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Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON NUREG-0956, "REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS -- DRAFT REPORT FOR COMMENT"

During its 306th meeting, October 10-12, 1985, the Advisory Committee on Reactor Safeguards discussed NUREG-0956 with representatives of the NRC Staff, and we completed our deliberations during the 308th meeting, December 5-7, l985. This report had previously been reviewed by a Subcommittee in meetings on May 2, August 1 and 2, and September 27, 1985. We also had the benefit of the documents referenced in l-5 and discussed the report in Reference 6.

We conclude that:

(1) Although the report is a useful description of progress that has been made in the NRC's Severe Accident Research Program, it provides only a part of the information likely to be needed in deciding whether and how to restructure existing regulations to deal with accidents beyond the current design basis accidents.

(2) Since much of the motivation for the severe accident research program came from observations made after the TMI-2 accident, some of which led several investigators to conclude that source terms previously used to describe severe accident consequences were much too large, we believe the report should either state that information developed to date indicates a significant difference compared to the predictions of WASH-1400, or that no significant difference is now believed to exist. The report is ambiguous on this point.

(3) The report is cast in a framework which depends on the use of a suite of codes to describe the course of severe accidents. Reference is made to the considerable uncertainty that exists in the results that the codes predict. No guidance is given as to how to take this uncertainty into account in making decisions related to licensing or regulation. Since dealing with this uncertainty is one of the more difficult parts of the decision process, more attention needs to be given to approaches for dealing with it.

(4) The suite of codes that forms much of the basis for the report deals with containment in a rather preliminary fashion. It appears to us that a much less ambiguous method for taking account of containment performance is needed, especially in light of the wide variety of containment types that exist in operating plants.

(5) Many of the phenomena and the processes described in this report have also been studied in some detail by those responsible for the Industry Degraded Core Rulemaking (IDCOR) Program. It would be valuable, in considering the results of and the conclusions drawn from NRC's research programs, to have some discussion of the differences and the similarities of the conclusions reached by the IDCOR group compared to those of this report.

Additional comments on these points and other features of the report are given in what follows.

It was recognized, following the TMI-2 accident, that more attention must be given to the risk posed by accidents beyond what were then being considered as design basis accidents. It was also known that new information and new understanding had been developed since the publication of WASH-1400, Reactor Safety Study. Accordingly, the NRC Staff undertook to collect, evaluate, and publish in NUREG-0772, Technical Bases for Estimating Fission Product Behavior, the best information then available concerning fission product release and transport during and following a severe core damaging accident.

On the basis of that collection, and of an evaluation of the information that would be needed by the NRC Staff as it prepared to deal with the severe accident issue, the Office of Nuclear Regulatory Research (RES) formulated a research program aimed at improving the accuracy with which the source term could be predicted. NUREG-0956 reports the results of that research.

The report was described by staff members of the RES Office as containing the scientific bases from which source term calculations could be made. It places major emphasis on the assembly of a set of computer codes which have been used for computing source terms for five reference plants. Several steps were taken to improve the validity of the codes: a validation study of the constituent computer codes done at the Oak Ridge National Laboratory, a quantitative uncertainty study performed by the Sandia National Laboratories, and an independent review of the results of the NRC's source term research by a study group of the American Physical Society.

Much of the research that forms the basis for this report was stimulated by the investigations associated with the TMI-2 accident. Several investigators concluded, primarily as a result of the radioactive iodine estimated to have been released to the containment atmosphere during the accident, that the source terms calculated and used in WASH-1400 were much larger than should be expected if one considered the release and transport of fission products in the light of a more careful investigation of the chemistry and the physics of the various processes involved.

Many of those who concluded that the source terms used in the WASH-1400 calculations were too large also predicted that when more appropriately chosen source terms were used, the calculated risk from severe accidents could be shown to be several orders of magnitude smaller than those previously calculated. One might therefore have expected this report to contain some conclusions concerning the risks to be expected when this newly developed set of codes, incorporating the new data resulting from an extensive research program, are applied to the analysis of severe reactor accidents. Comments in the report on this question are at best tentative.

