POLICY ISSUE
(Information)

February 4, 1991 SECY-91-025

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: FINAL VERSION OF NUREG-1150, VOLUME 3

Purpose: To transmit the final version of Volume 3 of NUREG-1150, containing the summary of comments received, and staff responses, on the first and second draft versions of the report.

Background: The draft final version of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," was transmitted to the Commission on November 7, 1990 in SECY-90-375. This version of the report was an update of the second draft version published in June 1989 and reflected the comments received from the Commission's peer review as well as the American Nuclear Society's review. In the transmittal paper, the staff noted that the Advisory Committee on Reactor Safeguards had completed its review, and its letter with comments was pending. The transmittal paper requested that the Commission approve the report for publication.

On November 15, 1990, the Advisory Committee on Reactor Safeguards transmitted to the Commission its comments and recommendations on the second draft version of NUREG-1150.

By a Staff Requirements Memorandum (SRM) dated December 7, 1990, the Commission approved publication of the report, subject to the completion of four items. Three of the items provided clarifications to text contained in the summary report. The substance of the fourth item was the modification of the report's Appendix E, which summarized comments received from the two review committees and staff responses, to also summarize and provide responses to the comments of the ACRS. This SRM item also indicated that the technical portions of the report (the summary report and Appendices A, B, and C) should be published as Volumes 1 and 2, once the three clarifications were made and the ACRS comments and their discussion were acknowledged. A new volume was to be created which contained Appendix D and the revised Appendix E. This volume was to be transmitted to the Commission for information, and published as Volume 3.

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

<u>Discussion</u>: The staff completed the required changes to Volumes 1 and 2 of NUREG-1150 and the final versions of these volumes were published in December 1990 and released in early January 1991.

The staff has reviewed the ACRS comments, completed the required modifications to Appendix E, and submitted it for publication. The final version is enclosed for the Commission's information. As may be seen, the staff has no significant disagreements with the ACRS comments.

In parallel, the final version of Appendix E is being provided to the ACRS for information.

James M. Taylor
Executive Director
for Operations

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APPENDIX E

RESPONSE TO COMMENTS ON SECOND DRAFT OF NUREG-1150

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TABLES

E.1 Introduction

In June 1989, the NRC published NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," as a second draft for peer review (Ref. E.1). At that time, the NRC also formed a peer review committee under the provisions of the Federal Advisory Committee Act to review the second draft report and answer certain questions with respect to its adequacy. This committee was chaired by Dr. Herbert J.C. Kouts; its entire membership is shown in Table E.1. In parallel, the American Nuclear Society (ANS) continued its review of the report, using a special committee of ANS members. This committee was chaired by Dr. Leo G. LeSage; its entire membership is shown in Table E.2. The comments of both committees were provided to the staff in the summer of 1990 (Refs. E.2 and E.3). This appendix summarizes the comments of the NRC-established committee (the "Kouts Committee") and the ANS committee. Summary staff responses are provided for each specific comment.

The second draft of NUREG-1150 has also been the subject of review by the NRC's Advisory Committee on Reactor Safeguards (ACRS). Its technical review was completed in October 1990; a letter providing its comments was submitted on November 15, 1990. This letter is provided as an attachment to this appendix.

Public comment was also requested on the second draft of NUREG-1150. Four comment letters were received (Refs. E.4 through E.7). These comments have also been assessed and, where appropriate, changes made in the final version of NUREG-1150.

Before discussing the comments provided by the committees on particular topics, it is worth describing the overall conclusions and findings expressed in their reports.

The overall conclusions of the Kouts committee were:

- "NUREG-1150 is a good report, and it represents a great deal of detailed high-quality work. It is commendable that an endeavor was made to consult a wider range of competence apart from that possessed by those directly engaged in producing NUREG-1150. The benefit of constructive openness to criticism is felt in the revised draft."
- "NUREG-1150 draws upon a decade and a half of practice of PSA [probabilistic safety analysis] beyond WASH-1400, mainly in the United States but also in other countries. In most respects, it represents the state of the art of this kind of analysis. It is a step forward from WASH-1400."
- "The data drawn on include many years of experience in plant operation, and a similar period of theoretical and experimental research into severe accident methodology."
- "The disciplined use of expert opinion elicitation was an important advance over previous methods of using expert opinion. It is noted that the prime motive of this technique was to assess the uncertainty in the results of the PSA."
- "The results were derived in great detail, and they are presented by methods which show well their probabilistic spread."
- "NUREG-1150 should be a valuable source of data and methodology to guide future PSAs for individual plants. Like its predecessor, WASH-1400, it should help to show the path for future PSA developments for some time to come." (Kouts 7.2)

The overall findings of the ANS committee review were:

- "NUREG-1150 is a major achievement."
- "The revised draft reports essentially a new study."
- "The revised draft provides a balanced presentation of the central tendencies and uncertainties in risk."

Table E.1 Membership of Special Committee to Review the Severe Accident Risks Report.

Herbert J.C. Kouts	Committee Chairman, Defense Nuclear Facility Safety Board
George Apostolakis	University of California, Los Angeles
E.H. Adolf Birkhofer	Gesellschaft fur Reaktorsicherhe rschungsgelande, Federal Republic of Germany
Lars G. Hoegberg	Swedish Nuclear Power Inspectorate
William G. Kastenberg	University of California, Los Angeles
Leo G. LeSage	Argonne National Laboratory
Norman C. Rasmussen	Massachusetts Institute of Technology
Harry J. Teague	Safety and Reliability Directorate, United Kingdom Atomic Energy Authority
John J. Taylor	Electric Power Research Institute

Table E.2 Membership of American Nuclear Society Special Committee on NUREG-1150.

Leo G. LeSage	Committee Chairman, Argonne National Laboratory
Edward A. Warman	Committee Vice Chairman, Stone & Webster Engineering Corporation
Richard C. Anoba	Carolina Power and Light Company*
Ronald K. Bayer	Virginia Power Company**
R. Allan Brown	Ontario Hydro, Canada
James C. Carter, III	Tenera Risk Management
J. Peter Hosemann	Paul Scherrer Institute, Switzerland
W. Reed Johnson	University of Virginia
Walter B. Loewenstein	Electric Power Research Institute***
Nicholas Tsoulfanidis	University of Missouri
Willem F. Vinck	Associated Consultant, Belgium****

* Currently with Science Applications International Corporation.
** Member in 1987 and 1988.
*** Member until June 1989.
**** Corresponding member.

- "The use of expert opinion in the revised study was greatly improved."
- "NUREG-1150 should supplant WASH-1400."
- "The NRC safety goals are shown to be met for all five plants studied."
- "The NUREG-1150 documentation is a useful compendium of current severe accident analysis information and data."
- "The quality of the report is substantially improved."
- "[The report] is adequate for its stated uses."

The general comments of the ACRS were:

- "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 [the Kouts Committee] and found them helpful. We have no serious disagreements with either of these reviews, nor with their findings."

- "The work described in this [second] 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."

The ACRS letter contained two other comments of particular note. These were:

- "It is disappointing that the staff asserts that virtually no general conclusions can be drawn from 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."

- The NUREG-1150 "results should be used only by those who have a thorough understanding of its limitations."

These last comments are discussed in Section E.8 ("Uses of NUREG-1150"). Specific limitations noted in the ACRS letter are discussed throughout this appendix.

The remaining sections of this appendix provide itemizations of comments (including more specific findings of the ANS committee) received from the review committees and the ACRS on the second draft of NUREG-1150 and staff responses. Comments relating to two general areas, scope and documentation, are itemized first (in Sections E.2 and E.3), followed by comments on specific technical areas: use of expert judgment; accident frequency analysis; accident progression analysis; and source term and offsite consequence analysis (in Sections E.4 through E.7). Finally, Section E.8 itemizes comments on the uses of NUREG-1150.

It should be noted that all committees concluded that issuance of the final version of NUREG-1150 should not be delayed for the conduct of further research or analysis. As such, the responses to certain comments indicate that issues requiring significant effort may be the subject of future NRC work rather than included in the final version of NUREG-1150.

E.2 Scope

Chapter 1 of the second draft of NUREG-1150 described the scope of the risk analyses and identified certain limitations of these analyses. The review committees also noted these limitations, as well as others. Some more general comments by the committees with respect to scope included:

- "The second draft of NUREG-1150 addressed many of the shortcomings identified in the first draft and it provided a more comprehensive and incisive view of risk from the existing light water reactors than did WASH-1400." (Kouts 4.1)

- "In general, NUREG-1150 represents state-of-the-art methodology in PSA and associated uncertainty analysis. However, comparison of resulting risk figures between individual plants and with quantitative safety goals must be made with caution, taking into account questions as to the completeness of the analysis and uncertainties in methods and data." (Kouts 4.12) (Such reservations are itemized in the comments below.)

- "Many of the limitations and uncertainties mentioned above [in Section 4.12 of the Kouts report] may be reduced by improved PSA methodology and by improved experimental and empirical data. Such improvements should be made part of the IPE [Individual Plant Examination] program, but not delay it. We note that many such improvements in methods and data have become available since the closure date for the NUREG-1150 analysis." (Kouts 4.12)

The review committees also provided a number of more specific comments. These are itemized below and staff responses provided.

Comment: The list of initiating events was extensive, and, in most respects, state of the art, but it was not complete. Initiating events not considered included:

- Human errors of commission;
- Incidents starting from low power and shutdown conditions;
- Leaks or breaks of PWR steamlines; and
- Sabotage (understandable in view of methodological and other difficulties involved) (Kouts 3.2.1.1; ACRS).

The effects of aging were not included in the analysis (Kouts 4.12, 7.2).

