UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 15, 1991

The Honorable Thomas S. Foley
Speaker of the United States
House of Representatives
Washington, D.C. 20515

Dear Mr. Speaker:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission. In our December 18, 1986 letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive annual report.

During the past year, we have reviewed the NRC Safety Research Program and other closely related matters in the following areas:

- Nuclear Power Plant Containment Performance Improvement Program
- NRC Safety Research Program Budget
- Severe Accident Research Program
- Evolutionary Light Water Reactor Design Certification Issues
- Human Factors and Other Organizational Issues
- Reactor Pressure Vessel Embrittlement
- NRC Computer Codes and Their Documentation
- Severe Accident Risk Assessment - NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"

We have provided reports to the Nuclear Regulatory Commission and the NRC staff on the above-mentioned matters. Copies of these reports are enclosed.

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PDR ACRS
GENERAL PDR

We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

David A. Ward

David A. Ward
Chairman

Enclosures:
1. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Containment Performance Improvement Program - Proposed Recommendations for MARK II, MARK III, Ice Condenser, and Dry Containments, March 13, 1990.
2. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Safety Research Program Budget, April 11, 1990.
3. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Severe Accident Research Program, April 24, 1990.
4. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990.
5. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Research on Organizational Factors, August 16, 1990.
6. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Yankee Rowe Reactor Pressure Vessel Integrity, September 12, 1990.
7. Report from Carlyle Michelson, ACRS Chairman, to James M. Taylor, Executive Director for Operations, U.S. NRC, Subject: NRC Computer Codes and Their Documentation, October 11, 1990.
8. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Review of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," November 15, 1990.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 13, 1990

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

Dear Chairman Carr:

SUBJECT: CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM - PROPOSED RECOMMENDATIONS FOR MARK II, MARK III, ICE CONDENSER, AND DRY CONTAINMENTS

During the 359th meeting of the Advisory Committee on Reactor Safeguards, March 8-10, 1990, we discussed the staff's proposed recommendations from the Containment Performance Improvement (CPI) program for plants with Mark II, Mark III, ice condenser, and dry containments. The staff intends to inform licensees with such plants of these recommendations in a supplement to Generic Letter 88-20 (Reference 1) and, by this action, will consider the CPI program completed. Our Containment Systems Subcommittee discussed this matter with the staff during a meeting on February 6, 1990. We also had the benefit of the documents referenced.

The CPI program is one element described in SECY-88-147, "Integrated Plan for Closure of Severe Accident Issues." Other elements in this plan are the Individual Plant Examination (IPE) program (Generic Letter 88-20), severe accident research, external event resolution, accident management, and improved plant operation. The CPI program was to identify any severe accident vulnerabilities that appeared to be generic to plants with a given type of containment. It was then to develop new regulatory requirements or guidance for reducing those vulnerabilities. Recommendations were to be derived by the staff and its contractors through study of risk analyses reported in NUREG-1150, other PRAs, and results from severe accident research. The intent was to identify any new requirements in the near term so that licensees could implement them along with any plant improvements identified in their own IPE efforts.

Mark I containments for BWRs were considered first. Staff guidance for Mark I plants was provided in Supplement No. 1 to Generic Letter 88-20 and in Generic Letter 89-16. We provided comments on the Mark I CPI program in our report dated January 19, 1989 to then NRC Chairman Zech.

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ENCLOSURE 1

The remaining four containment types have been considered as a group. The staff reports that it has "found no improvements for these containment types that would warrant generic implementation for all containments of a given type." However, it has identified some ways, unique to each containment design, in which plants may be particularly vulnerable to severe accident threats. While the staff has decided not to prescribe remedies for the generic problems it has identified, it does intend to provide licensees with technical insights and information that the staff believes to be of particular import. This will permit these lessons to be factored into IPEs and accident management programs that are being initiated by licensees. Summaries of the staff's concerns for each containment type are given in the proposed supplement to Generic Letter 88-20. More technical details will be provided in a series of reports that are being prepared by contractors to the staff and are expected to be available during June 1990.

The approach proposed by the staff is appropriate and we endorse the proposed supplement. We agree that the CPI program can now be terminated. As stated in our report of January 19, 1989 on the Mark I CPI program, the IPE program can be an effective and efficient means to identify and ameliorate risk-significant issues related to containment performance. The IPE and accident management programs will benefit by considering conclusions from these staff studies.

However, we recommend that the staff caution the licensees not to focus exclusively on the set of issues raised by the CPI program. For one thing, the designs analyzed in NUREG-1150 do not adequately represent the full spectrum of plants. For another, conclusions about risk and phenomena are subject to large uncertainties. Licensees should retain a broad perspective in their studies. The original intent of the IPE program, that is, to search "for possible ... 'outliers' that might be missed absent a systematic search," is applicable to issues of both prevention and mitigation.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:
1. Memorandum dated February 22, 1990 from Warren Minners, Director, Division of Safety Issue Resolution, RES, to Raymond F. Fraley, ACRS, Subject: ACRS Review of Supplement 2 [sic] to Generic Letter 88-20, Individual Plant Examinations, with enclosures:
   (a) Proposed Draft Supplement to Generic Letter 88-20, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities" (Predecisional)
   (b) Draft memorandum for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Recommendations of Containment Performance Improvement Program for Plants with Mark II, Mark III, Ice Condenser, and Dry Containments (Predecisional)
2. Letter dated November 23, 1988 from D. Crutchfield, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f) (Generic Letter 88-20)
3. Letter dated September 1, 1989 from James G. Partlow, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Installation of a Hardened Wetwell Vent (Generic Letter 89-16)
4. Letter dated August 29, 1989 from James G. Partlow, USNRC Office of Nuclear Reactor Regulation, to Licensees, Subject: Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f) - Generic Letter 88-2 Supplement No. 1
5. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants (Second Draft for Peer Review)," Volumes 1 and 2, June 1989
6. SECY-88-147, Memorandum dated May 28, 1988 for the Commissioners from Victor Stello, Executive Director for Operations, Subject: Integration Plan for Closure of Severe Accident Issues

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 11, 1990

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

Dear Chairman Carr:

SUBJECT: NRC SAFETY RESEARCH PROGRAM BUDGET

During the 360th meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 1990, we discussed the proposed NRC Safety Research Program and budget for FY 1991. Our Subcommittee on the Safety Research Program met with the Executive Director for Operations, representatives from the Office of Nuclear Regulatory Research (RES), and the Office of Nuclear Reactor Regulation (NRR) on February 7, 1990, and discussed the proposed FY 1991 budget along with the rationale for the continually dwindling NRC Safety Research Program budget and the associated impacts. After considering the information gathered at these meetings, we find ourselves concerned, not so much about the proposed FY 1991 budget, but about the trend of continually diminishing funding for the NRC research program. Unless this trend is arrested, the overall effectiveness of the agency will be seriously compromised.

We have been critical of certain parts of the NRC research program in the past and remain so (Refs. 1-6). It is not our intent to address program deficiencies in this report, but to communicate our belief that a viable research program is an essential part of the NRC regulatory process. In the following paragraphs, we describe the reasons for our concerns about the research budget trend, and offer suggestions for change.

TREND IN THE RESEARCH PROGRAM BUDGET

Pertinent figures from the NRC budgets for fiscal years 1975, 1981, 1983, and 1991 follow:

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ENCLOSURE 2

Fiscal Year	Total Agency Funding (in constant 1975 dollars)	Total Agency FTEs	Research Program Support Funding* (in constant 1975 dollars)	No. of FTEs* for Research
1975	$148.1M	2006	$ 61.2M	94
1981	294.6M	3139	129.5M	155
1983	277.4M	3403	110.0M	140
1991	218.0M	3240	36.1M	120

When the total NRC budget increased markedly in the late 1970s and early 1980s, the research budget increased proportionally. However, since 1981 funding for research has been much more dramatically diminished than that for the agency. From 1983 to 1990, the research program support budget, in 1975 dollars, was reduced by a factor of three.

