D900424

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SEVERE ACCIDENT RESEARCH PROGRAM

During the 360th meeting of the Advisory Committee on Reactor Safeguards, April 5-7 and April 18-19, 1990, we reviewed the Severe Accident Research Program (SARP) of the NRC. Our Severe Accidents Subcommittee discussed this program with the staff during a meeting held on March 20-21, 1990. We also had the benefit of the documents referenced.

During this review, emphasis was given to what the staff describes as its short-term program. Basically, the short-term program focuses on issues associated with early containment failure, e.g., BWR Mark I liner attack and direct containment heating (DCH). However, a description of the long-term program was also presented and discussed briefly. In what follows, we give a brief description of most of the elements of the SARP program, together with our comments and recommendations.

Adding Water to a Degraded Core

This investigation proposes to address a number of questions arising in connection with the in-vessel progression of severe core damage. The planned studies are said to address issues of in-vessel vapor explosion, thermal shock, and recriticality. Each study is an analytical investigation using a number of existing codes.

We are not convinced that the codes to be used are capable of providing the information being sought with sufficient validity that it can be used for the purposes listed, namely, the removal of uncertainty and the provision of information for use in the individual plant examination (IPE) program. Nor are we convinced that models that are to be developed can be demonstrated to be valid in a reasonable time, if ever. This program should receive the sort of analysis that is being developed in the Severe Accident Scaling Methodology (SASM) program discussed below.

Core Melt Progression

This is a program planned as a collaborative effort with the Federal Republic of Germany (FRG). Its purpose is said to be learning more about core melt progression in BWRs in order, presumably, to construct codes to describe the in-vessel melt progression process in this type of core. We were unable to determine how much additional information is needed in the regulatory process or whether the planned program will produce it in time for it to be used in, for example, the IPE program, which

is apparently its earliest planned application.

This program should be examined further, using the techniques developed in the SASM program. We commend especially the lessons learned from that program and the "Evaluation Questions for Proposed Severe Accident Experimental Programs" as were discussed with our Subcommittee on March 20-21, 1990.

Examination of TMI-2 Lower Head and Lower Head Failure Analysis Plan

The sampling of TMI-2 lower head material described to us seems to be worthwhile, since we hope and believe that incidents that produce such an opportunity will be infrequent, and one should learn as much as feasible from this accident, especially if it will help prevent future accidents. However, we were disappointed at the response when we asked how the information being collected would be used by the NRC. We were told that it would be used to calculate "the margin to failure." However, when we asked what one would do with this margin there seemed to be considerable uncertainty.

The lower head failure analysis plan is a rather extensive and ambitious effort to model various postulated modes for bottom head failure. Effort will be made to use the information collected from the TMI-2 lower head examination to validate some of the models. In particular, we assume that effort will be made to understand why the relocation of a portion of the hot core into the lower head caused such a limited temperature rise in the body of the vessel head.

Presumably, some estimates of likely lower head failure timing can be made from the results, and since the timing of lower head failure has an influence on the time at which containment failure is likely to occur, this information may be useful in risk estimates. We did not, however, receive any information that would lead us to believe that significant thought has been given to how the information developed will be used in or useful to the regulatory process. We observe also that the number of variables likely to enter into a determination of lower head failure is so large and largely unpredictable that predictions of the likelihood of the various possibilities may be subject to very large uncertainties.

Severe Accident Scaling Methodology

Much experimental research is performed under conditions of size, geometrical configuration, temperature, pressure, and in some cases with materials that are different than what is expected in severe accident conditions in a large power reactor. It is usually hypothesized that if a computer code which has been constructed to model the progression of a severe accident can predict the results of properly constructed experiments, the code can, with appropriate allowances for the differences in the experimental conditions and what is expected in the reactor accident, predict at least the important features of a severe accident. These allowances, or scaling factors, require a careful analysis of both the experiments

and the hypothesized accident. The process has not received the attention it should have had in much of the severe accident research performed for the NRC.

Members of the staff described a program which appears to be well designed and which, if it continues and if the results are applied, is likely to place future experimental work and code development on a much sounder basis. It will also provide guidance for any further experimental work that may be required, guidance that has frequently been unavailable in the past. We were impressed that the project manager was able to assemble an outstanding group of experts with representation from industry, the national laboratories, and academia, and was able to obtain significant cooperative effort from the group in performing the study. To ensure its applicability, the method of analysis that has been developed is currently being applied to the investigation of a DCH sequence. It gives promise of providing needed insight into this thorny problem, as well as providing guidance in the planning of other research programs.

The Probability of Liner Failure in a Mark I Containment
NUREG/CR-5423

The authors of this report have collected, from a variety of sources, and have correlated a considerable body of information about the important phenomena that contribute to the processes that begin with severe damage of the reactor core and result in a pool of molten corium in contact with the metal structure that forms the boundary of the dry well of a Mark I containment. This, and their efforts to construct from the information a coherent picture of the core melt-vessel breach-attack of the liner sequence represent a significant contribution.

