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

December 3, 1990

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

Dear Chairman Carr:

SUBJECT: SUMMARY REPORT - THREE HUNDRED SIXTY SEVENTH MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 8-10, 1990

During its 367th meeting, November 8-10, 1990, the Advisory Committee on Reactor Safeguards discussed several matters and completed the reports noted below. In addition, the Committee authorized Mr. Fraley to transmit the memorandum identified below.

REPORTS TO THE COMMISSION

- SECY-90-353, Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor (Report to Chairman Carr, dated November 14, 1990.)

- Review of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (Report to Chairman Carr, dated November 15, 1990.)

MEMORANDUM

- Proposed Final Amendment to 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (Memorandum for Eric S. Beckjord, RES, from R. F. Fraley, dated November 14, 1990.)

  Consistent with the Committee's decision, Mr. Fraley has informed Mr. Beckjord that the Committee members have decided that further review of the Proposed Final Amendment to 10 CFR Part 50.61 is not necessary and that they have no objection to issuing this amendment as a final rule.

INDIVIDUAL LETTER

- Letter by H. W. Lewis to Chairman Carr, dated November 23, 1990

  Based on his review of the statistical analysis used by the NRC staff in its resolution of Generic Issue B-56, "Diesel

Generator Reliability," Dr. Lewis has provided a letter to Chairman Carr commenting on the improper use of statistics by the NRC staff.

Remarks provided by Dr. J. Ernest Wilkins were included in this letter.

HIGHLIGHTS OF CERTAIN MATTERS CONSIDERED BY THE COMMITTEE

- Combustion Engineering System 80+ Design

  The Committee heard presentations by and held discussions with representatives of the NRC staff and of Asea Brown Boveri-Combustion Engineering (ABB-CE) with regard to the following:

  - Staff's comments and recommendations to the Commission on the Licensing Review Basis (LRB) document proposed by CE that are contained in SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor."

  - Design differences (deviations) of System 80+ from the EPRI-ALWR Requirements Document.

  The staff stated that it will continue to proceed with the review of the System 80+ design for areas where information has been submitted and no policy decisions are needed. However, no firm review schedules can be established until a policy decision is made by the Commission on the level of design detail required for design certification.

  The Committee provided a report to the Commission on this matter.

- Regulatory Impact Survey

  Representatives of the NRC staff briefed the Committee regarding their proposed actions to implement, as appropriate, the findings of the Regulatory Impact Survey that was conducted by the senior NRC management teams between September 25 and December 1, 1989.

  Based on the results of the survey, the staff recommends improvement in the following areas:

  - Consideration of the Cumulative Effects of NRC Generic Requirements

  - Scheduling and Control of Inspections, Especially Team Inspections

- Management Expectations, Training, and Oversight of Inspectors

This was an information briefing - the Committee took no action.

- <u>Level of Design Detail for Standardized Nuclear Power Plants</u>

  Members of the NRC staff briefed the Committee regarding the level of design detail that the staff considers necessary for certification under 10 CFR Part 52. The staff has developed SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," to provide recommendations to the Commission with respect to the following:

  - Level of detail required for an essentially complete nuclear power plant design in an application and available for audit for design certification, and for a combined license under 10 CFR Part 52.

  - Applicability of the industry's two-tier approach to design certification.

  - Flexibility to incorporate necessary changes and technological advances while preserving standardization.

  The ACRS Subcommittee on Improved Light Water Reactors is scheduled to hold a meeting on December 4, 1990 to discuss this matter further. This matter is also scheduled for discussion and appropriate action by the full Committee during the December 6-8, 1990 ACRS meeting.

- <u>Westinghouse SP/90 Standardized Plant Design</u>

  The Committee heard presentations by and held discussions with representatives of the NRC staff and the Westinghouse Electric Corporation regarding the draft Preliminary Design Approval (PDA) document for the Westinghouse SP/90 Standard Plant Design.

  This was an information briefing. The Committee plans to discuss this matter and a proposed report to the Commission during the December 6-8, 1990 ACRS meeting.

- <u>Biological Effects of Ionizing Radiation</u>

  Dr. Arthur C. Upton, Chairman of the National Research Council's Committee on the Biological Effects of Ionizing Radiation (BEIR) briefed the Committee regarding the findings and recommendations related to the health effects of low-level radiation exposures included in the BEIR V report.

