to address uncertainties in modeling assumptions. We agree with these comments, and the final report will be revised to clearly discuss the assumptions, range of applicability, and uncertainties in each of the failure models and how these uncertainties affect model predictions.

As was stated in the ACRS letter, "it was reported to us that SCDAP/RELAP5 still does not provide a good estimate of lower head temperature rise [during a severe accident]." Because of the lack of sufficient experimental data, severe accident codes, including SCDAP/RELAP5, have not been validated for the late phase of core melt progression (i.e., ceramic material melting and relocation, blockage formation and meltthrough, melt release from the core to the lower plenum). These late phase melt-progression phenomena play a vital role in estimating the lower head temperature rise and the mode and timing of vessel failure. Section 2.2 of NUREG-1365, Revision 1, discusses NRC's planned ex-reactor and melt-progression experiments that will provide improved understanding of core blockage vs. melt-drainage behavior, and metallic and ceramic crust behavior, respectively. Results of these experiments will be used for further assessment and validation of SCDAP/RELAP5 and MELCOR, which should in turn lead to improved estimates of the TMI-2 accident.

Review of Severe Accident Codes

With regard to the ACRS question on the staff plans to use MELCOR in evaluating IPE results, there are no specific plans to use MELCOR in the review of IPEs. In general, the review of IPEs has been focusing on a review of the process used by the licensee, to assure that the process would be capable of meeting the objectives of Generic Letter 88-20. The level of review planned to accomplish this purpose is discussed in Appendix D to NUREG-1335, "Individual Plant Examination: Submittal Guidance." The staff is, however, considering whether audit calculations with MELCOR would be desirable for some IPEs in the future. If any important discrepancies remain in MELCOR at that time, cautions explicitly addressing those discrepancies would be provided to the IPE reviewers.

The ACRS recommended that the RES staff should consider the development of procedures to make it less likely that significant problems would exist at an advanced stage of a code's development. We agree with this comment and have taken steps to improve the process for making modeling improvements to severe accident codes. The NRC has put a contract in place to provide an independent assessment and evaluation of models that are proposed for implementation in severe accident codes. The evaluations will be carried out by a panel of three experts knowledgeable in phenomena related to the proposed models. The panel will provide an independent assessment of proposed modeling capabilities, limitations, and adequacy of the models. The panel's report will recommend whether the models under consideration should or should not be implemented in given severe accident codes, or will recommend alternative models for implementation in the same code. Performing an independent review prior to making modifications to a code will provide means to improve code development activities and avoid problems in the future.

The ACRS was also "concerned that the modeling of parts of the severe accident sequence, which the code [SCDAP/RELAP5] treats, are said to be based on bounding models rather than on best estimates. This could lead to generation of misinformation ... " In response to this comment, all of the late phase models, with one notable exception, in SCDAP/RELAP5 should be considered best estimate models in that they reflect to the best of our ability what is shown by severe accident experiments or the TMI-2 accident. In those limited instances where user input can have a significant influence on the treatment of phenomena, default values are included in the code that reflect the best judgement of the values to be used. These input options are necessary in some cases because understanding of the phenomena is not sufficient to force a single value for all ranges of conditions. As our knowledge base improves, ultimately these user input options should be eliminated except where necessary to perform uncertainty analysis.

As for the one notable exception where a bounding model is used, SCDAP/RELAP5 does not have a best estimate model for the interactions between the melt, structures, and water as melt relocates from the molten pool to the lower plenum. As discussed earlier under the FCI section, we intend to have a peer review of the new information generated from that research program. At that point, an FCI best estimate model will b recommended for incorporation in the severe accident codes.

It is our recommendation that the results of calculations performed for the late phase behavior should be treated with caution since there are very little data to perform quantitative code assessment results for these conditions.

Use of Risk Analysis in the Planning of Severe Accident Research

The ACRS stated that it is not convinced that enough attention is being given to the results of risk analysis in the planning of severe accident research. The ACRS also stated that although there are some who would argue that the risk is already sufficiently low that additional research is not warranted, the ACRS has not yet reached that conclusion. The ACRS would like to see more evidence that the choice of research areas and the approach to the research is made with risk reduction as a principal focus.