Response:

The staff acknowledges that human errors of commission have not been included. The treatment of such errors has been the subject of considerable research for several years, but had not sufficiently evolved to permit its use when the NUREG-1150 risk analyses were initiated in 1985. The NRC is currently studying ways in which errors of commission can be practically included in future PRAs (Ref. E.8).

The staff acknowledges that accidents initiated during low power and shutdown operations have not been included in the NUREG-1150 analyses. Recent PRA studies and events in the United States and Europe indicate that the core damage frequency from accidents initiated in such plant operational modes may be significant. The NRC has initiated studies of low power and shutdown accident frequencies and risks for two of the NUREG-1150 plants, Surry and Grand Gulf. Interim, scoping results of these studies are expected in mid-1991. In addition, the NRC has initiated a more general review of non-full-power operational modes to identify the need for additional regulatory requirements. This review is scheduled to be completed in 1991.

Sabotage risks have not been included in the NUREG-1150 risk studies. While the effects of sabotage actions may be similar to that of accidents included in the risk studies, the estimation of the frequencies of such actions is highly uncertain and requires a detailed analysis of the spectrum of threats. Because this threat may be highly variable with time, the staff does not consider it meaningful to attempt to include sabotage risks in PRAs.

The potential for PWR steamline breaks to lead to core damage was assessed (using conservative screening analyses) and determined to be of little significance in the NUREG-1150 PWRs. For some break locations, a steamline break can be similar to a loss of power conversion system transient event and thus can be subsumed into that event. For other break locations, steamline breaks can be recovered through any one of several methods (e.g., feed and bleed cooling, or use of crossties of auxiliary feedwater or emergency core cooling injection from a second unit, if such crossties exist). Using such logic, the core damage potential resulting from such events was judged to be of sufficiently low frequency that it could be screened out early in the analysis. It should be noted, however, that steamline breaks could be important in other PWRs with different plant layouts and system redundancy.

Aging effects were not explicitly included in the analyses. Some consideration of such effects occurs indirectly, however, in that the data base of component and other failures includes failures resulting from aging. The NRC has an extensive program to investigate the impact of aging on plant equipment and to develop and test methods for more explicitly including aging effects in PRA. This work is described in Reference E.9.

Chapter 1 of NUREG-1150 has been updated to better reflect these comments on limitations of the risk analyses.

Comment: The Kouts committee had reservations with respect to the completeness of modeling of interdependencies of technical systems, including detailed modeling of auxiliary systems, formally regarded as not safety-related (Kouts 4.12).

Response:

A major portion of the analysis was devoted to accurately modeling the important auxiliary/support systems, such as component cooling water, and normal and emergency service water. Dependency matrices were developed to identify the dependence of each frontline system on such systems. Connections between safety and nonsafety systems, such as connections to electric power buses, were explicitly considered. Failures of the support systems were also explicitly considered as initiating events. Although most of these events could be ruled out (in initial screening analyses) as initiating events because of train separation, the use of alternative systems, and operator recovery actions, failures of some support systems did contribute to the estimated core damage frequencies (e.g., the component cooling water system failure at Zion and some electric power bus failures).

Comment: The Kouts committee had reservations with respect to uncertainties associated with probabilities mainly based on expert judgment, especially where considerable divergence of opinion existed (Kouts 4.12).

Response:

This comment is discussed in Section E.4.

Comment: The Kouts committee had reservations with respect to the impact of "safety culture" and the fact that the potential effects of management quality are not included (Kouts 4.12).

Response:

This comment is discussed in Section E.5.

Comment: Users of the report should be aware of assumptions made in the screening process in which low frequency accident sequences were eliminated from further consideration and that it may not be appropriate to screen out potential sequences in other plants based on the NUREG-1150 studies (ACRS).

Response:

The staff agrees with this comment.

Comment: The frequency of disruptive failure of the reactor pressure vessel was estimated to be between 1E-7 and 1E-6 per reactor year, yet the event was not treated in the analysis. Reviews published in recent years indicate failure probabilities typically in the range of 1E-6 to 1E-9 per reactor year based mainly on probabilistic fracture mechanics considerations. These considerations show a significant influence of plant-specific parameters such as material properties and aging, positioning of welds, and inspection programs. Thus, a somewhat more extensive discussion might have been warranted in NUREG-1150 (Kouts 3.2.1.7).

Response:

A limited screening analysis was performed for NUREG-1150 which indicated that the relative contribution of vessel rupture to core damage frequency would be negligible. For this reason, this issue was not pursued further.

One issue that could have a significant effect on the estimated core damage frequencies of PWRs due to pressure vessel rupture is pressurized thermal shock (PTS). In 1985, the NRC issued new regulations (Ref. E.10), and defined a screening criterion, to limit the potential impact of PTS. Estimates have been made as to when each licensed PWR would reach this screening criterion (Ref. E.11); none of the three PWRs studied in NUREG-1150 is close to reaching this criterion.

Comment: The lack of analysis of external events for three of the plants studied is a deficiency (Kouts 7.2).

The fire analysis in NUREG-1150 was limited to Surry and Peach Bottom. It was generally state of the art but should have been extended to all five plants (Kouts 4.3.3, 7.2).

Response:

The original intent of (what was to become) the NUREG-1150 risk analyses was to provide perspectives on the mid-1980's revisions to source term technology and thus early analyses did not include accidents initiated by external events. In response to comments on the first draft report, the risk analyses of two plants were extended to include external-event analyses. All five plants were not subjected to external-event analyses because of time and budget constraints. The staff concurs, however, with the basic point made that modern PRAs should include consideration of externally initiated accidents.

Comment: 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 (ACRS).

Response:

As discussed in Chapter 1, the seismic risk calculations are not described in NUREG-1150 because of certain issues relating to the nonradiological consequences of large earthquakes. While some data are provided in NUREG-1150 with respect to containment performance during seismic events, detailed information is provided in supporting contractor reports (Refs. E.12 and E.13).

Comment: The methods and data used [in the fire analysis] were probably the best available at the time the work was performed. However, certain issues identified more recently may result in increased fire risk estimates (ACRS).

Response:

The staff agrees that the more recently identified issues could be significant. The staff is currently investigating these issues further with respect to their importance to plant safety. As the results of these investigations become clear, the staff will reassess the adequacy of current PRA methods and, if appropriate, initiate work to improve the methods.

Comment: It is not clear as to why loss of instrument air was judged not to be important (Kouts 3.2.1.1).

Response:

The loss of instrument air was examined as a potential initiating event. The plants were examined to determine: if the loss of instrument air resulted in a plant trip and the need for decay heat removal; and the effects of loss of instrument air on accident prevention and mitigation systems. For the plants considered, this event was examined and determined to be of minimal importance. Reasons for this conclusion included plant-specific design features such as separation of air supplies, coupled with the availability of backup systems, and/or that loss of instrument air resulted in plant conditions similar to those of other initiating event groups of higher frequencies, such as a transient with the loss of the power conversion system.

Comment: Recognizing and supporting NRC's desire to publish a final NUREG-1150, we recommend that the report indicate the likely impact of Commonwealth Edison Company's committed modifications on the Zion plant results (Kouts 4.2.2).

Response:

The NRC staff has identified the specific modifications that have now been made to the Zion plant (Ref. E.14). Using this information, sensitivity studies have been performed to assess a revised mean core damage frequency and risk for the Zion plant. Chapters 7, 8, and 12 have been revised to indicate the impact of the modifications made at Zion. More detailed documentation of the sensitivity studies performed is provided in Section 15 of Appendix C.

E.3 Documentation and Display of Results

As discussed in Appendix D, the display of results in the first draft of NUREG-1150 was the subject of considerable controversy. Because of the displays used (and other reasons), the first draft was considered inscrutable. In response, the second draft of NUREG-1150 made significant changes to the displays. Some general comments made by the two review committees on this subject included:

- "[With respect to display of results,] the second draft, reviewed by this [Kouts] committee, followed a more conventional course, showing the probability distributions and the major parameters. This choice responds well to the criticisms of both WASH-1400 and the first draft of NUREG-1150, and the present Committee endorses the decision." (Kouts 4.11.1)

- "The current version does a much better job of presenting the results. A particularly helpful form of the results are the matrix-like figures in which mean values of accident progression bins are combined with mean plant damage states and their frequencies. Pie charts are used effectively to display qualitatively the contributions of various initiating events and accident progression scenarios." (ANS 2.a.12.c)

A related question to the choice of display techniques is the appropriateness of citing and using mean values (vs. median values) to describe uncertain parameters. The NRC-sponsored committee addressed this question and noted the following:

- "There has been much discussion over the matter of preference between use of the mean and the median as a point indicator in such cases. Which is the one that most accurately represents the full distribution? We leap forward to the answer: the preference depends on the precise question being asked. In some applications the mean would be preferred; in others it might be the median. There may be instances in which neither would suffice." (Kouts 4.11.3)

Some other general issues related to documentation were also addressed by the committees. These were:

- The (ANS) committee agrees with the decision not to include the radiological consequences of seismic events (ANS 2.a.9.b).

- The ANS committee agrees with the deletion of the analyses of accident prevention and mitigation features (ANS 2.a.10).

- The Kouts committee notes that the staff presentation of the Peach Bottom ATWS sequence demonstrated good traceability of the methods and data used in the analysis, as did the detailed documentation of the Grand Gulf case (Kouts 4.8.4).

The review committees also provided a number of more specific comments. These are itemized below and staff responses provided.

Comment: Experience shows that neglecting sequences with a frequency about two orders of magnitude below the calculated mean core damage frequency does not noticeably change the overall core damage frequency. Thus, for plants that have a mean core damage frequency of 1E-5 per year, a cutoff frequency of 1E-7 per year seems reasonable (Kouts 4.10.2).