For example, the report notes in the section on Risk Insights that a "comparative risk appraisal" (using WASH-1400 accident frequencies, but source terms calculated from the new set of codes) indicates a reduction in risk. The report concludes that the reduction (early fatalities are about a factor of ten lower -- delayed, about a factor of four) is about equally divided between that resulting from a change in the treatment of fission product release and transport, and that resulting from a different approach to describing containment behavior. In other cases the comments are ambiguous. For example, "New source terms have been calculated for selected accident sequences for five reference plants that represent major reactor containment types in operation in the United States. These selected sequences have provided a sufficient test of the capabilities of the computer codes." What was the "sufficient test"? How was the adequacy of the codes developed? One attempting to judge the merits of the code set, or to ascertain whether the risks predicted in light of the new information that has been developed are indeed smaller, would find more information helpful.

On the basis of our examination of the report, and of our extensive discussions with the Staff, we conclude that the report can best be characterized as a status report for a task well begun but far from conclusion.

In our efforts to evaluate the adequacy of the report we repeatedly raised the question of how and for what purpose the material in the report will be used. Several possible applications were mentioned, but we were told that details of usage will be developed by those who are to use it -- that this report contains primarily the science that has been developed, and not its application. This response is understandable, given the compartmentalization of the Staff that exists, but as a result, as uses are developed, questions will arise that are likely to require further investigation or additional explication of the material that has been gathered. We commend for consideration of the Staff the proposition that applied research is not completed until it is used.

We conclude that this report, and further investigations likely to be required in order to make its application to severe accident analysis feasible, can be understood only as part of a package made up of several identifiable components. This report is one of those. It includes, or refers to, the new information that has been developed concerning fission product release from fuel (both in and outside of the vessel), and its transport into containment. It also incorporates the suite of codes developed (as described in BMI-2104) for modelling the course of severe accident sequences following the onset of core damage.

The new risk calculations to be carried out for six selected plants and to be reported in NUREG-1150, Risk Perspectives and Rebaselining, form another component. The accident sequence initiator frequencies to be used in this set of calculations will presumably come from the Accident Sequence Evaluation Program. Presumably the modelling of containment performance to be used in the calculations will come from the Severe Accident Risk Reduction Program, although this is not clear.

The incorporation, yet to be accomplished, into one coherent method, of the various approaches being developed to describe containment performance is another, and an extremely important component. The formulation of methods for carrying out a detailed severe accident analysis for each operating plant, cited in the Severe Accident Policy Statement, is another.

Judged in this context we believe the report is a useful addition to the earlier information on fission product release and transport, and to the methods that have been used in the past to model the behavior and the consequences of severe accidents. However, we conclude that the codes, in their present form, should not be given much weight in making decisions.

For example, the report observes that considerable uncertainty exists in the results to be expected when the constituent codes are employed. Reference is made to further work to be done in defining uncertainties. However, no guidance is given to the prospective user on how to account for or how to deal with uncertainties. Nor is there any comment on whether the uncertainty to be expected from employment of the suggested new approach is greater than or less than that which might be expected if, say, the WASH-1400 approach is used. More information on the effects of the identified uncertainties is needed. Guidance on how to deal with existing uncertainties should be provided if the results of the report are to be used for making decisions. Furthermore, the description given to us by the Staff, of work which is planned to provide more nearly quantitative estimates of uncertainty, leads us to believe that what is proposed would be better described as a sensitivity analysis.

One of the "Source Term Insights" given in the report is that, "For most accident sequences, the largest single factor affecting source terms is containment behavior. A delay of several hours in containment failure will reduce source terms significantly." We agree with both statements. However, the guidance on containment behavior modelling is confusing. Appendix A gives some general discussion of containment types, and of their behavior in accident situations. Appendix B claims to be a summary of a Sandia National Laboratories' report which treats "Containment Event Analysis." It is intended to "provide a containment matrix for the risk perspective for the Surry plant and to discuss the containment behavior of the other plants analyzed" in BMI-2104. However, the discussion and the conclusions are laced with caveats, and the reader is warned that the evaluations are preliminary. The material in Appendix B also seems to be at variance with other NRC work related to containment behavior. For example, in Appendix B, in several places, there is reference to in-vessel steam explosions in a context which indicates that they are thought by the Staff to be a possible significant contributor to the likelihood of early containment failure. However, the report of a review by the Steam Explosion Review Group convened by the NRC Staff (NUREG-1116) indicates a consensus that the likelihood of early containment rupture caused by in-vessel steam explosion is so low as to be negligible. There is also a comment in NUREG-0956 indicating that steam generator tube rupture may be an important containment bypass mechanism. No guidance is given as to how to deal with it. We conclude that in light of the importance attributed to containment system performance, and in view of the preliminary status of current models, much more work is needed in this area. We emphasize, as we have in other comments on methods for severe accident analysis and decision making, that development of more elaborate computer codes is not the only way or even necessarily the best way to proceed. Some well defined method for describing containment behavior is needed.