We agree with the ACRS that the areas of research and the research approach should be made with risk reduction as a principal focus. By way of background, the thrust of the SARP, up to the time NUREG-1365 was issued in 1989, was to establish and refine the technical and scientific base of knowledge in the areas of severe accident phenomenology and to reduce the uncertainties in this knowledge base and the risk assessment that depends on it. NUREG-1150 was initiated so that risk perspectives deriving from this improved knowledge base could be ascertained. In 1989, the staff identified an integrated program to bring severe accident regulatory issues to closure, and the direction of the SARP was shifted to focus on the issues that contribute significantly to risk, i.e., those accident scenarios that could lead to early containment failure. At that time, the first draft of NUREG-1150 identified Mark I liner melt through and direct containment heating issues as the dominant risk contributors and identified the source term uncertainties and core melt progression uncertainties as the leading uncertainties in the overall risk for those plants analyzed. Through the focusing of the research program, and the resultant improved knowledge, the final NUREG-1150 concluded that some of the issues that were found to be risk dominant at the time the draft report was issued were no longer dominant risk contributors, e.g., direct containment heating (DCH) for the five plants studied. Although we accept this conclusion and have factored it in our research planning and issue resolution, it was inappropriate to terminate research based on those results for three reasons: (1) in many cases, NUREG-1150 relied on expert judgement to estimate the uncertainty ranges associated with a variety of issues. It was necessary to examine the importance of each issue as well as the underlying basis for the expert judgement to determine if more information was needed to reach regulatory closure, (2) the single-valued importances usually presented in PRAs are generally based on the contribution to the calculated mean value of the risk. However, the entire distribution must be considered, since the predicted risk is very sensitive to variations in the tails of certain distributions, and (3) results can be very plant-specific and the issue might be a dominant risk contributor for other plants. Therefore, technical understanding of an issue may need to be improved if we are to resolve the issue for other plants. For example, the reduced risk from DCH for the Zion and Surry plants as reported in the final NUREG-1150 report was based on research results which indicated that there is high likelihood that the primary system will be depressurized at time of vessel breach. Depressurization analyses and the code used for these analyses were the results of the ongoing work on core melt progression and natural circulation during severe accidents. However, because of the different types of accident sequences that may be dominant from plant to plant, and design differences that affect system response to severe accidents, we cannot conclude that for all plants and for all accident conditions, the RPV will be depressurized with a high degree of certainty. Therefore, we also performed research to understand better the loads associated with DCH for those situations where high pressure melt ejection could not be precluded. Final resolution of the DCH issue will take into account the results of analyses of the potential for unintentional depressurization and the magnitude of the containment load should DCH occur if the RCS remains at the high pressure (and the uncertainties in these analyses). In addition to the large variation of accident scenarios, the situation is aggravated further by the fact that there are at least 14 different containment configurations that can have an impact on the magnitude of DCH loads. Therefore, general understanding of mechanisms that can enhance or mitigate DCH are needed, and the significance of the uncertainties associated with these analyses, relative to the overall uncertainties in the risk estimates, must be considered. That is the focus of the DCH research program. In summary, risk results are key to our planning and we are focusing our research on those areas that can directly influence our current understanding of risk and risk uncertainty.

Closing Comments

We agree with the Committee's recommendation for better communication among the various units working on parts of a larger problem, but we did not intend to leave any impression that there is a lack of communication. For example, while it is true that the direct containment heating (DCH) research is being managed by the Accident Evaluation Branch (AEB), Division of Systems Research, the overall direct containment heating issue resolution takes into account the probability of the initiating events (Probabilistic and Risk Analysis Branch, Division of Safety Issue Resolution), the probability of the RCS remaining at high pressure at time of vessel failure (Reactor and Plant Systems Branch, Division of Systems Research), the magnitude of the loads associated with high pressure melt ejection (AEB), and the structural capability of the containment (Structural & Seismic Engineering Branch, Division of Engineering). The staff is familiar with the work done outside their individual branches. Research results generated in all of these branches that are relevant to DCH issue are being used by the staff to resolve the DCH issue. Furthermore, we work closely with the Division of Systems Technology of NRR to keep the appropriate branches and people there well informed on the directions, results and implications of SARP programs. Our presentation to the ACRS did not include either presentations by these individual branches, or a presentation which explicitly described the integral nature of our program. We will emphasize these relationships in future presentations on the severe accident program.