However, it is important that it be recognized for what it is, and for what the authors say it is, namely an attempt to take the existing information, to fill in the gaps of information needed to reach a conclusion about liner failure, using mostly engineering judgment, and to thereby construct a framework. This framework, given the existing information and the assumptions made to fill the gaps, permits the authors to reach conclusions about liner failure. And although the authors claim confidence in the conclusions they reach, with a few caveats, it would be unfortunate to use the results in making regulatory decisions without recognizing that conclusions about many of the important phenomena, for example the rate of release and the state of the corium from vessel breach, that have a significant impact on the final result, are supported primarily by the authors' judgment.

An important part of the report, and a part which was not available to us for our review (it is not yet complete), is Appendix F, that will contain the results of a peer review of the report. We have not had the rest of the report long enough to perform a thorough review. However, we do make the following observations:

The approach used appears to be sufficiently similar to that developed in NUREG-1150 that the authors might have estimated the uncertainty in the results of their calculations. This

would have added to the value of the report. During the presentation made to us, one of the authors argued that because computational uncertainty is not the only uncertainty in the result, it was not considered useful to estimate it. However the calculational process is the question at issue here, and unless some bounds can be set on the uncertainty of the results of the process, its value is diminished considerably.

Even though the authors chose not to make a quantitative estimate of uncertainty, they are, having gone through this extensive study, in a unique position to identify, at least qualitatively, where the greatest contributors to uncertainties lie. They should be encouraged to do this, as well as asking others to identify them as they chose to do.

Furthermore, the authors do not discuss whether the method used for those situations for which needed information is not available, i.e., estimating probability distribution functions, sampling these functions to get a range of possible values for the parameter of concern, and finally combining the results in a way which is something like calculating a mean, is any more nearly valid than estimating the mean value at each place where needed information is unavailable. The authors should be encouraged to justify that the method used is superior to simply estimating a mean value for uncertain parameters and using that value for further calculation.

Continuing Code Development

In our report of March 15, 1989 to then Chairman Zech, we noted that a review of NRC sponsored codes was being performed and that support for some codes that were found to be duplicative or no longer needed would be discontinued. This review has been completed and further support for several codes has been withdrawn. There has also been an increasing emphasis on documentation of those codes that exist as well as those being developed. We applaud this emphasis.

We were briefed on continuing development of two codes that are to be retained, CONTAIN and MELCOR, which are expected to provide much of the analytical capability which the staff will use in severe accident analysis. Unfortunately, the presentations and discussions were such that we were unable to obtain the information required for making any recommendation at this time. We will explore this further because the staff is expecting to make use of these codes in drawing conclusions about severe accident progression in both existing and new plants. MELCOR is, for example, to become the principal tool for calculating fission product sources.

Molten Core-Concrete Interactions

This experimental work is said to be needed because of continuing uncertainty about the contribution of Molten Core-Concrete Interaction (MCCI) to containment failure. The point was made that the contribution is primarily to late containment failure, and thus

may be less important than contributors to early failure. However, because the staff expects that advanced reactor designers will assume that the debris produced during core melt will be coolable in the designs being proposed, the additional information being sought is deemed essential to advanced reactor review. It is also claimed that MCCI is an important part of the Mark I liner failure issue.

Integration of SARP with Foreign Research

In addition to their own research, the Division of Systems Research has a systematic program in place to learn from and, in some cases, to participate in the research of several foreign countries. This program seems effective.

Long-Term Research

To a considerable extent, the research proposed could be said to be more of the same. Most of what is described is justified on the basis that uncertainties need to be decreased in such areas as Modeling Severe Accidents, In-Vessel Core Melt Progression and Hydrogen Generation, Hydrogen Transport and Combustion, Fuel-Coolant Interactions, Molten Core-Concrete Interactions, Fission Product Behavior and Transport, and Fundamental Data Needs. It will be recognized that these are not new, and indeed each has been an object of research almost from the beginning of the Severe Accident Research Program.

We do not have sufficient information to justify an endorsement of this program, although this may be because it is not yet well defined. We were unable to obtain satisfactory answers to questions such as:

How much uncertainty is acceptable?

How much will the proposed research reduce the uncertainty?

Will the information obtained reduce risk, or will it merely permit less conservative approaches to design and operation of plants?

This program is another that should be subjected to the type of analysis suggested in the SASM program.