Bearing in mind that early comments concerning the contribution of iodine gave impetus to much of the research on fission product chemistry that has occurred, and observing that the report points to better fission product chemistry as one of the major improvements that has been produced, we asked what changes in risk could be identified as a result of the changes in the way iodine is treated. We were told that the Staff had not attempted to identify these changes. We suggest that, especially in light of the designation of the report as a scientific document, efforts to identify the changes in risk due to differences in the treatment of a few key contributors would add considerably to the understanding and to the utility of the results. We also believe it would be valuable to identify and to discuss areas of agreement and of disagreement (with more discussion of the latter) between the Source Term Package reported upon here and other relevant work, the IDCOR approach, for example.

There are several key areas in the modelling of severe accident progression as described in the report, about which we have some reservations. The transport and the retention of radionuclides in the primary system are tightly coupled to the temperature distribution in the primary system. This in turn is likely to be a strong function of the buoyancy driven recirculation in the primary system. This phenomenon is not treated in the models described in NUREG-0956. Work by other groups suggests that it could have an important bearing on temperatures in the primary system. For example, some investigators have suggested that it might lead to transport and condensation of fission products in the steam generator tubes sufficient to produce tube rupture in some postulated high pressure accident sequences. It is also predicted by some that this mechanism may lead to a sufficiently high temperature of the upper level components of the primary system in PWRs, that rupture will occur, in high pressure sequences, before, for example, the postulated expulsion of molten core material from the bottom of the reactor vessel, leading to severe containment atmosphere heating, occurs. This possibly important mode of heat transfer deserves further investigation.

Fission product release from the fuel is highly temperature dependent. Core melt progression and core melt temperatures are based on the MARCH code. Even in its present form, the code provides only a crude representation of the physical processes it is meant to predict. As a result, the molten core temperature is subject to considerable uncertainty. This uncertainty is reflected in calculations of fission product release. Better understanding of the resultant uncertainties is needed.

Ex-vessel release of fission products from the melt is strongly dependent upon the melt temperature, and this in turn is highly dependent on the core-concrete interaction. Some investigators interpret the results of the Beta tests at the Karlsruhe Nuclear Research Center (Federal Republic of Germany) to indicate that the heat transfer to concrete is higher than that predicted by the code used to model the core-concrete interaction in this package. Because much of the fission product release following late containment failure is currently calculated to come from the nonvolatiles released during core-concrete interactions, this possible discrepancy deserves further investigation.

The report is based upon work described in a large number of documents, some not readily available. Because of the importance of a thorough understanding of the bases of the results reported and conclusions drawn, it is vital that care be taken to identify the documents to which a user can go to obtain further information. We emphasize the importance of complete documentation of the foundation reports from which NUREG-0956 is drawn.

We have commented in a letter to the Executive Director for Operations, dated August 13, 1985, that we believe the representative risk calculations to be carried out and to be reported in NUREG-1150, as well as the methods developed for analysis of individual plants, should take account of external initiators.

We express our appreciation to the Staff for providing us with thorough, well organized presentations on this report, and for their efforts in responding to a number of questions which we posed during the course of our discussions.

Sincerely,

David A. Ward
Chairman

References:

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3. U. S. Nuclear Regulatory Commission, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," USNRC Report NUREG-0772, dated June 1981
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6. U. S. Nuclear Regulatory Commission, "Risk Perspectives and Rebaselining for Six Reference Plants," USNRC Report NUREG-1150, to be published

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