POSSIBLE EXPLANATIONS FOR THE RESEARCH BUDGET TREND

Among the reasons that might be offered for the trend in research funding are:

- The Commission has explicitly decided that research has become less important than other agency activities. It may have concluded that nuclear power has reached relative maturity and that most of the technical questions relating to reactor safety and regulation have been answered. In competition with other demands on resources (e.g., the belief that more inspections of operating plants are needed), research has taken a "back seat."

- Research funding has been reduced as part of a policy directed by the Administration or the Congress, perhaps for the reasons mentioned above.

- Given the government budgeting process, it is easier to reduce funding for NRC research, which is largely allocated to persons and institutions not on the NRC payroll, than to curtail or terminate regulatory activities that directly involve NRC employees.

*Associated with actual research support which includes planning, coordination, and managing research projects. Does not include technical assistance support for developing rules and regulations, resolving generic and unresolved safety issues, or review of IPE/PRA submittals.

All of these reasons may have influenced the research funding trend, but we believe that the third reason has had a disproportionate influence. As evidence for this, staff presentations to us described the largest portion of the agency's budget, which includes funding for salaries, rent, travel, office accessories, etc., as "nondiscretionary." When pressed, the staff agreed that these funds were not really "nondiscretionary" in the sense that there is explicit guidance to that effect from the Commission.

HISTORICAL BENEFITS OF NRC RESEARCH

Since its inception, the NRC has expended over $2 billion (actual dollars) on research. Research has led to numerous important technical contributions to the NRC's regulatory program and nuclear safety. Several examples follow:

- In the thermal-hydraulics area, extensive research has confirmed that emergency core cooling systems would adequately respond to the worst credible loss-of-coolant accidents, resulting in revision to Appendix K, with a potential avoided capital cost of about $8 billion (Ref. 7). Later, improved methods of analysis provided guidance for responding to questions arising from the TMI-2 accident about plant operation, and have permitted optimizations in reactor systems and operations.

- Several elements of the plant aging research program have led the way in assessing the effects of aging on nuclear power plant components and structures. They have also led to the development of examination and testing techniques and the identification of the essential elements for managing the effects of aging. The results of these research elements constitute the principal technical basis for addressing the aging-related issues associated with nuclear plant life extension and license renewal.

- In the geophysics and seismic areas, NRC-sponsored research programs have provided better understanding of the Eastern U.S. seismicity, which has permitted more realistic assessment of risk from earthquakes.

- In the area of materials science, NRC-sponsored research has provided means to improve and ensure the reliability of inspection methods and has provided key information in managing problems of stress corrosion cracking in BWRs. Additionally, research has provided the means for dealing with the pressurized thermal shock issue. Other research has made it possible to improve reactor safety by justifying the elimination of unnecessary pipe supports.

* NRC-sponsored research has led the way in development of methods for risk analysis. In addition, research has made it possible for the NRC to come to grips with severe accident questions.

Beyond these technical accomplishments is another benefit which is not always explicitly recognized, yet is as important as the others. We believe it to be generally accepted that the NRC's research program has been an important contributor to the high technical quality of the staff. The research program has not only developed important safety information, but has attracted capable people to work for the NRC and its contractors, and has provided a resource of technical expertise to all activities of the agency.

REASONS FOR CONTINUING A COMPREHENSIVE RESEARCH PROGRAM

Important questions about nuclear safety and regulation remain unanswered. Applications of nuclear energy involve demanding technologies, and society expects nuclear activities to be carried out to extremely high standards of public and environmental safety. While analysis indicates that the NRC has been largely successful in its task of ensuring safe practices, significant uncertainties in risk predictions and lack of understanding of certain important phenomena remain. These involve technical areas such as components and materials performance, seismic risk, accident management, severe accident phenomena, and human behavior. Continuing research can gradually provide information and understanding that will be valuable in dealing with these questions and uncertainties.

In addition, it is necessary to maintain the technical quality and credibility of the NRC staff. We were told that the average age of the research staff is now about 50. Vital and consistently funded programs will retain the contributions of experienced researchers and attract capable new people to the agency, in both research and nonresearch positions.

Many of the manifestations of several years of decreasing research funding are already visible:

* Important research programs are being curtailed or terminated.

* The national laboratories are systematically moving their better people to more attractive programs.

* RES is having difficulty in attracting competent technical personnel with research experience, which has led to an overall reduction in quality.

* The results of several expensive experimental programs have been lost.

- University programs have essentially ceased to exist in most areas.
- The role of RES as a world leader in research has diminished.
- The use of large-scale and separate-effects facilities has ended.
- RES participation in major cooperative foreign experimental programs is diminishing.

CONCLUDING REMARKS

It is difficult to establish the proper magnitude of support for research. Two aspects should be considered.

First is the absolute magnitude. In 1975, NRC research was funded at $61 million. In 1981, research funding had increased to $197 million, which was about $130 million in 1975 dollars. In 1991, the budget calls for about $78 million for NRC research which is about $36 million in 1975 dollars. Appropriate funding for a research program must be sufficient to retain vitality in programs, personnel, and facilities. What is appropriate depends on a number of factors, many of them imponderables. The nature of important research questions, the existence or nonexistence of appropriate facilities, results of early research, and experience in plant operation are among them. In the face of these uncertainties, the Commission must make judgments about funding research. Our judgment is that the present research funding level is below the minimum. If there are further reductions, RES will not be able to support and maintain an effective research program.

The fraction of the total NRC budget allocated to research is also an important consideration. It is a measure of the extent to which research programs can be expected to help maintain the technical expertise of the agency. We mentioned above that the research budget has been reduced from over 40 percent in the earlier years of the agency to about 16 percent in 1991, and that may be further reduced by the Congress. We believe there is evidence that this is too low and suggest that a guideline of at least one-quarter of the agency budget is more appropriate for a viable research program.

Finally, we suggest that you not take just our word for it. The agency has in place an excellent panel of experts to advise the RES Director, namely the Nuclear Safety Research Review Committee. We suggest that they focus more on their primary mission, which is to advise on general safety research philosophy and long-range strategy, rather than on the details of specific ongoing research programs. They should consider questions of what might constitute a viable research program, in terms of the technical areas and funding requirements, both absolute and relative.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. ACRS Report dated March 15, 1989, from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: Proposed Severe Accident Research Program Plan.

2. ACRS Report dated June 7, 1988, from David A. Ward, Acting ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Research Related to Heat Transfer and F' d Transport in Nuclear Power Plants.

3. ACRS Letter dated December 8, 1987, from William Kerr, ACRS Chairman, to Victor Stello, Jr., EDO, Subject: ACRS Comments on Memorandum from Victor Stello, Jr., EDO, dated October 7, 1987, Regarding the Embrittlement of Structural Steel.

4. ACRS Report dated September 16, 1987, from William Kerr, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: ACRS Comments on Code Scaling, Applicability and Uncertainty Methodology for Determination of Uncertainty Associated with the Use of Realistic ECCS Evaluation Models.

5. ACRS Letter dated July 15, 1987, from William Kerr, ACRS Chairman, to Victor Stello, Jr., EDO, Subject: ACRS Comments on the Embrittlement of Structural Steel.