The BEIR V report addresses the health effects of exposure of human populations to low-level radiation. In addition, it addresses the delayed health effects that are induced by low linear energy transfer radiations such as x-rays and gamma radiation and, where possible, makes quantitative risk estimates based on statistical analyses of the results of human epidemiological studies and laboratory animal experiments.

- Meeting with the Commissioners

  The Committee members met with the Commissioners on November 8, 1990 and discussed the following issues:

  - Essentially Complete Design
  - Decoupling Siting and Source Term
  - Resolution of Generic Safety Issue B-56, "Diesel Generator Reliability"
  - Containment Design Criteria for Future Plants
  - Systematic Assessment of License Performance

  During the discussion of the issue related to essentially complete design, the Committee members committed to provide a report to the Commission commenting on the recommendations proposed by the staff in SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52." The Commission asked that the Committee comment on the following:

  - What information in an application for design certification should be codified in a manner that cannot be changed without an amendment or exemption?
  - What process should be used for changing the design below that level of detail, keeping in mind the objective of encouraging standardization?

  The ACRS members requested that they be informed of the Commission's resolution of issues in cases where the staff advises the Commission of a disagreement between it and the ACRS. The Commission agreed to indicate its position on the disagreement through its normal decision making process, such as Staff Requirements Memoranda or letters to the ACRS.

- Appointment of ACRS Members

  The Committee approved a press release which states that the Commission plans to consider qualified candidates to fill an existing vacancy on the Committee. This vacancy was created by the recent resignation by Mr. Minnick due to health-related problems.

- ACRS Meeting Dates for Calendar Year 1991

  The Committee approved the following meeting dates for Calendar Year 1991:

369th Meeting	January 10-12, 1991
370th Meeting	February 7-9, 1991
371st Meeting	March 7-9, 1991
372nd Meeting	April 11-13, 1991
373rd Meeting	May 9-11, 1991
374th Meeting	June 6-8, 1991
375th Meeting	July 11-13, 1991
376th Meeting	August 8-10, 1991
377th Meeting	September 5-7, 1991
378th Meeting	October 10-12, 1991
379th Meeting	November 7-9, 1991
380th Meeting	December 12-14, 1991

- Meeting with the General Services Administration (GSA)

  Representatives of the ACRS and OGC met with members of GSA on November 7, 1990 to seek clarifications from GSA regarding the applicability of the Federal Advisory Committee Act (FACA) requirements to ACRS Subcommittee/Subgroup activities.

SUBCOMMITTEE MEETINGS

Since the last summary report of ACRS activities, the following Subcommittee meetings have been held:

- Advanced Pressurized Water Reactors, November 1, 1990

  The Subcommittee discussed the licensing review basis document for the CE System 80+ design.

- Plant Operations, November 1, 1990

  The Subcommittee discussed the efforts by the NRC staff and the Nuclear Management and Resources Council concerning reconstitution of design basis documentation for nucelar power plants.

FUTURE ACTIVITIES

The Committee agreed to the following tentative schedule for the 368th, December 6-8, 1990, ACRS meeting:

- Reactor Operating Experience (Open) - Briefing by representatives of the NRC staff regarding experience gained from reactor operations including problems with the operability of safety systems resulting from egress of noncondensible gas, a loss of AC power event at the Brunswick plant, and a malfunction of the feedwater regulating systems and subsequent failure of the RCIC at the Pilgrim plant.
- Containment Design Criteria (Open) - Discussion of proposed ACRS report to the NRC on containment design criteria for future nuclear plants.
- High-Level Radioactive Waste Disposal (Open) - Briefing by a representative of the Board on Radioactive Waste Management of the National Research Council regarding the National Academy of Sciences/National Research Council report on "Rethinking High-Level Waste Disposal."
- Safety Research (Open) - Briefing by and discussion with representatives of the NRC staff regarding research related to the development of a scaling methodology for direct containment heating phenomena.
- Certification of Standard Designs - Level of Design Detail (Open) - Discussion with NRC staff representatives regarding proposed requirements for the level of design detail required for certification of standardized plant designs. Representatives of the nuclear industry will participate, as appropriate.
- Full Term Operating Licenses for the Palisades Nuclear Plant and the Dresden Unit 2 Nuclear Station (Open/Closed) - Review of proposed conversion of Provisional Operating Licenses to Full Term Operating Licenses for these plants. Representatives of the NRC staff and the licensees will participate, as appropriate.
- Standard Technical Specifications (Open) - Briefing by representatives of the NRC staff regarding the status of the program to develop new standard technical specifications for nuclear power plants.
- Certification Requirements for APWRs (Open) - Discuss proposed ACRS report on additional certification requirements for evolutionary light-water reactors and their relationship to current regulatory requirements.
- Westinghouse Standard Plant SP/90 (Open) - Discuss a proposed report on the proposed preliminary design approval for the Westinghouse standard plant SP/90. Representatives of the NRC staff and the Westinghouse Electric Corporation will participate, as appropriate.
- ACRS Subcommittee Activities (Open) - Hear and discuss reports of assigned ACRS subcommittee activities, as appropriate.