ENCLOSURE III

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Nuclear Safety Research Review Committee
Washington, D.C. 20555

11 August 1992

Mr. Eric S. Beckjord
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Beckjord:

Enclosed please find a copy of a report of the review of the "Severe Accident Research Program Plan Update," Draft NUREG-1365, Revision 1, that was prepared by NSRRC's Severe Accident (SA) Subcommittee based upon its meeting on June 2-3, 1992. The SA Subcommittee's report was reviewed by the members of the NSRRC and was discussed by a quorum of the members in a telephone conference call on August 10, 1992. Members participating in the telephone conference call were David Morrison, Herbert Isbin, Thomas Boulette, Spencer Bush, Sol Burstein, Edwin Kintner, Fred Molz, and Richard Vogel. The Committee concurs with the findings and recommendations made by the SA Subcommittee and submits the Subcommittee's report verbatim to you as a report of the Committee.

If you have any questions on this NSRRC report, please contact Dr. Herbert Isbin or me.

Sincerely,

David L. Morrison

David L. Morrison
Chairman
Nuclear Safety Research Review Committee

DLM/sje

Attachment

REVIEW OF THE "SEVERE ACCIDENT RESEARCH PROGRAM PLAN UPDATE"
Draft NUREG-1365, Revision 1

The Severe Accident (SA) Subcommittee submits the following report for approval by the Nuclear Safety Research Review Committee (NSRRC).

The Severe Accident NSRRC Subcommittee met with NRC RES on June 2, 1992, to review NUREG-1365, Revision 1. Presentations were made covering the 11 major SA issues, and the discussions served to provide additional clarification and input, including assignment of priorities, the budgets for FY 1992 and the projected FY 1993, milestones, and user identification. Also included were new research results, and briefings on peer reviews. The current status of the NRC review of MAAP was given. The Committee has in prior meetings reviewed the process of identification of NRC needs and requirements for research, and in the course of the Subcommittee's discussions, recognition was made that as the various research programs progress that there may be other potential users of the research findings. The Subcommittee recommends that RES use the most effective means for including the additional information provided to the Subcommittee, along with other suggestions being made in this report, in updating the SARP report or in future Five-Year Plans which might well eliminate the need for the periodic revisions to NUREG-1365. The Subcommittee recognizes that RES's response to this report also represents an appropriate way to document additions, clarifications, and improvements in the Severe Accident Research Program.

The priorities assigned to the 11 major SA issues are noted as follows:

High Priority
- Closure of Mark I Liner Failure
- Closure of Direct Containment Heating (DCH)
- Advanced Light Water Reactors
- Severe Accident Codes

Medium Priority
- Fuel-Coolant Interactions and Debris Coolability
- Core Melt Progression and Hydrogen Generation
- Hydrogen Transport and Combustion

Issues Almost Complete and Continuing Studies Considered Confirmatory including International Work
- Scaling
- Source Term
- TMI-2 Vessel Investigation Project
- Core-Concrete Interaction (with refinements to the CORCON-MOD 3 code and continuation of validation)

The SA Subcommittee concurs with RES on the general ordering of priorities and on the programs underway. Results of these research programs are applicable to operating plants, updating the source term, generic rulemaking involving severe accidents, probability risk assessments, and resolution of generic safety issues.

The SA Subcommittee also notes its concurrence with the goals of SARP:

"...complete all the major severe accident experimental programs within the next 2 to 3 years"
and
"...closure of all severe accident issues ... in 4 years"

For termination of a research activity, RES augmented the SARP report on criteria for closure with discussion of how judgments are to be used for regulatory closure of an issue, and what specifically is needed to close an issue. The Subcommittee recommends that the SARP report reflect additionally the comments made by the RES Deputy Office Director on closure. Committee meetings in the past have considered this matter, and the Committee would like to be involved for such specific actions in the future. Further, the Committee takes note of the recent memos involving the June 3, 1992, Commissioner F. Remick to E. Beckjord, and the June 15, 1992, response "Closure of Research Projects and Maintenance of Capabilities".