6. ACRS Report dated July 15, 1987, from William Kerr, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: ACRS Comments on Draft NUREG-1150, "Reactor Risk Reference Document."

7. Letter dated February 8, 1985, from E. P. Rahe, Jr., Nuclear Safety Manager, Westinghouse Electric Corporation, to D. F. Ross, Office of Nuclear Regulatory Research, NRC, Subject: LOCA Margin Benefits.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 24, 1990

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.

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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 i 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

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 26, 1990

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

Dear Chairman Carr:

SUBJECT: EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 358th, 359th, and 360th meetings of the Advisory Committee on Reactor Safeguards, February 8-10, March 8-10, and April 5-7 and 18-19, 1990, we discussed with representatives of the NRC staff the staff's positions and recommendations concerning the evolutionary light water reactor (ELWR) certification issues ' ntained in SECY-90-016 (Ref. 1). During some of these meetings, had the benefit of discussions with representatives of the Electric Power Research Institute (EPRI) and the General Electric Company. We also had the benefit of the documents referenced.

We were told by the staff that the positions for which they are seeking Commission approval are described in the underlined portions of the enclosure to SECY-90-016, entitled "Evolutionary ALWR Certification Issues." Unless indicated otherwise, our comments relate to these staff positions. Our comments and recommendations on the staff positions are given below.

I. GENERAL ISSUES

1. Evolutionary LWR Public Safety Goals

The NRC staff has concluded that the quantitative goals submitted for Commission consideration in draft SECY-89-102 (Ref. 2) are acceptable for ELWRs. The staff notes that both public safety goals in the EPRI ALWR Requirements Document (Ref. 3) and the ABWR Licensing Review Basis Document (Ref. 4) are considerably more restrictive than the large-release guideline defined in draft SECY-89-102. The staff further notes that additional Commission guidance on quantitative safety goals will assist the staff in its continuing assessment of ELWRs.

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ENCLOSURE 4

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We believe, as stated in our previous reports (e.g., ACRS report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs, dated July 20, 1988), that the Commission's Safety Goal Policy is appropriate guidance for regulatory decisions relating to ELWRs, other advanced reactors, and the operating plants. We regard it as not inappropriate that applicants should work to tighter standards when it serves their purposes, but we do not believe it is appropriate that the NRC should require such standards. In its Safety Goal Policy the Commission, in effect, said it would regulate to a level of safety that is adequate, not to the highest level that is possible.

2. Source Term

This issue is dealt with by a proposal to assure that evolutionary designs meet the requirements of 10 CFR 100 (Reactor Site Criteria). The requirements of this regulation include a limit on doses experienced by an individual at the exclusion area boundary, and at the boundary of the low population zone during the course of an accident. In calculating these doses, the instructions in 10 CFR 100 prescribe that the fission products released to the containment must be those which would be expected from accidents which "result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." For plants currently operating, regulatory guides have delineated specific, but somewhat arbitrary, quantities of fission products that are acceptable to the staff in calculating the leakage from containment and the resultant doses at the specified boundaries.

In contrast, for the ELWRs, the staff proposes to explore the specification of a source term on a case-by-case basis, rather than using the arbitrary source term prescribed in the past. Since the issue of siting of these plants is not yet resolved, and since revisions to 10 CFR 100 are being considered, there may be no alternative to proceeding as the staff proposes, however awkward it may seem.

However, we can make no informed judgment concerning the appropriateness of the procedure until we know more about the criteria to be used in the selection of a source term, and the results of its application.

II. PREVENTATIVE FEATURE ISSUES

3. Anticipated Transients Without Scram (ATWS)

The staff recommends that the Commission approve the staff's position that diverse scram systems be required for the ELWRs.

It appears to us that a design that can ride out an ATWS without serious damage is feasible for PWRs and is preferable to a scram system with diverse logic, which has a reliability calculable, at best, with large uncertainty. We recommend that the staff permit demonstration that the consequences of an ATWS are acceptable as an alternative to a diverse scram logic. The uncertainty in such a demonstration is probably considerably less than that in demonstrating that the contribution of an ATWS to risk is made acceptable by installation of a diverse scram logic system.

4. Mid-Loop Operation

We have been told previously of evidence that events initiated during mid-loop operations may be major contributors to risk in PWRs. However, shutdown operations are generally not accounted for in PRA studies, such as those reported in NUREG-1150 (Ref. 5), so the risks are not well quantified. For the operating plants, this issue has been dealt with through resolution of Generic Issue 99 (Improved Reliability of RHR Capability in PWRs). For the ELWRs, the staff recommends that PWR applicants propose design features to ensure high reliability of the shutdown decay heat removal system.

We agree with the staff's proposal, but recommend that more specific requirements be considered for mid-loop operation:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low-liquid levels at the junction of the DHR system suction lines and the RCS
- Provisions to ensure availability of reliable systems for decay heat removal
- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the DHR system suction lines and the RCS
- Provisions for maintaining containment closure or for rapid closure of containment openings

5. Station Blackout

The Station Blackout Rule (10 CFR 50.63) requires that each light-water nuclear power plant licensed to operate must be able to withstand for a specified duration, and then recover from, a station blackout as defined in 10 CFR 50.2. This rule permits the utilities to submit alternative methods for coping with station blackout. This rule also states that a method based on an alternate ac power source, as defined in 10 CFR 50.2, will constitute an acceptable capability.

For the ELWRs, the staff recommends that the Commission require the installation of an alternate ac power source as the only basis taken to demonstrate compliance with 10 CFR 50.63. The staff recommends that the alternate ac source have capacity to supply power for one safety train, including one complete set of normal safe shutdown loads, and that it be of diverse design. The alternate ac power supply must be designed to serve any safety train when needed, thereby serving as an additional backup power supply for the Class IE power supplies. The staff has stated that the diversity requirement will not preclude use of diesel generators, even though diesel generators are used for the Class IE emergency power supplies.

Although taken by itself this may seem to be desirable, it has not been demonstrated that it is required to conform to the safety goal. Nevertheless, we endorse the staff's recommendation.

6. Fire Protection

The staff concluded that the fire protection issues raised through operating experience and the Individual Plant Examination for External Events (IPEEE) Program (Ref. 6) must be resolved for the ELWRs. To accomplish this, the staff is proposing that the current NRC guidance for fire protection be enhanced as described by the staff during the March 27, 1990, meeting of our Subcommittees on Extreme External Phenomena and Severe Accidents. The enhancements proposed by the staff when combined with the requirements of 10 CFR 50.48 (Fire Protection) without exception and the guidance provided by the Standard Review Plan Section 9.5.1 (Fire Protection Program) should constitute an acceptable basis for prescribing fire protection features for the ELWRs.

The proposed enhancements represent a significant improvement in physical separation requirements and in the need to consider the effects of smoke, heat, and fire suppressant migration into other areas. In particular, redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommend that the proposed enhancements include separation of environmental control systems.

The fire-risk issues that were examined in the Fire Risk Scoping Study (Ref. 7), however, are not fully addressed in SECY-90-016. They should be.

We agree with the staff's recommendation for resolution of this issue with the above caveats.

7. Intersystem LOCA

The staff's position is that designing low-pressure systems to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

III. MITIGATIVE FEATURE ISSUES

8. Hydrogen Generation and Control

The staff recommends that the ELWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of water with 100% of the fuel cladding surrounding the active fuel. (Note: This is not 100% of the fuel rod cladding, nor does it include other metal in the core which could produce hydrogen if it were heated to a red heat in the presence of steam.) There is substantial uncertainty in establishing the amount of hydrogen that might be formed in a severe accident. We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform concentrations of hydrogen greater than 10%. The EPRI ALWR Requirements Document specifies 13%. We are not aware of any experimental or analytical work that demonstrates that the detonation of hydrogen at the 10%, 13%, or some other level could damage the integrity of the containment and essential components. It is our impression that the effect, if any, is something that experts dealing with gas explosions can calculate with reasonable confidence. We suggest that the staff seek further technical information on possible effects, including stratification, before establishing a limit for the average hydrogen concentration.