It is reasonable to neglect individual risks that are about one order of magnitude or more below the value associated with the U.S. safety goals. A *de minimis* threshold of 1E-7 per year would appropriately represent this reasoning (Kouts 4.10.3).

Taking into account remaining uncertainties in the PRA methodology, e.g., with respect to completeness in the treatment of human factors and external events, estimated core damage frequencies much below 1E-5 per reactor year should be regarded with some caution (Kouts 4.12).

Response:

The staff basically agrees with the frequency cutoff suggested above. In general, accident sequences identified in NUREG-1150 as having frequencies roughly two orders of magnitude or lower below the accident sequence with the highest mean frequency were eliminated.

The staff also basically agrees with the suggestion of neglecting individual risks at levels one order of magnitude or more below the NRC safety goals. In some circumstances, however, values below such levels have been included in NUREG-1150 to permit comparisons with such risk measures as the frequency of a "large release" goal (see Section 13.2).

Chapter 1 has been modified to discuss the cautionary statements on interpretation of PRA results. Throughout the report, figures and tables have also been modified to indicate these cautions.

Comment: The last six chapters of the second draft of NUREG-1150 are the least effective and most difficult to follow portions of the report. Certain of the material is very worthwhile but much of the discussion seems forced, and the observations range from the obvious to those for which the analysis provides no apparent basis (ANS 2.a.12.d).

Response:

These chapters have been reviewed by the staff and its contractors and updated as appropriate. In addition, Appendix C has been expanded to provide additional discussion of issues important to the results and perspectives provided in Chapters 8 through 13.

Comment: Appendix B provides a valuable example of an accident sequence carried through from accident initiation to offsite consequence estimates. However, the example provided did not include early containment failure; hence many of the more interesting issues that are important to risk are not included in the discussion (ANS 5.e.2).

Response:

An example containing early containment failure was originally considered for Appendix B. The early containment failure example was considered interesting but, however, not typical. That is, a more typical sequence was chosen to avoid giving the wrong impression about the importance of early containment failure to risk at Surry. More detailed discussion of specific risk-important issues is provided separately in Appendix C.

Comment: The purpose of Appendix C was to provide some insight to the resolution of key issues. These discussions are sketchy and the information and reasoning that led to the expert judgments generally not provided. There seems to have been no concerted effort to provide a discussion of those issues that were most important to risk (ANS 5.e.1).

Response:

Appendix C has been reviewed and expanded to address other important issues. However, the information provided is still at a somewhat summary level. The reader seeking more detailed information than that in Appendix C should turn to the extensive issue discussions provided in References E.15 and E.16.

Comment: Recovery actions should be discussed in Chapter 2 and their impact quantified in Chapters 3 to 7 (Kouts 7.3).

Response:

Appendix A has been modified to clarify how recovery actions were treated in the risk studies. Important operator actions (including recovery actions) are addressed in qualitative terms in Chapters 3 through 7 along with other types of failures. The large number of events involved makes it impractical to provide discussion in the summary report. However, more detailed information, including sensitivity studies and importance calculations, is provided in Appendix C and in the plant-specific accident frequency analysis reports (Refs. E.17 through E.21).

Comment: To facilitate a comparison between estimates of offsite consequences in WASH-1400 and NUREG-1150, it is suggested that the final version of NUREG-1150 include comparisons of estimated probabilities of exceeding whole-body or thyroid doses as a function of distance from the site. These data are available from calculations already completed, so no delay in issuance of the report should be caused by incorporating such comparisons (Kouts 5.5; ANS 2.b.3, 2.b.10).

Response:

Although the consequence model used in NUREG-1150, MACCS 1.5 (Ref. E.22), can calculate centerline whole-body and thyroid doses as a function of distance from the site, neither of these specific results was generated and saved in the NUREG-1150 analyses. Thus, this information is not now available for generating dose versus distance plots. Because of the time required to develop such information and transform it into a form directly comparable with the Reactor Safety Study (Ref. E.23), it has not been included in the final version of NUREG-1150 but may be appropriate for study and publication in other forums.

Comment: The contributions of the unavailabilities of safety systems to the total core damage frequency should be displayed (Kouts 7.3).

Response:

The calculation and display of system unavailabilities is most appropriately performed on an accident sequence basis and should account for the operability states of support systems (e.g., the unavailability of the auxiliary feedwater system is different if ac power is or is not available). The staff believes that tabulating a single unavailability contribution (e.g., to core damage frequency) could thus be somewhat misleading and has chosen not to include such information in the final version of NUREG-1150. More detailed tabulations of system unavailabilities, accounting for support system availability, etc., could not be generated in a time period consistent with completion of the final report and thus have also not been included.

Comment: Since the supporting documentation upon which NUREG-1150 depends could be helpful to those performing an individual plant examination (IPE), these reports should be published as soon as feasible (ANS 2.b, ACRS).

Response:

Roughly 80 percent of the contractor reports supporting NUREG-1150 (including methods descriptions, computer code descriptions, and documentation of data and results) have now been published. The present staff and contractor schedules indicate publication of all reports by the end of March 1991.

Comment: In the plant-specific chapters, the substantial differences in the methods used for the Zion plant analysis are not highlighted (ANS 2.a.12.b).

Response:

Chapter 7 has been modified to highlight the differences in methods for the Zion accident frequency analysis.

Comment: The final version of NUREG-1150 should clearly state that it should be viewed as a new study and as a replacement for the first draft (ANS 2.b.6).

Response:

Chapter 1 has been modified to clearly state that the final version of NUREG-1150 is so different from the first draft that the latter should no longer be used.

Comment: The first draft of NUREG-1150 was better in one respect, in that it provided a schematic drawing of the containment and reactor coolant system in each plant-specific section (ANS 5.a).

Response:

Plant schematic diagrams have been added to each of the plant-specific chapters (i.e., Chapters 3 through 7).

Comment: Some presentations of results are so small, or so little contrast provided, that the results are unreadable (ANS 5.b).

Response:

Presentations of results throughout the report have been reviewed and improved where needed.

E.4 Use of Expert Judgment

The use of expert judgment is another issue that was the subject of considerable controversy during the review of the first draft of NUREG-1150. Serious criticisms of the methods used in the first draft to obtain these judgments led the staff and its contractors to implement more formal and rigorous methods. The committees reviewing the second draft had a number of general comments on the use of expert judgment. These included:

- "The formal methods that NUREG-1150 employed for such elicitation and the extensive debates that have ensued constitute a significant advance in PSA methodology, since they force visibility on the use of 'engineering judgment,' which is abundant, yet often hidden, in safety studies. The critical element of the whole process, e.g., the selection of the experts, is now widely recognized and appreciated." (Kouts 4.7)

- "Expert opinion elicitation is technically less satisfying than the use of detailed, validated analytical procedures, or experimental data. Considering the lack of understanding of some phenomena, the uncertainties in the scenarios, and the state of development of many of the analytical procedures, some form of expert opinion was unavoidable, however." (Kouts 4.4) (The committee then continued with a set of more specific comments, some of which are appropriate for staff response. These are discussed below.)

- "It can be hoped that, in the long term, the accumulation of experience will help to narrow the distributions in many inputs and outputs of risk assessments. This is, however, unlikely for many of the important ones, because the objective of safety is specifically to avoid just those events that would generate the data useful for risk analysis." (Kouts 4.11.2)

- "There is a 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. 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 [second] draft must be regarded as having major uncertainties in both calculated mean values and in estimated uncertainties." (ACRS)

More specific comments by the review committees are itemized below and staff responses provided.

Comment: Formal, professionally structured expert opinion is preferable to the current alternative, according to which the individual PRA analysts make informal judgments that are not always well documented. However, it is not as technically defensible as analysis using detailed, validated codes. The reproducibility of expert opinion results is a concern (Kouts 4.4).

Response:

The staff agrees that a PRA will be improved by having as many robust calculations as possible. However, it should be noted that it will also never be possible to remove expert judgment from a PRA. A PRA is a procedure for assembling information from many sources, including experimental data, theoretical calculations, and mechanistic code calculations, some of which are conflicting and incomplete. The process of obtaining expert opinions such as used in NUREG-1150 provides a way to review this information and put it in a form that is suitable for use in a PRA. The outcome of this process will always be improved by better information, including calculations by detailed, validated codes. However, some type of expert judgment is always associated with the use of code calculations for several reasons. First, a code calculation is performed for a very specific accident, but the results of this calculation are used in a PRA for groups of "similar" accidents. This type of aggregation requires judgment since the performance of a calculation for every possible accident is not feasible. Second, it is not possible to fully "validate" the mechanistic codes that are used in reactor accident calculations. Thus, there is always a judgment that must be made with respect to the acceptability of a code calculation for a specific application. Third, judgments with respect to model formulation and model parameters must be made to use a code. Thus, the opinion of this "expert" will always enter into the calculations and results.

In the NUREG-1150 uses of expert judgments, two factors acted to reduce the potential impact of this concern: the information being obtained from experts was in the form of probability distributions rather than single or best estimates; and, for key issues, a diversity of judgments was sought. Nonetheless, the staff agrees that the reproducibility of expert judgments can be of concern and expects to support research in this area in the future.

Comment: There is always a question as to "who is an expert on a given issue." The membership of expert panels for the second draft of NUREG-1150 seemed to be better than for the first draft. Yet it still seemed to be unbalanced in that panels still contained more analysts and fewer persons with practical engineering experience who might have expertise on the phenomena; the panels included more users and fewer generators of data than is preferable (Kouts 4.4, 7.2; ACRS).