- ACRS Management/Administration (Open/Closed) - Members will discuss anticipated subcommittee activities and items proposed for consideration by the full Committee, qualifications of candidates proposed for appointment to the Committee, election of officers for CY 1991, and administrative matters, as appropriate.
- Miscellaneous (Open) - The Committee will discuss matters which were not completed at previous meetings as time and availability of information permit.
- Reactor Safety Research (Open) - Discuss the scope of the ACRS annul report to the Congress on the NRC Safety Research Program and budget.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

UNITED STATES
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WASHINGTON, D.C. 20555

November 14, 1990

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

Dear Chairman Carr:

SUBJECT: SECY-90-353, LICENSING REVIEW BASIS DOCUMENT FOR THE COMBUSTION ENGINEERING, INC. SYSTEM 80+ EVOLUTIONARY LIGHT WATER REACTOR

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we reviewed the staff's SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990. Our Subcommittee on Advanced Pressurized Water Reactors also considered this matter during a subcommittee meeting on November 1, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of Asea Brown Boveri Combustion Engineering. We also had the benefit of the documents referenced.

The staff has recommended that the Licensing Review Basis (LRB) effort for the Combustion Engineering (CE) System 80+ design, which is well advanced, be continued to completion. There does not appear to be any substantive disagreement between the staff and CE on issues addressed in the LRB document.

The only approved LRB document was proposed by the General Electric Company (GE) as a way of obtaining early agreement with the staff on major process and technical issues for the review of its advanced boiling water reactor design certification application. It was approved by the Director of NRR in a letter to Mr. R. Artigas, GE, on August 7, 1987. This letter contains the qualification that the LRB represented the approach in "certain key areas" that GE was committed to follow ". . . until final Commission positions and staff requirements are defined and implemented." At that time, neither 10 CFR Part 52 nor Commission-approved staff positions relating to the certification of advanced light water reactors such as SECY-90-016 (referenced) were available. We note that 10 CFR Part 52 does not discuss the use of LRB documents as a part of the final design approval or certification process. These regulatory requirements and others under development have preempted the need for and diminished the usefulness of an LRB document for the CE System 80+ design. We recommend that no further effort be devoted to the proposed LRB document for the CE System 80+ design.

Additional comments by ACRS members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr., are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr.

We understand that this LRB document can be completed and issued with relatively little additional effort. If so, we would prefer to see an orderly disposition of this LRB document in accordance with the staff recommendation in SECY-90-362 (referenced). We would agree with our colleagues that the CE System 80+ LRB effort be terminated now if the Commission, the staff, and the ACRS need to invest any significant additional effort.

References:

1. SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990.
2. SECY-90-362, "Staff Comments on the Continuing Need for a License Review Basis Document for Each Passive Design," dated October 24, 1990.
3. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," dated January 12, 1990.
4. Letter LD-90-005 dated January 22, 1990 from A. E. Scherer, Combustion Engineering, to R. Singh, Subject: System 80+ Licensing Review Basis Document.
5. Letter LD-90-060 dated August 28, 1990, from E. H. Kennedy, Combustion Engineering, to Thomas V. Wambach, NRC, Subject: Licensing Review Basis for the System 80+ Standard Design.

UNITED STATES
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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. INTRODUCTION

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. GENERAL COMMENTS

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 presentation 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

Carlyle 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.