9. Core-Concrete Interaction - Ability To Cool Core Debris

The staff proposes that the ELWR designs provide sufficient reactor cavity floor space to enhance debris spreading, and provide for quenching of the debris in the reactor cavity. Quantification of what constitutes sufficient reactor cavity floor space is still an open question, as is the means by which one quenches the core debris. The resolution of this issue will require engineering judgment as many of the physical processes are not fully understood. We agree with the staff's recommendation.

10. High-Pressure Core Melt Ejection

To cope with the possible effects of direct containment heating (DCH), the staff concludes ". . . that ELWR design should include a depressurization system and cavity design features to contain ejected core debris."

This is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is probably the preferred approach.

11. Containment Performance

The staff recommends that a containment performance guideline, expressed as a conditional containment-failure probability (CCFP) of 0.1, be used in evaluation of the ELWR designs. As an alternative, the staff proposes a deterministic performance goal that it believes would offer comparable protection.

We have previously recommended (ACRS Comments on An Implementation Plan For The Safety Goal Policy, dated May 13, 1987) such a quantitative guideline for containment performance as a part of the implementation of the Safety Goal Policy. However, this should be regarded as guidance to the NRC staff in its development of requirements for applicants. Merely passing on this guidance to applicants is not enough because the definition of CCFP is too imprecise. The deterministic performance criterion for containment systems suggested by the staff is also difficult to interpret.

We have undertaken an effort (ACRS report on Containment Design Criteria, dated March 15, 1989) to propose containment design criteria for future plants. But, as we said at the beginning of our study, we did not expect that it would directly affect the certification of the ELWR designs. This was, to some extent, because we recognized that our study would take some time to complete, but principally because the ELWR designs are now essentially complete and have been for some time.

We understand that the staff, assisted by the Brookhaven National Laboratory, is developing a regulatory guide that would serve as a basis for review of ELWR containment performance. We believe that the staff proposal will be adequate for ELWR review if it is supported by an appropriate regulatory guide developed on a timely schedule, and if it can be reasonably demonstrated that a containment that meets this guidance has a CCFP of not more than 0.1.

12. ABWR Containment Vent Design

During our April 5-7, 1990 meeting, we heard presentations from the staff and the General Electric Company regarding the staff's proposal that the Commission approve the use of severe accident design features that include a containment overpressure protection system in the ABWR design. We recommend that use of a containment overprotection system be approved subject to the results of the regulatory review.

13. Equipment Survivability

The staff recommends that features provided in the ELWR designs that are intended only for severe accident protection (prevention and mitigation) need not be subject to 10 CFR 50.49 (Environmental Qualification Requirements), 10 CFR 50, Appendix A (Redundancy and Diversity Requirements), and 10 CFR 50, Appendix B (Quality Assurance Requirements). However, the staff will require that mitigation features must be designed so there is "reasonable assurance" that they will perform their intended function in the severe accident environment and over the time span for which they are needed. Further, the staff proposes that at least one train of features provided for design basis accident protection, but also relied upon for severe accident protection, must be able to survive severe accident conditions for the time period that is needed to perform its intended function with "high confidence." In addition, the staff proposes to require that severe accident mitigation equipment be capable of being powered from an alternate power supply, as well as from the normal Class IE on-site systems.

To accomplish "reasonable assurance" and "high confidence," the staff will require that severe accident protective features use high quality industrial grade components which will be selected for the service intended and qualified by analysis or tests.

We endorse the staff's position. We note, however, that in this instance the staff's position includes much more than the underlined portions of the enclosure to SECY-90-016.

IV. NON-SEVERE ACCIDENT ISSUES

14. Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

The staff states that it has not yet developed a position on this issue that can be applied generically to all future designs and recommends that the Commission approve a design-specific approach. We have no objection to the staff considering exemptions to the requirement that the OBE be at least one-half the SSE, where this can be justified. We note that this has been done in the past for 14 plants at 9 sites, but in each case using site-specific data. Other bases for justification may have to be provided for un-sited standard plant designs.

In the longer term, we recommend that the staff and the industry attempt to develop a position that can be defined generically. One approach worthy of study would be to abandon the use of two earthquake levels for the design of structures, systems, and components. Instead, the design could be based only on the SSE, with appropriate load factors and limit states, and a smaller but more likely earthquake could be established as a threshold for plant shutdown and inspection.

15. <u>Inservice Testing of Pumps and Valves</u>

The staff proposes that certain aspects of the testing and inspection of pumps and valves be enhanced to ensure the necessary level of component operability for the ELWR designs. We endorse the staff's proposal with the following clarification and additions:

- Although not stated explicitly, we were told during the March 7, 1990 meeting of our Subcommittee on Mechanical Components that the staff intends to apply the requirements of Generic Letter 89-10 (Ref. 8) to the ELWR plants as well. We endorse this intention.

- We recommend that the staff's requirement for full-flow testing capability be extended to other safety-related valves (e.g., MOVs) not just check valves. The requirement for flow testing of MOVs is included in Generic Letter 89-10.

- We recommend that the staff resolve the issue of check valve testing and surveillance requirements for existing LWR plants and indicate how it is to be applied to the ELWRs prior to issuing the FDAs.

- We recommend that the staff be encouraged to entertain proposals from the FDA applicants regarding alternative ways of meeting the in-service testing and surveillance requirements.

Additional comments by ACRS Members Harold W. Lewis and James C. Carroll and ACRS Members William Kerr, David A. Ward, and James C. Carroll are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

<u>Additional Comments by ACRS Members Harold W. Lewis and James C. Carroll</u>

Apart from one paragraph submerged as part of Item 1, this letter endorses the scattershot approach the staff has taken to the important question of regulation of new reactors. It therefore deserves to be called Camel II, in deference to the Committee's similar letter of January 15, 1987. The differences are that this list has in fact had more careful consideration, and that its elements originated with specific staff proposals. Indeed, in many cases the genealogy can be traced to industry initiatives, and the staff is simply proposing to make mandatory those things that the industry has previously proposed to do on its own. None of this pays the slightest attention to the Commission's Safety Goal Policy, nor is there any hint of an effort to seize this opportunity to move the regulatory process in the direction of coherence and consistency. This is a pudding without a theme.

Let us then try to provide some perspective, since the Committee has chosen not to do so.

The Committee has often commented on the central role of the Safety Goals in providing a focus and objective for the body of regulation. Since this list sets the tone for the licensing of the next generation of light-water reactors, it is particularly important that its relation to Commission policies, especially the Safety Goal Policy, be clear. At the risk of repetition, we, and we believe the Committee, have never urged that specific regulatory decisions (such as these) be judged individually in the context of the Safety Goals, but only that the body of regulation be judged in that light. Individual decisions must still be made deterministically, with expertise and good judgment, but as part of a coherent overall body of regulation. Still, fifteen items come close to being a "body", and it is informative to see the role of the Safety Goals in the formulation of the staff recommendations. The Safety Goal policy, and other commission policies, are supposed to provide the glue that binds the whole structure together.