In connection with both the long-term and the short-term programs, we perceive a lack of communication between those planning the research and those who will use the results. It is indicative of the loose coupling between severe accident research and regulatory activities that in his summary of the research program, provided to the subcommittee, the Director of the Division of Systems Research commented that the "Agency doesn't have a definite regulatory use for severe accident data, i.e., no rule or regulation, no user needs letter." He did go on to indicate that there are a number of "indirect" uses. However, it appears to us that the main point he made is valid, and is a point of some concern. As early as 1975, WASH-1400 illuminated the risks associated with severe accidents. This led to the conclusion that absent a severe

accident there is little or no risk to the public from the operation of nuclear power plants. Yet since that time, even in the light of the TMI-2 and Chernobyl accidents, little change in the regulations that govern the operation of nuclear power plants has occurred. Even for plants not yet licensed, there are virtually no new regulatory requirements dealing with the performance of the plant systems in the course of a severe accident. We have, for example, virtually the same rules governing containment performance requirements as we had in 1971. Of course it is required that new plant designs be accompanied by a PRA, but how the PRA is to be used in judging the acceptability of the design is undetermined. Under these circumstances, it is difficult to judge what new research in severe accidents is needed. Of course, it is possible that nothing more need be done, but aside from the Commission's Safety Goal Policy Statement and its Severe Accident Policy Statement, there has been no formal recognition of severe accidents, even for new plants.

It may be that the current emphasis on what happens in the plant after breach of the vessel is overdone. Examination of the results of most of the existing PRAs indicates that none show risks in excess of the Safety Goal quantitative objectives (not all of these, however, include seismic risk). However, several show core damage frequencies in excess of the sometimes proposed goal of 1E-4 per reactor-year. Thus, in a situation in which resources are limited, it may be that more emphasis should be placed on decreasing the likelihood of core damage. For example, for PWRs, many PRAs estimate that off-site risks are dominated by the ISLOCA (this is the case for Millstone-3, Seabrook, Surry, and Sequoyah, for example). Here phenomena occurring after vessel melt-through are of little consequence in risk determinations.

In the presentation to the subcommittee, the staff representatives stated that they believe uncertainties in the vessel failure scenario, and subsequent events, are the major contributors to risk uncertainties, based on PRA results. This is at least questionable in view of the risk attributed to seismic events and to human performance, and the large uncertainties associated with both of these.

Comments and Recommendations

There is much of deja vu in the proposed severe accident research. The same areas that were being explored at the beginning of the program almost ten years ago are still being investigated. The justification given by the Office of Nuclear Regulatory Research is that uncertainties exist which are large enough that regulation is difficult or impossible. However, there is little assurance that the proposed research will reduce the uncertainties to an acceptable value. Nor does there seem to be a very specific idea of what an acceptable value would be. This is probably not altogether the fault of the Office of Research.

A decision on what is acceptable is difficult to make, and requires, as a minimum, a close collaboration of the Office of Research with the Office of Nuclear Reactor Regulation. There appears to be an improvement in this collaboration, but from what we can tell, more teamwork on the issue of what research is needed

is essential if the research is to be properly focused.

We are enthusiastic about the SASM program. Moreover, the approach that is being developed, if applied to planning the NRC's severe accident research program, can result in focusing the program to areas where it is most needed, and in making it more likely that the projects undertaken will produce useful information.

The MCCI work is a further pursuit of information on ex-vessel severe accident phenomena. Although we were not provided with enough information to reach firm conclusions concerning the worth of the proposed research, we observe that estimating the contribution of MCCI to late containment failure requires information beyond establishing the cavity area that will ensure quenching of core debris.

Concerning the programs discussed above we have the following recommendations:

We recommend that the proposed research projects on Adding Water to a Degraded Core and Core Melt Progression not be undertaken until they are subjected to a review of the type developed in the SASM program. If they survive the review they will be much more likely to enhance the regulatory process.

We were told that, in light of the staff's view of the success of the study described in NUREG/CR-5423, consideration is being given to applying this same type of analysis to the DCH issue. In our view, a SASM-type approach is likely to produce more useful information than will the NUREG/CR-5423-type analysis in its present state of development. We recommend that a SASM-type study be used as an alternative to the NUREG/CR-5423 approach.

In connection with the TMI-2 vessel examination, we recommend that further thought be given to the way in which the information being collected might be used. We consider the examination worthwhile, but believe there must be applications beyond calculating the margin to failure.

We recommend that the Lower Head Failure Analysis be subjected to the SASM process. If this study is to be done, it should have more of a relationship to regulatory needs than we are able to discern.

We recognize that this report may seem unduly critical. However, our comments reflect our perception that the various elements of the SARP lack focus. We do not attribute all of this lack of focus to the Office of Research. Part of it comes from the inability of the agency to deal with severe accidents in a regulatory context.

Sincerely,

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1365, "Revised Severe Accident Research Program Plan FY 1990-1992," August 1989

2. U.S. Nuclear Regulatory Commission, NUREG/CR-5423, "The Probability of Liner Failure in a Mark-I Containment," T. Theofanous, et al. (UCSB), February 1990

3. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Volumes 1 and 2, June 1989