RES's use of peer reviews to reach consensus and resolution of technical issues provides an open process for experts to interact with the ongoing research programs. The Subcommittee strongly endorses this activity. Additionally, the Subcommittee strongly endorses the various international and cooperative research programs underway. Not only do such international programs provide partners in sharing costs, but provide a broader technical base for ensuring more effective research and enhanced safety of nuclear plants worldwide. Severe accident codes are being used also on an international basis. For example, MELCOR involves users (both domestic and international) in a newly organized MELCOR Cooperative Assessment Program, MCAP. The Subcommittee recognizes the continuing need for research involving ongoing code improvements, and the need for the current and planned assessment tasks, using a disciplined approach. Users of codes as well as code developers have been aware of problems and limiting applications of the severe accident codes, and these problems have been restated and augmented through peer reviews. The Subcommittee was briefed on the progress of the RES response to peer review findings. The Subcommittee encourages RES in its continuing programs to resolve code deficiencies and to hold code developers to strict standards of scrutability. Further, the Subcommittee notes RES's programs for reducing the number of codes under development and for planning assessments for the remaining codes. These are areas that the full Committee will address in future meetings.

Further, the Subcommittee recognized that while computer codes play an important role in practically all aspects of modern science and engineering, the NRC research program dealing with severe accidents has developed knowledge and insights that go beyond what can be incorporated in a code. It is this knowledge and insight that is the primary product of research activities, and should guide the limitations and applications of code development as a means for summarizing and conveying information in a form that is manipulated easily. The experience of Subcommittee members is that the relative novelty of modern computers and graphics systems can sometimes induce individuals, including highly skilled peer reviewers, to over-emphasize the importance of computer codes at the expense of the broader knowledge base that is behind them.

The SA Subcommittee also agrees that research requires the coordination and management of "...a continuing focussing and refocussing..." to "...provide the basis for improved judgments as to where to expend future efforts." The Subcommittee recognizes that RES has a six-point integrated plan for closure of severe accident issues, and, in future meetings with RES, will discuss progress by reviewing such elements as the assessment of individual plant examination and inclusion of external events, issues involved with containment performance improvements, fuel-coolant interactions and debris coolability, and how research findings affect accident management.

The Subcommittee concurs with RES that major accomplishments in the severe accident program include the process for reviewing and directing the programs which involves peer reviews; the progress made with the Mark I Liner Failure Issue including the Risk-Oriented Accident Analysis Methodology (ROAAM); and the setting and achieving of milestones for the 11 major severe accident issues. An important new development for resolving the DCH issue was presented to the Subcommittee. The proposed activity is a six-month cooperative program between Sandia and the University of California, Santa Barbara, and will use the integral test data and the methodology of ROAAM. Preliminary indications using the results of the integral facilities at Argonne and at Sandia are very promising for resolving the DCH issue for Zion-like containments. Test results to be obtained this year for the Surry-like containments will be used to confirm resolution of the DCH issue for these containments, too. Successful completion of this program will guide what additional considerations need to be given for containments not like Zion nor Surry.

The Subcommittee agrees that progress has been made in improving the data base and analytical studies for fission product release and transport, in code developments for VICTORIA, for in-vessel source terms, and for CONTAIN, for ex-vessel source terms; in core-concrete research and code developments; reaching closure on hydrogen transport and combustion with only a residual issue involving high-temperature mixtures to be studied and resolved; and with the TMI-2 vessel investigation project.

With respect to the Severe Accident Scaling Methodology (SASM), the Subcommittee recognizes the key to its application for a specific case, such as DCH, lies in the exploratory research that is required to identify key phenomena. This was accomplished in the integral tests that have been undertaken at Sandia and at Argonne. The Subcommittee concurs with RES that no further work needs to be done with SASM.

Programs being initiated for severe accidents involving advanced light water reactors will be followed through the cooperative efforts of the NSRRC Advanced Reactors Subcommittee and the Severe Accident Subcommittee.

Overall, the Subcommittee was impressed with RES management of the Severe Accident Research Program as reflected in the revisions to NUREG-1365, the use and response to peer reviews, the broadening of the technical support through user programs involving severe accident codes, and through the extensive international cooperative programs.