In effect, the staff says that it has proposed to the Commission a set of new safety goals (SECY-89-102), the Commission has not acted on them, either way, and therefore the staff will use them as if the Commission had approved. While we sympathize with the staff predicament, we think that is entirely inappropriate. The staff proposals include such things as a core-damage probability of 1E-5 per reactor-year, a "large accident" probability of 1E-6 per reactor year (with a bizarre definition of large accident), and a so-called conditional containment-failure probability. Not one of these has been approved by the Commission, yet the staff has used them in formulating its proposed policies on these items. It has rationalized this usurpation of power by asking for Commission action on SECY-89-102, and by stating that its own safety goals are "consistent" with those of the Commission. Of course any set of goals more stringent than yours will be consistent with your own, and acceptance of this argument will mean that the staff can regulate beyond your policies, more or less at will. That is precisely the situation your original goals were intended to foreclose. The Committee has often recommended that your Safety Goals be used as a final statement of "how safe is safe enough", not as a rigid minimum level of safety, beyond which the sky is the limit. Of course the industry may well have good reason to go further, but that is another matter.

In addition, as your own OGC has pointed out in SECY-90-016, this has the potential to open a Pandora's box, in which each party to a licensing proceeding may be able to claim the rights the staff claims--to insist on improvements beyond the rules. You will have to face this problem at some time, and the sooner the better.

We do not wish to understate the difficulty involved in translating a safety-goal policy into a workable body of regulation. The Committee has written you of its own recommendations for an organized approach to that problem, but we believe it can and should do more. Nuclear safety is not helped by letting that problem fester--the fact that it is difficult is no excuse for inattention. It is too much to expect regulation to be coherent and rational in the absence of an objective for that regulation.

We do think it was useful for the Committee to respond to your specific request for technical help on the fifteen questions posed, but you should recognize that this was done in the absence of a measuring rod. Each item was therefore judged on its own, and the Committee has turned its back on the opportunity to respond in a structured and coherent way. Any one of these items might have come out differently if it had been measured against an underlying rationale. In our view, the Committee has forfeited a chance to be of real service to both you and the public.

Additional Comments by ACRS Members William Kerr, David A. Ward, and James C. Carroll

By the "rulemaking" approach to design certification the Commission has sidestepped the development of revisions to regulations that would reflect knowledge gained from experience and research over the last ten or more years. As a result, important new requirements are being imposed on applicants through a variety of staff actions and reactions. This is a loosely controlled process in which major policy decisions are made without an appropriate intensity of review. Contributing to the lack of discipline is what we believe to be a serious ambiguity in the Commission's policy on advanced reactors. The Commission has said it expects future reactors to be safer. But, whether this is a mandate or simply an expectation that a maturing industry will produce safer plants is not clear. The staff has interpreted it as a mandate and has translated this into an unauthorized extension of the safety goals. This is despite the statement in NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," published June 1988 (p. 4-1) that "the Commission expects but does not require enhanced safety margins other than those that may be required by the Safety Goal Policy." The Commission should not indefinitely postpone the development of a modern set of regulations. Only in this way will a proper balance be struck between adequate protection of the public health and safety and the advantages to the public that can come from efficient development of the nuclear power option.

References:

1. SECY-90-016, memorandum dated January 12, 1990, from J. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements

2. Draft SECY-89-102, memorandum dated March 30, 1989, from V. Stello, Jr., Executive Director for Operations, NRC, to the Commissioners, Subject: Implementation of Safety Goal Policy

3. Electric Power Research Institute (EPRI), Advanced Light Water Reactor Requirements Document (Chapters 1 through 13), issued December 1987

4. General Electric Company, Advanced Boiling Water Reactor Licensing Review Basis Document, issued August 1987

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

6. Memorandum dated March 8, 1990 from W. Minners, Office of Nuclear Regulatory Research, NRC, to R. Fraley, ACRS, Subject: Proposed Generic Letter on Individual Plant Examination for Severe Accident Vulnerabilities Due to External Events (IPEEE) and Supporting Documents (Predecisional)

7. Sandia National Laboratories, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NUREG/CR-5088, published January 1989

8. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," issued on June 28, 1989 to licensees for all power reactors, BWRs, PWRs, and vendors in addition to General Codes applicable to generic letters.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 16, 1990

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

Dear Chairman Carr:

SUBJECT: NRC RESEARCH ON ORGANIZATIONAL FACTORS

During the 364th meeting of the Advisory Committee on Reactor Safeguards, August 9-11, 1990, we reviewed, at the request of the NRC staff, the Commission's program of research related to organizational factors. Our Subcommittee on Human Factors also reviewed this matter during a meeting on July 31, 1990. During these meetings, we had the benefit of discussions with representatives of the NRC staff and its contractor, Brookhaven National Laboratory. We also had the benefit of the document referenced.

The NRC research program on organizational factors is intended to provide a scientific basis for improving the organizations responsible for operating nuclear power plants. The Commission has expressed concerns about the feasibility of such research and has asked to be briefed on the status of the program. We recognize the reasons for these concerns; the issues are difficult and are outside the mainstream experience of the NRC and the industry. This does not mean the issues should be ignored since they are of vital importance to nuclear power plant safety.

The Commission, the ACRS, and the nuclear power industry have recognized for the past several years that the quality of management associated with nuclear power plant operations is of cardinal importance to plant safety. During our August meeting, Dr. Herbert Kouts, the Chairman of the Special Committee to Review the Severe Accident Risk Report (NUREG-1150), summarized the results of that Committee's review. In response to a question, he noted his support for continuing NRC research on human reliability analysis, in particular research on the influence of organizational factors.

An important component of good management is an effective plant organization. Little quantitative basis exists for optimizing plant organizational design with respect to safety. This contrasts with the comprehensive technical bases that support many other aspects of nuclear power plant safety and design.

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ENCLOSURE 5

Under the present research program, the staff and its contractors are studying organizations ranging in scope and size from the total licensee force at a nuclear power plant site to the shift crews and smaller teams that perform essential functions of operation and maintenance. Depending on the results of this work, the program may be expanded at a later time to study the effect of utility and other organizations external to the plant. It has been recognized that complex nuclear power facilities are operated and maintained by teams of people, not by individuals. Therefore, something more than training and licensing of individual operators is necessary to ensure plant safety.

The research program described to us by the staff appears to be focused on agency needs and can make a contribution to future improvements in the effectiveness of nuclear power plant organizations. We do have a concern that the research program seems to be directed toward the need to consider operator performance in PRAs in a more quantitative manner. This is a desirable ultimate goal; however, we believe that more emphasis should be placed on communicating to nuclear power plant licensees the insights developed on effective managerial approaches.

Continued support and encouragement for this research program from the Commissioners and the NRC staff management will be necessary. The research staff and its contractors are undertaking a difficult and pioneering effort. We will follow progress of the program with interest.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I was less impressed than my colleagues. All of American industry is concerned about the effect of organization on productivity and effectiveness - courses are taught, books are written, etc. I don't believe the need is for research, but for application of what is known to the NRC's regulatory problems. Not only the industry, but NRC itself, could benefit. No one doubts the importance of the subject.

Reference:
Draft SECY paper to the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Organizational Factors Research Progress Report (Predecisional), transmitted by Memorandum dated July 5, 1990 from Tom Ryan, Office of Nuclear Regulatory Research, NRC, to Herman Alderman, ACRS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 12, 1990

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

Dear Chairman Carr:

SUBJECT: YANKEE ROWE REACTOR PRESSURE VESSEL INTEGRITY

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we discussed the degree and consequences of the Yankee Rowe reactor pressure vessel embrittlement due to neutron irradiation. Our Subcommittee on Materials and Metallurgy discussed this matter with representatives of the NRC staff and the Yankee Atomic Electric Company during a meeting on September 5, 1990. We also had the benefit of the documents referenced.