Response:

The method used to select the members of the expert panels for the NUREG-1150 risk analyses is discussed in Reference E.24. As described there, one goal was to select experts with a diversity of backgrounds. However, experts familiar with reactor safety were usually selected for practical purposes. That is, the project schedule did not permit the time, in general, to educate experts in very specialized areas in the more general area of reactor safety. Two experts on specific phenomena with no familiarity with reactor safety analysis were selected: one on the source term panel and one on the containment loadings panel. One of the experts felt uncomfortable extrapolating his knowledge to reactor accident sequences and declined to continue participation. The second expert went through the effort to educate himself on reactor risk and provided valuable input.

Comment: Expert opinion may have been relied upon too heavily in some instances. An important example is the treatment of core cooling after containment failure. In this case, expert opinion was used to argue that equipment would fail 70-80 percent of the time if environmental temperatures exceeded equipment qualification limits. No explicit analysis was performed to determine the impact of local environmental conditions on equipment heatup and the potential for subsequent failure. It may have been thought that the analysis would have been too time-consuming. It would have been appropriate if possible to have developed these analyses and then to have subjected them to critical review to which expert opinion could have been directed (Kouts 4.4).

Response:

The staff and its contractors did obtain additional information and perform extensive analyses to eliminate the need for or support expert judgments and to supplement the information available in the literature. For the specific issue cited, the experts did receive, for example, information on equipment tolerances and lubricant breakdown temperatures. More generally, many calculations were commissioned specifically for the NUREG-1150 study and presented to the expert panels for review. Some examples of code calculations commissioned include those performed with CONTAIN, CORCON, the Source Term Code

Package, MELCOR, and MELFROG. Such calculations were performed for specific issues and are described in Reference E.16.

Comment: There are some subjects for which the expert opinions were either incomplete or were not targeted on the correct issue because definition of the issue evolved subsequent to the elicitation process and resources were lacking to update it. In these cases, the Sandia staff modified the expert opinion in order to treat the redefined issue. Unfortunately, these new calculations were not reviewed with the expert panel and are not reported in the NUREG-1150 main report or other documentation available to the Kouts committee (Kouts 4.4).

Response:

There were issues in which the responses of the experts were used in a slightly different context than was originally intended. There were two reasons for this:

- The experts had different perceptions of the question asked of them; thus, the information was received from the individual experts in different formats. To aggregate these issues, it was necessary to extrapolate and interpolate some of the expert responses.

- The definition of the issue sometimes evolved subsequent to the elicitation process. In some cases, the issue was much more complex than was anticipated at the time of the elicitation; an example is the treatment of multiple containment failure modes during fast pressure rises. In these cases, the information from the expert panels was reformatted or extrapolated in order to aggregate the response.

In all cases, the original elicitation notes for the accident progression issues and the source term issues have been documented (after review by the experts) in Reference E.16. Any manipulations that were performed on the expert elicitation are described in a section that preceded the individual expert issue documentation, entitled "Method of Aggregation." In virtually all cases, the manipulations were discussed with the experts prior to its use to ensure that the information was not misused.

Comment: The study assigned equal weight factors to the opinions of all experts. Other methods that can develop unequal weight factors were not used (Kouts 4.4).

Response:

The staff and its contractors considered a variety of methods of combining expert judgments, including methods using unequal weighting factors. As noted in Appendix A, the method of equal weighting was chosen because this simple method has been found in many studies (e.g., Ref. E.23) to perform the best.

Comment: The ACRS was 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 (ACRS).

Response:

Roughly 30 percent of the experts were funded directly by the NUREG-1150 study. However, the remainder of the experts were supported by two groups: the NRC and the nuclear industry (e.g., EPRI). Overall, approximately 30 percent of the experts were supported directly by the NUREG-1150 study, 45 percent by other NRC projects, and 25 percent by the nuclear industry.

Comment: The expert opinion procedure is complex, time-consuming, and expensive. Therefore, the full scope of the methodology may have very limited future application. It is unlikely that an expert opinion procedure of this magnitude will be repeated for several years, although expert elicitation on single or narrow issues may be practical. However, it should be remembered that throughout the study analysts had to decide how to use technical information of all kinds; this "expert judgment" is necessary in all PRAs (Kouts 4.4; ACRS).

Response:

The staff agrees that the expert judgment methods used in NUREG-1150 may have limited utility in future work because of the time and cost involved. The staff intends to pursue research in this area with the intent of making the formal uses of expert judgments and the performance of quantitative uncertainty analyses more practical.

Comment: The discussion of issue quantification could be substantially improved, with much clearer indication of what probability distributions were developed by the staff and which specific issues were quantified by the expert review panels (Kouts 7.3; ANS 2.a.8.a).

Response:

The staff agrees with this comment. A table indicating what variables were included in the uncertainty study for the Surry plant (Ref. E.12), and how they were quantified (by expert panels, by NUREG-1150 staff, or by user function), is provided as an example in Section C.1 of Appendix C. Similar tables for the other four plants are provided in References E.13 and E.26 through E.28.

E.5 Accident Frequency Analysis

The review committees and the ACRS had a number of general and specific comments on the accident frequency analysis performed in the NUREG-1150 project. These comments are itemized below, beginning with the subject receiving the most comment, human reliability analysis, discussed in Section E.5.1. Section E.5.2 then provides a discussion of comments on external-event analysis, and Section E.5.3 provides a discussion of other comments on the accident frequency analysis.

E.5.1 Human Reliability Analysis

The Kouts committee provided considerable comment on the subject of human reliability analysis (HRA). As a general comment, the committee noted that:

- "Given the current state of the art in HRA, it would be unreasonable to expect NUREG-1150 to resolve all the outstanding issues including use of a universally accepted model." (Kouts 4.8.2)

The ACRS also provided a general comment on this subject:

- "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 these deficiencies."

In addition, a number of specific comments were provided. These are itemized below along with staff responses.

Comment: NUREG-1150 pioneered the explicit treatment of model uncertainties and the use of expert panels to weigh the relative merits of alternative methods of analysis, but such an approach was not been employed for human actions such as errors of commission and complex situations in control rooms such as in the early phases of an BWR ATWS accident (Kouts 4.8.2).

Response:

The staff agrees that the human reliability analysis should have been performed in a manner more consistent with the remainder of the risk analyses.

Human reliability analysis has been the subject of extensive research in the past few years and has led to the development and initial application of techniques to deal with such issues as human errors of commission. NRC continues to perform a substantial amount of research in HRA, as described in Reference E.8. The demonstration and more widespread use of improved HRA methods in PRA is planned to be the subject of future work by NRC.

Comment: It would have been valuable if the theoretical HRAs of the ATWS sequences had been tested against analysis of real events as a basis for an in-depth analysis of uncertainties in HRA. This could be done as part of expert opinion input on the merits of different HRA models. Such an approach to the

ATWS HRA appears more appropriate and consistent with the use of expert panels in the remainder of NUREG-1150 (Kouts 4.8.4, 7.2).

Response:

The validation of human reliability models (by comparisons with actual events, simulator exercises, etc.) is an integral part of the present NRC program in HRA (Ref. E.8). Future NRC PRA work will make use of such models and thus should provide a better assessment of human performance and its importance to risk.

Comment: For NUREG-1150, the argument was advanced that the conservative screening procedures that have been employed and the wide uncertainty ranges that have been assigned to human error rates have the effect of including the results that other models would have generated. However, such an approach goes against the presumed goal of a PRA, namely, the realistic estimation of risks. Furthermore, the use of an error factor does not necessarily cover the possibility that the models systematically overestimate or underestimate the human error rates (Kouts 4.8.2).

Response:

Conservative screening values were used in the initial quantification of human error probabilities. However, for those events that were potentially significant contributors to core damage frequency, more detailed analyses were performed (this approach being designed to expend significant resources only on those events that are most important). Different types of probability distributions, such as maximum entropy or lognormal, were assigned as appropriate. It is possible that the mean values produced in the analyses could be systematically high or low because of various types of systematic errors. However, the uncertainty analysis did account for these errors in the sense that many of the human error uncertainty distributions were correlated. That is, when a value near the high end of the distribution was chosen for one variable, then a value near the high end of the distribution was chosen for all similar human errors. Thus, the variability did account somewhat for systematic errors. This approach, coupled with the fact that very wide uncertainty distributions were applied to these variables, leads the staff to believe that the treatment of human error uncertainties was adequate for the types of actions included within the scope of the study, recognizing the state of technology of HRA at the time when the work was performed. As noted above, the NRC is currently funding considerable research in the area of HRA (Ref. E.8).

Comment: Considering the different Grand Gulf and Peach Bottom analyses of operator failure to initiate the standby liquid control system during an ATWS event, it is unclear to what extent the differences in estimated probabilities is due to the different methods employed and to the different groups of analysts that have implemented them. It may be questioned if the relatively simple methods used are the most appropriate for very complex, high-stress situations (Kouts 4.8.4).

Response:

The HRA methods used for the Grand Gulf and Peach Bottom ATWS analyses included a detailed task analysis, using the THERP method (Ref. E.29) for Grand Gulf and the SLIM-MAUD method (Ref. E.30) for Peach Bottom. The staff acknowledges that use of different methods and analysts can have an impact on the results obtained and that the impact on the two plant ATWS studies of these differences cannot be easily estimated.

While the use of different analysts can influence the results, it should be recognized that plant design differences were found to be important in NUREG-1150. With respect to ATWS accident sequences in Grand Gulf and Peach Bottom, several such important design differences exist. For example, the standby liquid control system in Grand Gulf is designed to inject boron via the high-pressure core spray sparger, while in Peach Bottom boron is injected into the bottom of the reactor vessel. This difference leads to differences in timing of ATWS events and the procedures established by the plants (operator actions to lower and raise water levels required at Peach Bottom are not needed in Grand Gulf).