It has recently come to the staff's attention that the reference temperature nil ductility transition (RT_{NDT}) of parts of the Yankee Rowe pressure vessel may substantially exceed the temperature limits for action delineated in the pressurized thermal shock (PTS) rule (10 CFR 50.61). The main reason is that the Yankee Rowe core inlet temperature is about 50°F lower than that of other plants. Another reason is the higher nickel content of the lower vessel plate. These increase the rate of rise in RT_{NDT} with fast neutron irradiation.

The exact value of RT_{NDT} for the vessel is uncertain because of:

- Uncertainty in the copper and nickel content of the circumferential weld near the reactor vessel beltline.

- The absence of surveillance data for areas that appear to have the largest shift in RT_{NDT}, namely the circumferential weld and the lower plate of the vessel.

Assurance of vessel integrity is further hindered by:

- The absence of any inservice inspection for flaws in the reactor vessel beltline region. Such inspection has been infeasible due to the design of the vessel internals.

- Relatively low toughness (low upper shelf energy) of the plate and welds near the core.

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ENCLOSURE 6

Analysis of the various safety issues involved leads to the conclusion that PTS is the issue of most concern. One bright spot in this picture is that several features of the plant's design make it less susceptible to overcooling events than more modern plants.

The licensee and the staff have both arrived at estimates of the shift in RT_{NDT}. Both agree that the circumferential weld and the lower plate of the pressure vessel have the highest RT_{NDT}. However, in each case their estimates differ by about 150°F. The licensee's representatives argue that due to the particular microstructure of the steel in the vessel, the shift in RT_{NDT} is independent of irradiation temperature and nickel content. We do not believe these arguments are valid, and agree with the staff that temperature and nickel effects must be included in a valid estimate of the shift in RT_{NDT}. An additional difference between the staff and the licensee concerns estimates of the copper content of the circumferential weld. There being no measurements for the composition of the circumferential weld and a large spread in copper values found in other plants, the staff prefers to choose a bounding value. The applicant chose more of an average value. In view of the uncertainty in the value for the Yankee Rowe vessel, we would choose the staff's bounding value.

Given that RT_{NDT} values for parts of the vessel probably exceed those requiring action under the PTS rule, is there significant risk in operating the plant? The low probability of a PTS challenge leads to a low risk, even with a high RT_{NDT}. Thus, we agree with the staff that operation for one more cycle is acceptable, provided the licensee initiate an active program to better characterize the material in the vessel near the reactor vessel beltline. To do this the staff requires determination of the composition of the circumferential weld metal in the beltline by removing samples from the weld and development of an inspection method for the beltline welds and plate to depths of an inch below the inside surface of the vessel. Both of these have been required by the staff for completion before the startup of the 22nd fuel cycle (now scheduled to begin in early 1992). It is not clear that both can be achieved in that time, but certainly they should be accomplished in two fuel cycles.

The staff also requires "tests on typical Yankee Rowe base metal" to determine the effect of irradiation, austenitizing temperature and nickel content on embrittlement. It is doubtful that any tests that the licensee could perform during the next fuel cycle would convince us that the effects of temperature and nickel on embrittlement are substantially different from those established by the much more extensive studies already available. The effects are not well understood, and we believe prudence dictates tending more toward bounding values rather than best estimates based on limited new data that may become available.

However, the above will not adequately address the long-term operation of the plant. This is the lead PWR plant in the industry's Plant Life Extension (PLEX) program, and long-term operation with such large uncertainties in vessel integrity is unacceptable. The extended operation of this plant would be acceptable only if:

- A state-of-the-art ultrasonic inspection can be done on essentially all of the radiation affected inner surface of reactor pressure vessel, e.g., one that complies with Appendices VII and VIII of Section XI of the ASME Code. This inspection should also check for significant thinning in the lower head as a result of loose parts (irradiation capsules). Continued operation would be dependent on the absence of significant flaws.

- A reanalysis of the PTS question is made using well established compositions for the material in the beltline region, or using limiting values of copper and nickel. This analysis should also include the fact that the crack arresting ability of such material will be lower than more modern steel because of its low upper shelf energy. Such an analysis must show acceptable risk.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. Letter dated July 5, 1990 from John D. Haseltine, Yankee Atomic Electric Company, to Richard Wessman, NRR, transmitting Reactor Pressure Vessel Evaluation, dated July 9, 1990

2. Letter dated August 31, 1990 from Thomas E. Murley, NRR, to Andrew C. Kadak, Yankee Atomic Electric Company, Subject: Yankee Rowe Reactor Vessel, with Enclosure

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555

October 11, 1990

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: NRC COMPUTER CODES AND THEIR DOCUMENTATION

During the 366th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 1990, we continued our deliberations on the subject of the development of NRC's computer codes and their associated documentation. This topic was previously discussed during our 365th meeting, September 6-7, 1990. It was also discussed during a joint meeting of the Decay Heat Removal Systems and Thermal Hydraulic Phenomena Subcommittees held on August 28, 1990, in Idaho Falls, Idaho.

A portion of the regulatory process depends heavily on the results of calculations done for the NRC by the national laboratories or other contractors. The codes used for these calculations range from thermal hydraulic codes like RELAP5 or TRAC to severe accident codes like SCDAP or MELCOR. Many of these codes are poorly documented, thus leaving one unable to determine either their capabilities, or perhaps more importantly, their limitations. In some cases, it appears that even the cognizant NRC staff representatives are not sufficiently knowledgeable of a given code's content.

The NRC has a responsibility to make the basis for its computer codes as scrutable as it requires of the industry. Many code developers consider the documentation phase of the code development process distasteful. Nevertheless, the RES program managers should see that adequate documentation is provided, particularly for models and correlations and for developmental assessment. We have seen evidence that they have not done so. One of the central problems is the tendency to defer the preparation of such documentation until the end of the program. Although such a deferral may be understandable, given the natural progression of the development program, it is essential that program management ensures that documentation is provided in a timely manner and within budget.

The August 28, 1990 Subcommittee meeting was held to review the nearly completed work related to the development of the RELAP5/MOD3 thermal hydraulic code. Discussions during this meeting provided

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ENCLOSURE 7

evidence that the associated documentation was incomplete. The contractor personnel were new to the program and not well enough acquainted with the code's details to respond to questions from the Subcommittee. The potential exists for similar problems with the completion of the development program for the TRAC-PF1/MOD2 code. Deliberate attention by RES program managers is needed to ensure the documentation for these codes is adequate.

Another example that illustrates our concern involves the thermal hydraulic code known as REMIX, which has been used by the NRC to evaluate the potential for pressurized thermal shock given certain accident scenarios. Relevant experimental data were generated as part of the cooperative 2D/3D program, among the United States, Germany, and Japan, and these data were compared with REMIX code calculations. Although a Research Information Letter citing this work was issued in 1988, a report documenting these comparisons has never been issued by the NRC. Recent review of the Yankee Rowe pressurized thermal shock issue would have been well served by knowing how well the downcomer fluid temperature can be predicted, using a code such as REMIX, at the beltline welds following a small break loss of coolant accident.

Many millions of dollars have been spent on the development of the computer codes used by the NRC, nearly $20 million for RELAP5 alone. The NRC should make sufficient funding and resources available to ensure that the documentation associated with the development of the agency's codes is adequate.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Reference:
Memorandum dated August 24, 1988, from Eric S. Beckjord, Office of Nuclear Regulatory Research, for Thomas E. Murley, Office of Nuclear Reactor Regulation, Subject: "Research Information Letter No. 155, Full Scale Fluid Mixing Test Results in Support of Pressurized Thermal Shock Resolution."

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555

November 15, 1990

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

Dear Chairman Carr:

SUBJECT: REVIEW OF NUREG-1150, "SEVERE ACCIDENT RISKS: AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS"

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we discussed the second draft of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." The Committee had previously discussed this matter with the staff and its consultants and with Dr. Herbert Kouts, Chairman of the Special Committee to Review the Severe Accident Risk Report. Our Subcommittees on Severe Accidents and Probabilistic Risk Assessment discussed this report during a number of joint meetings with members of the staff, Sandia National Laboratories (SNL) and the American Nuclear Society (ANS) Special Committee (Dr. Leo LeSage, Chairman). We also had the benefit of the documents referenced.

1. <u>INTRODUCTION</u>

In this report, we first offer some general comments. We then offer recommendations concerning the publication of NUREG-1150 and provide comments and cautions concerning interpretation or use of some of the components of this document. And finally, we provide more detailed comments on some key parts.

We have reviewed the reports prepared by the ANS Special Committee and by the Special Committee to Review the Severe Accident Risk Report appointed by the Commission and found them helpful. We have no serious disagreements with either of these reviews, nor with their findings.

2. <u>GENERAL COMMENTS</u>

The work described in this draft of NUREG-1150 is an improvement over that described in the first version entitled, "Reactor Risk Reference Document." Many previously identified deficiencies in the expert elicitation process have been corrected. The exposition and organization of the report have been improved. The presenta-

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ENCLOSURE 8

tion of results is clearer. There is considerable information that was not in the original version.

The portion that deals with accident initiation and development up to the point at which core heat removal can no longer be assured is unique, compared to other contemporary PRAs in that a method for estimating the uncertainty in the results has been developed and applied. This method and its application are significant contributions. Although the larger contributions to uncertainty in risk come from the later parts of the accident sequences, this portion is enhanced also by an extensive identification of events that can serve as accident initiators as well as an associated set of hypothesized event trees. This information should be of considerable assistance to licensees in the performance of an Individual Plant Examination (IPE). It should also be useful to plant operators and to designers.

The formulation of a more detailed representation of accident progression after severe core damage begins, and an improved description of containment performance, contribute some additional information to this important area. However, understanding of many of the physical phenomena that have an important bearing on this phase of accident progression is still very sparse, and the report may give the impression that more is known about this portion of the accident sequence than is actually the case.

The part of the sequence that begins with the release of radioactive material outside the containment is treated by a relatively new and unevaluated code system. Furthermore, there is no estimate of the uncertainties inherent in the calculations that describe this part of the sequence. Those who use the quantitative values of reported risk must recognize that these uncertainties are not accounted for in the calculated results.

3. RECOMMENDATIONS

We recommend that the current version of NUREG-1150, with the corrections suggested by several of those who have already reviewed it in detail, be published. However, its results should be used only by those who have a thorough understanding of its limitations. Some of these limitations are discussed in subsequent sections of our report.

Since the supporting documents upon which NUREG-1150 depends could be helpful to those who perform an IPE, we recommend that these also be published as soon as feasible.

Both the Commission and the ACRS have raised questions about generic conclusions that might result from a careful examination of the results of this study. It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost five years and seventeen million dollars to complete. We recommend that the Commission encourage the staff to mine more deeply the wealth of information that has been collected in the course of this study in an effort to identify generic conclusions that might be reached (see Section 5.5 of this letter).

4. COMMENTS AND CAUTIONS CONCERNING USES OF THE MATERIAL IN NUREG-1150

We discuss below certain areas in which the methods or results should be used with caution.

4.1 Differences Among Levels of the PRA

The phenomena which contribute to sequence progression in Level 1 are generally well understood. Power plant or other related experience with system and component performance has provided sufficient data to permit predictions of sequence progression with considerably greater confidence than for those parts of the sequence described in Levels 2 and 3. NUREG-1150 is unique in the amount of effort that went into estimating uncertainties in the calculated Level 1 results. It is our view that the results of Level 1 can be used with more confidence than those of Levels 2 and 3. However, as other reviewers have reported, there are recognized deficiencies in the state-of-the-art treatments of human performance; and this report is not free of those deficiencies. In addition, some possibly important initiators, e.g., those at low power operation or at shutdown, and sequences initiated by fire, are either treated superficially or are neglected altogether.

The Level 2 analyses in NUREG-1150 include more detailed containment event trees than those found in any previous PRA. However, we have some concern that the amount of detail may lead to a conclusion that much more is known about the phenomena in this area than is actually the case.

Since there is a dearth of information concerning many of the phenomena that determine severe accident progression, expert elicitation was used most extensively in the Level 2 portion of the PRAs. There is general agreement that the techniques used for eliciting expert opinion in preparation of the second draft were significantly better than those used for the first draft. However, with insufficient information there can be no experts. Thus, use of the term "expert opinion" in a description of some of the Level 2 work may be misleading. (Further comments about the expert elicitation process are given in Section 5.3). We applaud efforts to improve on the Level 2 treatment of previous PRAs. We nevertheless believe that the results from Level 2 presented in this latest draft must be regarded as having major uncertainties in both calculated mean values and in estimated uncertainties.

The MELCOR Accident Consequence Code System (MACCS) was used for the consequence calculations of Level 3. Use of MACCS is a departure from many existing PRAs that use the Calculation of Reactor Accident Consequences (CRAC) series of codes. MACCS is a relatively new code, still under development. It has been neither benchmarked nor validated. Thus, in addition to the uncertainties inherent in the physical phenomena that enter into consequence modeling, additional uncertainties are introduced by the use of a new and relatively untested code.

No effort was made to estimate the uncertainties in the Level 3 calculations. Thus, the estimates of uncertainties in risk that are given in the report are only those arising from the uncertainties calculated for Levels 1 and 2. It is our judgment that the uncertainties in modeling the consequences of a release can be at least as large as those estimated for Level 2. For example, the health effects, especially for low dose exposures, are subject to large uncertainty, and the exposures themselves depend on actions (e.g., evacuation, sheltering, interdiction of land and crops) for which the uncertainty in prediction is largely unknown.

4.2 Assumptions Made in Screening

Users of the report should be aware of the assumptions made in the screening process for low-probability, high-consequence events. For example, the analysts assumed that the probability of total loss of DC power was less than 1×10^{-7} per year and thus could be neglected. The same assumption was made for loss of all service water. Thus, those who use the results in IPE work should recognize that these assumptions may not be valid for all operating plants.

4.3 Credit for Decay Heat Removal by Feed and Bleed

The success of the feed and bleed operation is highly dependent on human performance. Everyone seems to agree that there are large uncertainties in its treatment in this report. In addition, it is likely that the performance of valves, which must function if this maneuver is to be successful, are not well represented by the data for valve performance used in the calculations.

4.4 Performance of Motor-Operated Valves

There is now a significant body of evidence which indicates that the failure probability used to describe the operation of certain key motor-operated valves is too low. This may have an important bearing on the outcome of several accident sequences described in the report.

4.5 Contribution of Pump-Seal Failure to the Risk of Small Break LOCAs

We believe that more recent information and some new seal designs developed since the study was made would lead to a prediction of risk less than that reported.