Comment: It is beyond the capabilities of present PRA models to account for the influence of management quality on risk; thus it is understandable that NUREG-1150 does not address these issues. While management quality may not be quantifiable in PRA in the near future, its impact on safety is currently being addressed through other NRC and INPO [Institute of Nuclear Power Operations] work. It is important to bear in mind that management quality is not reflected in the risk information as results and perspectives are used (Kouts 4.9).

Response:

Such influences have not been included in NUREG-1150 (or in any other PRA). The present NRC human factors research program (Ref. E.8) includes the study of organizational and management influences on plant safety, including consideration of how such influences can be accounted for in risk studies such as NUREG-1150. Completion of this research should provide some perspective on the degree to which these influences can be incorporated.

Comment: The inclusion of some recovery actions was state of the art. However, the assumptions behind actual recovery curves are not always clear (Kouts 3.2.1.7).

Response:

The recovery analysis included an evaluation of both the time available for recovery and the probability of the operator correctly performing the task. For some faults, actual historical data exist. For example, data exist for all electrical-type faults (i.e., offsite power and diesel generator faults) and faults associated with the power conversion system. For other type faults, historical data did not exist. This recovery information is documented in Reference E.31. For these situations, an HRA or recovery analysis was performed to determine the probability of failure to recover. These recovery curves and "generic" human behavior curves are obtained directly from use of the THERP method (Ref. E.29).

Comment: Innovative recovery actions not covered by operating or emergency procedures should not be included in the baseline analysis, but should be reserved for potential reductions in risk (Kouts 3.2.1.7).

Response:

For some of the accidents analyzed in NUREG-1150, several hours pass before the onset of core damage. In severe accidents of such time duration, an emergency response team would be involved to support the operating crew. It would, therefore, be unrealistic not to allow any innovative recovery actions, considering that such options would be under active investigation and consideration. For these reasons, and recognizing the goal of performing realistic analyses in PRA, credit for innovative recovery in such accidents was permitted in the NUREG-1150 analyses.

It should be noted that, while permitted, very few innovative actions were ultimately incorporated into the analyses. Although several innovative recovery actions were proposed, some of these were incorporated into plant procedures (by the licensee), while others were found to be unnecessary for further analysis because of the already low estimated frequency of the associated accident sequences or the low probability of success.

Comment: Special attention should be given to further development of human reliability analysis, and to proper calibration of the procedures used for it, to enable comparisons to be made between plants and quantitative safety goals (Kouts 4.12, 7.3).

Response:

As discussed in the responses to a number of the previous comments, the NRC has a significant research program under way in the area of human reliability analysis (Ref. E.8).

E.5.2 External-Event Analysis

Specific Comments

Comment: A simplified approach was taken in NUREG-1150 in defining seismic initiators, which leads to failure from all resulting transients, small or large (Kouts 4.3.2).

Response:

All seismically induced transients were not assumed to result in "failure." It is assumed in the analysis that an earthquake will lead to at least one initiator that will require the plant to shut down (either automatically or as a result of operator action).

The occurrence of any of these initiating events, however, does not necessarily imply that core damage will occur. Given that such an initiating event occurs, the same (event tree and fault tree) process is used to assess the conditional probability of core damage as was performed for the internal-event analyses. System failure probabilities may be higher because of earthquake-induced damage, but they are not assumed to be of unity probability.

Comment: Although plant experience was used to establish fire initiation frequencies, judgmental factors were used to determine whether a fire, once started, would persist and cause damage in spite of fire mitigation systems and actions. It would seem that the same data base that was used for fire initiation could and should have been used to give a more realistic value for fire persistence (ANS 3.f.3.a).

Response:

Credit was taken for fire mitigation systems, both manual and automatic, in each fire scenario where applicable. In the case of manual suppression, the same fire data base used to develop the fire initiation frequencies was also used to develop a probability of suppression in any given time frame. For automatic suppression systems, several other studies were used to determine reliability values, as these could not be determined directly from the fire occurrence data base. The data indicated that, for fires in critical locations, the fire was always eventually suppressed (either automatically or manually) but seldom before damage to critical equipment would be predicted to occur (using the COMPBRN model of fire propagation (Ref. E.32)).

Comment: Research in seismic modeling is warranted with the objective of improving the basic model for prediction of attenuation and ground motion and for developing a consensus of the use of one model or model set based as much as practicable on region-specific spectral shapes. Effort should also be made to improve the basic model to reflect greater source depths and regional variations with the appropriate reflections of substrata waves (Kouts 7.3).

Response:

NRC and others continue to sponsor research to improve the general understanding of seismic hazard, including the areas noted above. Such work is described in Reference E.33.

E.5.3 Other Accident Frequency Comments

In addition to the comments itemized above on human reliability and external-event analyses, the committees had a number of other comments on the NUREG-1150 accident frequency analysis. Some general comments provided by the committees included:

- "[The plant damage state analysis] was more detailed than the corresponding analysis in other recent PSA's. It provided an efficient interface with the detailed and complex accident progression and containment loads analysis, and constitutes an advance in PSA methodology." (Kouts 3.2.1.8)
- "In respect to including the modes of containment failure, and in the level of detail, the [accident sequence event tree] analysis was advanced other than typically seen in Level 1 PSA's performed at the time of the NUREG-1150 analysis." (Kouts 3.2.1.2)
- "Although NUREG-1150 is described as being 'a set of modern PRAs, having the limitations of all such studies,' the level of modeling in the accident frequency analysis is not as detailed in some areas as that found in other current PRAs." (ANS 2.a.8.b)
- "A rigorous analysis would always combine the generic and plant-specific [failure data] information. In fact, this is often done using Bayes' theorem. However, we note that in general the numerical differences between the approximate methods of NUREG-1150 and the rigorous approach are insignificant." (Kouts 3.2.1.6)

More specific comments made by the committees are itemized below and staff responses provided.

Comment: Since the first draft was issued, considerable effort was devoted to making the accident frequency analysis more robust. However, the NRC staff recognizes that the state of the art with respect to common-cause failures and human reliability analysis is imperfect and that further improvements can be made in these crucial areas. These areas have not been treated as top-level issues in the expert elicitation process (Kouts 6.3).

Response:

Common-cause failures and selected human error probabilities were offered to the accident frequency analysis panel as issues. The panel concluded that the approach being taken by the analysis team for common-cause failures was appropriate and that expert judgment would not significantly improve the process. Human errors could not be readily considered as a single issue because each action being considered was unique, requiring a separate analysis. The panel did consider several specific human error issues considered to be particularly important. In addition, sensitivity studies on the importance of human error were performed, as discussed in Appendix C.

Comment: The consideration of operating experience in the so-called subtle interactions, represents a good attempt to ensure completeness of failure modes. The method of treatment of dependent failures was state of the art in most respects. The documentation of common-cause failure analysis is difficult to follow. For example, in some instances references were made to EPRI common-cause methods and data, but it appears that in reality a modified beta-factor method was used, which was itself state of the art. The probability of failure of all station batteries is critical to the final results and therefore necessitates better substantiation. Recovery from common-cause failure was restricted to selected electrical equipment (Kouts 3.2.1.4).

Response:

Common-cause failures are discussed in Appendix C and in Reference E.31. The common-cause analysis used in the NUREG-1150 analyses was based primarily on EPRI methods and data. EPRI generic component beta factors were used in the calculation of the common-cause failure (CCF) rates. The CCF rates were calculated as follows:

$$CCF = Q * \beta_n$$

where

Q = Total failure rate

β_n = Beta factor for n components.

For some components, there was not a generic component beta factor for the number of components modeled. In these cases, the EPRI beta factor was modified. In addition, for some components (e.g., batteries, air-operated valves), there were no EPRI generic component beta factors. For these components, other sources or methods were used to calculate the beta factor.

Common-cause failures of the batteries were analyzed in detail in other studies (Ref. E.34) and were used in the NUREG-1150 analysis. Recovery credit for common-cause failures was included where data existed.

Comment: In the analysis of loss of feedwater initiating events, it was assumed that condensate would also be lost, thereby eliminating a potential source of injection capability. For such an initiating event, the recovery potential may be underestimated because of this assumption (Kouts 3.2.1.1).

Response:

The loss of feedwater (LOFW) was treated on a plant-specific basis. For Grand Gulf, upon examination of LOFW, it was determined that condensate would not be lost. For Peach Bottom, it was assumed that condensate was lost with LOFW; however, credit was given for the recovery of the power conversion system, which included recovery of condensate. For PWRs, loss of condensate was included as one of the contributors to LOFW. However, because the LOFW initiating event was not an important contributor to the estimated core damage frequency, no credit for recovery of condensate was considered necessary nor given.

Comment: In general, it appears that very little plant-specific thermal-hydraulic analysis was conducted. Instead, the analysts relied on the results of generic analyses and made judgments as to the degree of applicability in many scenarios (Kouts 3.2.1.2).

Response:

When necessary, plant-specific thermal-hydraulic calculations were performed (e.g., BWR ATWS sequences, ice condenser containment spray actuation timing, and boiloff calculations). Additional thermal-hydraulic calculations were not deemed necessary because a large library of calculations already existed, including those from NRC research programs, vendor analyses, and other industry programs. In addition, actual plant experience was used. For example, the thermal-hydraulic response to a steam generator tube rupture was based in part upon the data from the North Anna tube rupture incident.

Comment: Some success criteria may be too conservative, e.g., both PORVs are assumed to be required for feed and bleed in PWRs (Kouts 3.2.1.2).

Response:

As much as possible, success criteria were developed to be realistic, as opposed to conservative. For example, low-pressure systems were allowed to lead to success in BWR ATWS sequences, including loss of the standby liquid control system, whereas previous studies might not have considered that possibility. In some cases, the success of a particular system was questionable based on information available in the time frame of the study. In these cases, conservative choices were made. Plant procedures (e.g., those that called for both PORVs to be opened in the case of feed and bleed cooling) were also influential in the decisions made in such cases.

Comment: The Grand Gulf ATWS analysis included the two event tree branches of early and late closure of the main steam isolation valves. In the Peach Bottom ATWS analysis, it was, probably conservatively, assumed that the main steam isolation valves (MSIVs) closed for all scenarios. We have found no justification for this difference based on design data or plant operating experience (Kouts 3.2.1.3).

Response:

A plant-specific analysis of MSIV response during ATWS was performed for both Grand Gulf and Peach Bottom. It was not assumed in the Peach Bottom ATWS analysis that all scenarios resulted in MSIV closure. Based on a detailed analysis, it was concluded that all ATWS sequences (with MSIVs open) would lead to isolation signals to the MSIVs.

Comment: Electrical control and actuation circuits were not included in the common-cause failure analysis (Kouts 3.2.1.4).

Response:

Electrical control and actuation circuit faults were included as part of the component random failure rate. The same applies for the common-cause failures. The faults comprising the common-cause failures for components (i.e., valves, pumps, diesels, etc.) were dominated by electrical control and actuation circuit faults.

Comment: Expert judgments assign large uncertainty to the issue of reactor coolant pump seal failure, which is actually susceptible to experimental determination. It is not readily apparent how the bimodal distribution of NUREG-1150 would be affected by the revised estimates of leakage rates and times for initiation of leakage (Kouts 4.6.2).

More recent information and the development of some new reactor coolant pump seal designs since the NUREG-1150 risk studies were completed would lead to a prediction of risk less than that reported (ACRS).

Response:

The expert judgment process was intended to characterize the current understanding of the issue rather than provide resolution. The information base used by the experts included data from experimental programs by Westinghouse, by NRC, and in France. (Appendix C now includes a section of how this issue was addressed; a detailed description is provided in Ref. E.15.)

It should also be noted that the expert judgment process considered the potential importance of the new Westinghouse seal design (not yet in place in the plants analyzed). The experts concluded that seal failure with the new seals would be very unlikely. This would have two effects on the NUREG-1150 analyses. First, the core damage frequency would be reduced because more recovery time would be available prior to core damage. However, for those accident sequences that continue on to core damage, the core damage may occur with the reactor coolant system at high pressure, leading to high containment building loadings at time of reactor vessel breach.

Comment: It is likely that the performance of relief valves, which must function if the feed and bleed operation is to be successful, is not well represented by the data for valve performance used in the NUREG-1150 calculations (ACRS).

Response:

The staff agrees that there is now operating experience data that suggest that the PORV failure rates are optimistic. However, since failure of the feed and bleed function is assessed to be dominated by human errors (to actuate the system), it does not appear that increased failure rates for the PORVs would significantly affect the likelihood of failure to feed and bleed.

Comment: There is now a significant body of evidence to indicate 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 (ACRS).

Response:

The staff agrees that there is now evidence that motor-operated valve failure rates are, under some conditions, higher than those used in NUREG-1150. The NUREG-1150 analyses have not been reevaluated in detail to assess the potential impact of the newer failure rates. It is the staff's judgment that, while the impact would be noticeable, it would not be dominating.

Comment: Plant-specific information is becoming increasingly important in PRA; such information should be collected and placed on file for future use (Kouts 7.3).

Response:

The NRC has developed a data base for the accident frequency analysis models developed in NUREG-1150 (and for other PRAs as well). This data base can be accessed via two computer codes, SARA and IRRAS (Refs. E.35 and E.36), which permit the manipulation of the data for sensitivity analyses, etc. These codes and the data base have been installed and are seeing use in several locations at NRC (and its contractors).

In 1990, the NRC initiated work to assess the feasibility of developing a similar data base and acquisition/analysis system for the accident progression, source term, and risk analysis models of NUREG-1150. This system would make use of data generated with the detailed NUREG-1150 codes, such as EVNTRE (Ref. E.37) and PRAMIS (Ref. E.38).

Comment: The NUREG-1150 documentation does not allow a reviewer to determine how particular events contributed to the frequency of loss of offsite power and subsequent recovery (Kouts 3.2.1.1).

Response:

As noted in the report, NUREG-1150 provides a summary of the methods and results of the five PRAs performed. More detailed information is contained in the underlying contractor reports (Refs. E.15 through E.21, E.12, E.13, and E.26 through E.28). Even these, however, do not contain some of the raw data used to develop and quantify the risk models. Such data are retained in the project files. Included in these files are the data on specific losses of offsite power and its recovery. These data included all events at U.S. nuclear power plants through 1987. These included plant-centered, grid, and weather-related faults. Particular events that could not occur at a particular site were eliminated from the data base for that plant. Further, the analysis considered the operating history at each plant. Plant-specific recovery curves were then generated based on an aggregate of all loss of offsite power events, as opposed to separate recovery curves for each type of failure event.

E.6 Accident Progression Analysis

The review committees had a number of specific comments on the NUREG-1150 accident progression analysis, the most important of which appears to relate to the level of detail in the analysis, compared with the detailed accident phenomenogical computer codes and with the present level of understanding of accident phenomenology. These specific comments are itemized below and staff responses provided. Comments dealing with the closely related subject of accident source term methods are discussed in Section E.7.

Comment: The level of detail in the accident progression analysis appears to have exceeded the understanding of the phenomena involved. It implied greater insight into the processes assumed to be taking place than was justified (Kouts 3.2.2.1, 7.2; ACRS).

If phenomenological models are not provided and directly used, the dependence of the results of the accident progression analysis on governing physical phenomena is hidden (Kouts 3.2.2.1, 7.2).

The generality of the structure of trees and the flexibility to use different levels of modeling capability and details to answer the questions at branch points make the method very powerful, but concern can arise about the meaningfulness of computed results if little information is available about the issues. The possibility of introducing high-level issues makes the method efficient, but this feature should be used with caution if applied to issues with a weak information basis (Kouts 3.2.2.1, 7.2).

We note that in the back end subjective distributions are given for high-level parameters ("issues") that describe the outcomes of complex physical or chemical processes whose basic uncertainties are at lower levels. Mechanistic computational models that would relate these lower-level parameters to the higher-level issues are not employed (for example, the amount of core debris involved in ex-vessel steam explosion is an issue, and its dependence on such lower-level parameters as heat generation rates and chemical reaction rates is not modeled explicitly). Developing subjective probability distributions for high-level parameters may not always be the best approach since the physics of the underlying processes does not get the attention that would be desirable (Kouts 4.7).

Response:

The comments first question whether the detail exceeds the state of knowledge. The staff does not believe so. The intended use of a study to some extent defines the appropriate level of detail. The level of detail was chosen to pass the appropriate information on to the source term analysis and to allow the variation of parameters in the integrated uncertainty analysis. In order to meet these two objectives, it was necessary to form the probabilistic models with high-level issues. Uncertain responses to the high-level issues resulted in wide uncertainty distributions. The use of wide uncertainty distributions to characterize processes that are not well understood should not imply greater insight into the process than is justified but should highlight the uncertainty of that process.

The information presented in NUREG-1150 provides insight into the importance of the high-level parameters and not the governing physical phenomena (e.g., chemical reaction rates). Also, the accident progression event trees used to model the accident progression are based on these high-level parameters. To evaluate the branch point probabilities, however, the high-level parameters are decomposed to the level of the governing physical phenomena (as documented in Ref. E.16).

Because of the complexity of the accident progression, it would have been computationally impossible to model the accident framework for each accident sequence at the physical process level (heat transfer correlations, oxidation rates, etc.). To obtain the insights necessary on the underlying physical processes, it is necessary to establish what high-level parameters are important and then refer to the copious documentation provided on parameter distribution development in Reference E.16.

The staff agrees with the comment that the user should interpret the results of the study carefully when there is a weak information base associated with high-level issues. The NUREG-1150 approach was to include these issues in our models and apply appropriate uncertainty bounds to the parameter distributions.

Comment: There is inconsistency in the detail of the accident progression analysis. This is in part because the state of knowledge with respect to severe accident phenomenology in BWRs versus PWRs is different, the use of expert elicitation for severe accident issues was not the same for all plants, and there was a large uncertainty in operator behavior with respect to post-core-damage recovery actions (Kouts 6.3).

Response:

The general process for performing the accident progression analysis was consistent between PWRs and BWRs. The BWR accident progression event trees (APETs) tended to be larger and more complex because there are more interactions between the containment and the reactor coolant system in BWRs and because the failure location in BWR containments can have a large impact on release fractions. The quality of the information available for input parameter distribution varied for the different issues because of the different amount of experimental and analytical studies performed, but was not clearly superior for either BWRs or PWRs.

There are some issues that have been studied more extensively for PWRs (in-vessel melt progression, direct containment heating). This may have resulted in some inconsistency in the quality of the response on some issues, but the selection criteria for the expert elicitation issues were applied consistently to all plants analyzed.

Comment: The bin "no vessel breach" has a relatively high conditional probability for all plant damage states of PWRs. Yet, the capability to model the issue of core degradation before vessel breach is rather poor. We are unable at present to judge the validity of the conditional probabilities associated with this accident progression bin (Kouts 4.6.1).

Response:

The staff agrees that there are considerable uncertainties associated with this issue. However, it is felt that the approaches used in this study adequately represent the knowledge base as it pertains to this issue. The approach used in the NUREG-1150 analyses is described in Part 6 of Reference E.16.

Comment: Only one of the three experts whose opinions were elicited provided a distribution function for temperature-induced hot leg failure. The other two made the statements "...if necessary conditions for high temperature were met, the leg would always fail...," and "... if high temperatures lasted long enough hot leg would always fail. For shorter time at high temperature hot leg would sometimes fail..."