4.6 Containment Performance

The lack of information about many of the physical phenomena that determine the performance of a containment system in a severe accident situation is such that only educated guesses can be made for some sequences that might make significant contributions to risk. Although the large number of event trees developed in the containment analyses is indicative of what was hypothesized by the analysts, the amount and quality of information concerning a number of key phenomena that determine behavior at branch points are low. The difficulty of arriving at a result with significant confidence is illustrated by two examples. In the analysis of the performance of the Mark I containment used in early BWRs, the experts in the original study predicted a large conditional probability of early failure. In the second study a different group of experts produced a bimodal distribution because part of the panel concluded that the probability of early failure was high, and part considered it low. A second example is the calculation of risk produced by postulated direct containment heating (DCH). In the first study, the calculated risk due to DCH for PWRs with large dry containments was a major contributor to the total risk. In the second version, its contribution was significantly less. In neither case had there been a major change in the information about relevant physical phenomena available at the time of the first study. Further, we find no consideration of the impact of ex-vessel steam explosions on early containment failure. There is little unambiguous guidance here for a licensee performing an IPE.

5. AREAS FOR SPECIAL COMMENT

In this section, we provide more detailed comments on some areas that appear to us to deserve special attention.

5.1 Fire Risk

The fire contribution to core-damage probability was estimated for two plants using insights gained during previous fire PRAs and studies, the latest methods and data bases developed under NRC sponsorship, and the benefits of extensive plant walkdowns. The methods and data used were probably the best available at the time the reported work was performed. Nevertheless we conclude, on the basis of later information, that the results should be viewed as being incomplete. The models used were not able to take full account of several issues identified by SNL in a scoping study of fire risks that was completed more recently. These are issues that have not been adequately considered in past fire risk studies and may increase the risk. Of particular concern are seismic-fire interactions, adequacy of fire barriers, equipment survival in the environment generated by the fire, and control systems interactions. The PRA for the LaSalle nuclear plant, which is nearing completion, may provide insights concerning the risk importance of these issues.

5.2 Seismic Risk

The seismic PRAs for the Surry and Peach Bottom nuclear plants were performed using two quite different representations of the seismic hazards. The results however, at least for sequences leading to core damage, were similar in terms of which accident initiators and sequences were important. This tends to support the acceptability of using the seismic margin approach rather than a PRA in the search for plant-specific seismic vulnerabilities in the IPE-External Events (IPEEE) program. However, the success of either approach in finding vulnerabilities depends strongly on walkdowns to identify those systems and components to be evaluated. Knowledge of what to look for is derived chiefly from PRAs done on other plants, and these have tended to focus primarily on core damage rather than releases of radioactive material to the environment. Although containments are usually quite rugged seismically, this is not necessarily true for containment cooling systems, containment isolation systems, etc.

Although the two seismic PRAs in NUREG-1150 have been carried through Level 3, these results have not been reported. We believe that these results might provide valuable insights about seismic vulnerabilities of containment systems.

5.3 The Expert Elicitation Process

There is general agreement that the use of expert elicitation in the preparation of the results in this draft of the report is improved compared to that used for the first version. However, we have reservations about some parts of the application of the process. For example, during our discussions of the choice of the participating experts we got the impression that an effort was made to choose participants in such a way that a wide spectrum of viewpoints would be represented. This was defended as proper, based on the assumption that unless this wide spectrum of opinion was represented, the uncertainty in expert opinion would not be appropriately accounted for. We found this argument unconvincing, and would have preferred to see individuals chosen primarily on the basis of their knowledge and understanding of the phenomena being considered. Furthermore, we were told that the budget for the study provided only enough funding to support the participation of about 20 percent of the experts who served on the panels. The remainder were drawn from the NRC staff or from organizations with contractual relationships to the NRC. This biased the selection toward people whose organizations depend upon the NRC for support. We also observe that the membership of the panels seems to have been dominated by analysts in contrast to those who have done significant research on phenomena of importance to the accident sequences being described.

5.4 Source Term Description

The staff, or at least that part of it closely associated with this study, has discarded for future use the Source Term Code Package (STCP) that was one of the resources used by the expert panels in the preparation of NUREG-1150. The expert elicitation method is too resource intensive to be used generally. At this time, only the MELCOR code is available to the staff for source term calculation. Although it appears to be an improvement over the STCP, it is not yet fully developed, nor is it generally available in its current form. Some method for calculating a source term will be needed by the staff and its contractors for performing or reviewing PRAs, as well as for other tasks, such as a revision of the siting rule.

5.5 Lack of General Conclusions

We have asked the staff whether the results reported in NUREG-1150 shed any light on the risk expected due to operation of the population of plants now licensed. With few exceptions, it is the staff's view that one can tell little or nothing about the expected risk of plants not studied from the results of the study of these five plants in NUREG-1150. In spite of these statements, however, those who prepared the report propose that applications will include evaluation and resolution of generic issues and prioritization of future research and prioritization of inspection activities. If, as we were told, the results from the analyses of these plants have little or no generic significance, application of these results must be made with considerable caution.

We believe that the large amount of information collected as input to the calculations made during this study, and the results of the large number of analyses undertaken, must surely permit some more general conclusions to be drawn than we find in this report. For example, the risk calculated for each of the five plants analyzed (although calculated only for internal initiators) falls within the Quantitative Health Objectives (QHOs) set forth in the Safety Goal Policy Statement. Each was designed and constructed and is operating within the rules and regulations promulgated by the Commission. There must be some significance in the fact that plants supplied by a number of different vendors, constructed at different locations, under supervision of different organizations, over a period of more than a decade, with rather different balance of plant configurations, and different containments, nevertheless fall within the QHOs. Is application of the NRC's regulations achieving the objectives of the NRC Safety Goal Policy?

Another area of interest is the risk reduction achieved by some recently promulgated rules. The report indicates that station blackout is a significant risk contributor for three of the plants studied. Answers to questions we asked during our meetings with the staff indicated that some of the plants analyzed had implemented most of the requirements of the Station Blackout Rule, while others had only just begun the process. Could one draw any conclusions from the plants studied as to the risk reduction to be expected from implementation of the Station Blackout Rule? Or could one estimate the risk reduction for some "average" plant? This would be interesting, since in the typical cost benefit analysis associated with backfit it is assumed that some such conclusion can be drawn about plants generally. It would be useful to see what an examination of these five plants would indicate.

The five nuclear power plants chosen for the study were selected partly on the basis of the different types of containment represented. We find little or no discussion of relative containment performance or identification of containment designs that might be expected to have superior mitigation capabilities. For example, in light of the containment being proposed for the Advanced Boiling Water Reactor (ABWR), it would be helpful to have any information or conclusions that were developed during the course of the study as to relative efficacy of the containment being proposed for that design as compared to the Mark I or the Mark III containments. Or, for large dry containments, does the subatmospheric operation of the Surry system provide a substantial decrease in risk (because, for example, of its continuous indication of leak tightness) as compared to a large dry containment operated at atmospheric pressure?

Although it may not be feasible to make major changes in containments of reactors now in operation, it is possible to choose containments with superior mitigation characteristics for nuclear plants not yet constructed. It might even be feasible, as a result of the study, to recommend a containment design that combines the best features of several of the existing systems. If in the course of this study information has been developed that could be used to reduce the conditional failure probability of containment, given severe core damage, the risk uncertainty in new designs might be reduced without requiring any additional studies of core damage progression.

Sincerely,

Carlyle Michelson

Carlyl Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Volumes 1 and 2 (Second Draft for Peer Review), dated June 1989.
2. American Nuclear Society, "Report of the Special Committee on NUREG-1150, The NRC's Study of Severe Accident Risks," L. LeSage (Chairman), dated August 1990.
3. U.S. Nuclear Regulatory Commission, NUREG-1420, "Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)," H. Kouts (Chairman), dated August 1990.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Reactor Risk Reference Document," Volumes 1, 2, and 3, Draft issued for comment, dated February 1987.