Since the crucial point in the analysis is the estimation of the hot leg temperature, we cannot see how the two cited statements were incorporated into the aggregated probability distribution presented in NUREG-1150. Therefore, we are unable to judge the validity of the result (Kouts 4.6.3).

Response:

The three experts that considered the temperature-induced hot leg failure all addressed the estimation of the hot leg temperature in their assessments. Two of the experts' decompositions of the issue established continuous distributions for failure probability, the other decomposition provided a point estimate. Each decomposition of the issue was different, yet all addressed hot leg temperatures.

There were many cases in which the distributions (and the associated rationales) provided by experts on the same issue differed significantly. For example, one expert might have felt that the uncertainty in an issue was primarily stochastic in nature while another expert might have felt that the uncertainty was entirely the result of the lack of understanding of the physical process. The method of aggregating distributions (described in Ref. E.16) accommodated different perceptions of the results. Further information on the specific expert analysis of temperature-induced hot leg failure may be found in Part 1 of Reference E.16.

Comment: The treatment of the pressure rise at vessel breach as a single issue by the expert panel obscured a more complete understanding of how the various components contributed to the reduced probability of early containment failure (Kouts 4.6.4).

Response:

The containment loads expert panel felt that tightly coupled phenomena were responsible for the loads that accompany vessel failure. Furthermore, the experts felt that there were synergistic relationships among the various phenomena. Thus, because a simple relationship that ties together the various phenomena involved did not exist, the expert panel did not believe that these phenomena could be isolated without sacrificing the credibility of the final distribution (i.e., the load experienced by the containment). It was their opinion that artificially breaking apart the loads would not provide a realistic picture of the events that are taking place.

The phenomena that contribute to the loads at vessel breach and the importance of the various phenomena for a given distribution are discussed in Reference E.16. From the descriptions of the experts' rationales, the importance of various events to the loads at vessel breach can be obtained. Discussion of the reasons for which these loads are less important now than in the first draft of NUREG-1150 is provided in Section C.5 of Appendix C.

Comment: We note that the concrete erosion progresses faster and with greater intensity than is estimated in NUREG-1150, with a corresponding influence on hydrogen production. However, we agree with the assessment in NUREG-1150 that the meltthrough per se introduces no important influence on health risk (Kouts 4.6.5).

For reasons explained in the section on basemat meltthrough, we believe that this process (MCCI) [molten core-concrete interactions] is modeled incorrectly, with the consequence that the hydrogen generation rate in the ex-vessel phase of accidents in PWRs would be underestimated (Kouts 4.6.6).

Response:

The hydrogen generation rate during core-concrete interactions was based on calculations with the CORCON computer code, as discussed in Reference E.16. The amount of hydrogen produced in the core-concrete interaction phase is dependent on how much unoxidized metal is available, which in turn is dependent on how much has been oxidized in prior phases. From the CORCON calculations, it appears that most of the unoxidized metals remaining in the debris as core-concrete interactions begin are oxidized rapidly. The staff therefore believes that most release rates predicted by current techniques and considering experimental evidence are bounded by the range of release rates in NUREG-1150, when considering in-vessel hydrogen production rate, the at-vessel-breach hydrogen release rate, and the early core-concrete interaction hydrogen production rate.

Because early containment failures and containment bypass accidents tended to dominate the risk, and because there was already so much hydrogen in the containment at the beginning of core-concrete interaction, the amount of hydrogen produced during core-concrete interaction was not considered to be highly risk significant and thus was not varied in the overall uncertainty analysis. As such, while an estimate of hydrogen production based on CORCON calculations may not agree closely with all estimates from experiments such as those performed in the BETA facility, such differences are not believed to be important to the overall risk estimates.

Comment: A separate accident progression bin should be used for basemat meltthrough because knowledge of the consequences of this form of release, though not important from the standpoint of damage to the public, is useful for other purposes (Kouts 3.2.2.3).

Response:

The accident progression results that are shown in NUREG-1150 are summary accident progression bins, grouped together for presentation purposes only. It is possible to separate the basemat meltthrough event from the long-term containment failure event when presenting results of the detailed accident progression analysis. However, in the source term analysis, the two events were binned together. Thus it is not possible to extract separate source term and consequence results for the basemat meltthrough and late containment failure events without performing new calculations. This was not done in the NUREG-1150 analyses because of the low estimated risk significance of both of these failure mechanisms.

Comment: 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. Some important phenomenological issues (e.g., direct containment heating, Mark I shell meltthrough) were characterized quite differently in the first and second drafts even though there was not a major change in the information base. Further, no consideration was found on the impact of ex-vessel steam explosions on early containment failure. There is little unambiguous guidance here for a licensee performing an IPE (ACRS).

Response:

While the staff believes that significant progress has been made in the understanding of severe accident phenomenology, it also agrees that there is a need for more information on a number of specific issues, such as those highlighted by the ACRS. The staff recognized this in developing the guidance for licensee individual plant examinations (IPEs). Appendix 1 to the IPE generic letter (Ref. E.39) provides guidance on how licensees should deal with this lack of information.

It is correct that a number of phenomenological issues were characterized quite differently in the first and second draft versions of NUREG-1150. This reflects a greater information base on a number of important issues such as direct containment heating. The technical bases used by expert panels to assess such issues are discussed in considerable detail in Reference E.16.

The consideration of ex-vessel steam explosions is discussed in Section C.9 of NUREG-1150. This phenomenon was assessed to be of relatively minor importance in the five plants studied, in part because of the greater impacts of such issues as hydrogen combustion loads, etc. Its most prominent impact was in the Grand Gulf plant; Section 6.3 describes its importance relative to other phenomena.

Comment: The aggregate distribution for the probability of drywell shell meltthrough depends critically on the composition of the expert panel. Since this issue combines severe offsite consequences with very large uncertainties, a better resolution of the issues involved is clearly demanded (Kouts 4.6.7).

Response:

The staff agrees with the comment that a better resolution to the drywell shell meltthrough issue is advisable. This issue has been the subject of continuing research by NRC, as discussed in Reference E.40.

Comment: Large uncertainty contributions associated with some phenomena indicate the need for further research. These include the thermal-hydraulic phenomena associated with reactor coolant system (RCS) depressurization (as an accident management strategy), the ways in which the RCS may fail during high-pressure accident sequences in PWRs, and the assessment of threats to (and means to ensure the integrity of) the containment structure in case of a core meltdown resulting from pressure vessel failure (Kouts 7.3).

Response:

The staff agrees that the wide uncertainty distributions associated with specific phenomena provide one indication of where further research is desirable. Other considerations include the importance of the phenomena in question to risk (some wide uncertainty distributions may be acceptable if the contribution to risk is negligible) and the feasibility that further research will reduce the uncertainty bounds. All of these considerations are included by the staff when identifying and prioritizing future research.

Comment: Containment failure from seismic events was based on broad assumptions rather than structural analyses (Kouts 4.3.2).

Response:

Past work has shown that gross structural failure of a typical reinforced concrete containment due to earthquake motion is highly unlikely (Ref. E.41). Rather, it is the pipe penetrations that are most likely to fail because of the loads put on the penetrations by motion of the pipes passing through the penetrations. The loads most likely to cause penetration failure would arise from large motion or support failures of steam generators (in PWRs) or the reactor vessel (in BWRs). Hence, in the NUREG-1150 seismic analysis, containment failure was based on failure of the penetrations resulting from the failure of supports of the major reactor coolant system components. Since the vessel failure and the large LOCA initiating events included in the seismic analysis are based on support failures, it was assumed that some failure of the containment would occur, given either of these initiating events.

This assumption is based on a review of typical containment penetration configurations and discussion with structural experts and is based on the assumption that support failure would result in piping displacements of 1 to 2 feet, and that this would provide a sufficient load to fail the penetration. There are currently no data on the failure capacity of penetrations, given such loads. Hence, this assumption is based on engineering judgment.

In addition, estimates were needed on the size of the leak, given the failures described above. Again, based on typical penetration configurations, it was judged that the most likely crack size would be approximately 1/2 inch by 18 inches, similar to the small leak definition used in the rest of NUREG-1150. It was then assumed that a small leak would occur with a conditional probability of 0.9 and that a larger leak would occur with a conditional probability of 0.1. This assumption was based on the fact that piping supports inside containment would absorb a significant portion of the displacement-induced load and thus limit the leak size. Again, however, there are no data or calculations to substantiate this assumption.

E.7 Source Terms and Consequences

E.7.1 Source Terms

The Kouts committee had two general comments in the area of source term analysis. These were:

- "The overall strategy for generating the uncertainty values in Level 2, including the use of the XSOR codes, appears reasonable, since the tests that were made indicated that the uncertainties introduced by the codes are small compared to the overall Level 2 uncertainties." (Kouts 4.5)

- "Considerable caution is recommended in the use of the results obtained with the approximate XSOR codes without confirmation by more detailed codes." (Kouts 7.3)

The ANS committee had the following general comment:

- "The source terms reported in NUREG-1150 and the resultant offsite consequences should be considered as approximations, due to the reliance on the simplified mass balance XSOR models used to produce large numbers of source terms." (ANS 3.d.3)

In addition to these general comments, the review committees had a number of more specific comments. These are itemized below and staff responses provided.

Comment: The readers of NUREG-1150 should be aware that, of the thousands of source terms results presented, only a few were obtained using the detailed state-of-the-art calculational methods. The remainder were calculated using the parametric XSOR codes. This was a tradeoff to meet the need to generate many results in order to evaluate the uncertainties. The XSOR codes should be used with caution without confirmation by more detailed calculations (Kouts 7.3; ANS 2.a.8.d).

Response:

The XSOR codes were used for two reasons: (1) to generate source terms for the large number of accident progression bins identified in the accident progression analysis, and (2) to provide a means of incorporating the uncertainty in important analysis parameters into the integrated plant studies. Even if uncertainties were not being incorporated into the plant studies, it would be a very demanding undertaking to perform a mechanistic source term calculation for every accident progression bin. The alternative choices are assigning all accident progression bins to the results of a limited number of mechanistic calculations or attempting to modify the results of these calculations to more appropriately match the conditions associated with individual accident progression bins. The latter approach was chosen for the NUREG-1150 analyses.

The XSOR codes actually consist of three parts: a data base developed in the expert elicitation process, a mapping between the accident progression bins and this data base, and an algorithm for constructing source terms on the basis of individual accident progression bins and their associated data. In developing the data base, an attempt was made to use all available sources of information, including mechanistic code calculations, analytic solutions, and experimental data. Thus, the results of mechanistic calculations, as interpreted in the expert review process, are incorporated into the source terms generated by the XSOR codes.

Calculations were performed in which the SOR codes were benchmarked against a Source Term Code Package calculation for a specific scenario. The SOR code was then used to estimate the source terms for a similar scenario. The results compared favorably to a Source Term Code Package calculation made specifically for the second scenario (Ref. E.42).

Chapter 2 and Appendix A to NUREG-1150 have been modified to clarify the role of the XSOR codes.

Comment: Because of the approximate nature of the XSOR codes, the final version of NUREG-1150 should note the need for more exacting analysis of risk-significant accident sequences. The more detailed analysis should be performed and published in a supplement to NUREG-1150. This analysis should concentrate on best-estimate modeling and should be compared with the source terms in the final version of the report (Kouts 7.3; ANS 2.a.8.d).

Response:

The staff agrees with the comment. The staff intends to investigate the practicality of linking risk analysis calculations more closely to accident analysis codes such as MELCOR (Ref. E.43), potentially reducing the dependence on the XSOR codes. As noted below, the staff intends to initiate more detailed studies of the bypass accident sequences.

Comment: With respect to the containment bypass source term, it would be helpful to cite recent work (by EPRI) to help guide the reader to detailed assessments of some of the most important accidents identified in NUREG-1150. Citing more recent studies should help guide the users of NUREG-1150 to existing analyses that provide detailed assessments of some of the most important accident sequences identified in NUREG-1150 (Kouts 4.2.1).

The source terms for containment bypass accident sequences, including interfacing-system LOCAs and steam generator tube ruptures, were not the subject of detailed analyses and may be characterized as conservative approximations (ANS 2.a.8.d).

Response:

A number of Source Term Code Package computer analyses were performed to estimate the source terms for bypass accidents (Ref. E.44). Model development would be required, however, to more realistically treat certain aspects of such accident sequences as deposition in steam generators in a steam generator tube rupture-initiated core damage accident. The staff intends to perform more detailed studies of bypass sequences in followup work to NUREG-1150 and to compare the results of the new studies with those of NUREG-1150. More recent work by EPRI and others will be reflected in such followup comparisons.

Comment: A time cutoff of 24 hours after the onset of core degradation for the release of radionuclides was used throughout NUREG-1150, although no mention of this fact is contained in the report (ANS 2.a.8.d).

Response:

A time cutoff of 24 hours after the onset of core degradation was only used when considering the issue of late revolatilization from the reactor coolant system. Some of the members of the source term expert panel were concerned that the majority of the releases were going to occur extremely late in the accident (much later than 24 hours after the beginning of core damage). The project staff instructed the panel to consider late releases only up to 24 hours after core degradation. The reason for this was that some operator action to cool the reactor coolant system would be expected by that time (e.g., using external cooling by the containment sprays). The time cutoff was not an issue for the other source term processes that were considered because the majority of radionuclides were released well before 24 hours.

Appendix A has been modified to acknowledge this assumption.

Comment: The source terms and consequences of two classes of accidents, containment bypass and early containment failure, should be reported separately as well as the combined data presently displayed (ANS 2.b.9).

Response:

The plant-specific risk reports (Refs. E.12, E.13, and E.26 through E.28) present exceedance frequency curves for the source terms associated with different types of accidents, including containment bypass and early containment failure (e.g., see Figs. 3.3-4 and 3.3-9 in Ref. E.12). Equivalent information for consequences was not generated. However, the individual plant studies do present detailed information on the contribution of different accident types to risk.

Comment: It is not clear how credit is taken for radionuclide retention in the auxiliary building for PWR containment bypass accidents and the reactor building for BWR containment failures (ANS 5.e.4).

Response:

Two types of bypass accidents are considered in the PWR analyses: steam generator tube ruptures (SGTRs) and interfacing-system LOCAs (Event V). During an SGTR accident, the radionuclides are released directly to the environment; therefore, no radionuclide retention in the auxiliary building is considered. For the Event V accident, two methods for retention of radionuclides in the auxiliary (or safeguards) building are considered: retention associated with the building itself and retention from either water pools or water sprays. (At Surry, retention in the relatively small safeguards building is limited; however, there is the potential that the release will occur under a pool of water. At Sequoyah, the release could be mitigated by the fire spray system in the auxiliary building.)

Radionuclide retention in the Peach Bottom reactor building was considered, but none was considered for the Grand Gulf analysis. That portion of the reactor building surrounding the Grand Gulf containment is a relatively weak structure (compared with possible severe accident loadings), and it was judged to have little retention value. The decontamination factors applied in all these plants were provided by the source term expert panel and are documented in Reference E.16.

Comment: 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 Source Term Code Package, 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 other tasks (ACRS).

Response:

The MELCOR code is intended to be the staff's principal analytical model for the accident progression portions of its risk analyses. It has been used in the NUREG-1150 work (e.g., Ref. E.45) and is now being used to support other staff risk analysis work. The staff's planning for further MELCOR development, etc., is described in Reference E.40. As noted above, the staff also plans to investigate the practicality of more closely linking risk analysis calculations to codes such as MELCOR, reducing dependency on parametric models such as the XSOR codes.

E.7.2 Offsite Consequences

The review committees had a number of specific comments in the area of offsite consequence analysis. These are itemized below and staff responses provided.

Comment: The uncertainties in offsite consequences were not included in the NUREG-1150 risk uncertainty estimates (Kouts 7.2; ACRS).

Response:

As indicated in the report, it was not possible because of time constraints to include offsite consequence uncertainties in NUREG-1150. The development of needed probability distributions for parameters included in offsite consequence assessments and the incorporation of these distributions into risk uncertainty assessments is planned to be initiated in 1991.

Comment: There are also a number of uncertainties in the modeling of consequences due to decisions that would be made only during or after a severe accident. These decisions, of a sociopolitical nature, include such things as evacuation, interdiction of land and foodstuff, and the value of real property. These uncertainties have not been included in NUREG-1150, although they have been discussed elsewhere. Recent experience suggests that much lower interdiction levels than those used in NUREG-1150 are sometimes used, which would have the effect on NUREG-1150 results of increasing economic impacts and decreasing health impacts (Kouts 3.2.4, 4.12).

Response:

The staff agrees that issues such as interdiction levels actually used in the event of a reactor accident may be quite different than those used in NUREG-1150. As discussed in Chapter 2 and Appendix A, the evacuation and interdiction assumptions in NUREG-1150 were based on Environmental Protection Agency and Food and Drug Administration guidelines, respectively (Refs. E.46 and E.47). The results of sensitivity studies on these assumptions are provided in Chapters 11 and 12 of the summary report.

Comment: The MACCS code used in NUREG-1150 for offsite consequence analysis is a relatively new code, still under development. It has been neither benchmarked nor validated. Additional uncertainties are introduced by the use of such a new and relatively untested code (ACRS).

Response:

The staff agrees that the use of relatively new computer codes introduces additional uncertainty. Two efforts were undertaken as part of the NUREG-1150 project to improve the reliability of the MACCS code. These were an independent review of the chronic exposure pathway model in the code (Ref. E.48) and an independent line-by-line review of the code (Ref. E.49).

Benchmarking of the MACCS code is now under way under the auspices of an international project sponsored by the Committee on the Safety of Nuclear Installations and the Commission of the European Communities.

Comment: Important information on the offsite consequence calculations is not provided, such as the fact that inhalation doses reflect lifetime dose commitments (ANS 2.a.8.e).

Response:

In its role as a summary document, NUREG-1150 can only give a relatively brief description of the individual models used in the analysis. Detailed descriptions of the individual models are given elsewhere. For the MACCS program used to calculate offsite consequences, detailed descriptions of both the models and the computer program are given in Reference E.22. Further, the data used in the NUREG-1150 consequence calculations are described in Part 7 of Reference E.16.

E.8 Uses of NUREG-1150

The review committees had a number of specific comments in the area of the uses of NUREG-1150. These are itemized below and staff responses provided.

Comment: NUREG-1150, along with other PRAs and recent work in severe accident analysis, should be used to close out as many open issues as can reasonably be achieved and help prioritize limited research resources on the remaining safety issues. A definitive program for the use of NUREG-1150 and its supporting documents should be developed and implemented (Kouts 7.3).

The information presented in NUREG-1150 must be carefully examined in the context of the plant being studied to determine the priority ranking of safety issues, and we caution against broad generalities (ANS 6).

Use of NUREG-1150 to assist in prioritization and resolution of safety issues should be considered a priority application and a principal benefit of the substantial resources expended on this multiyear study (ANS 2.a.13.e).

Response:

As discussed in Chapter 13, the risk analyses of NUREG-1150 are intended to be used as one tool in the prioritization of research and safety issues, as well as in a number of other ways by the staff. Some applications of NUREG-1150 methods and results have already been made, such as in supporting the development of guidance for individual plant examinations (Refs. E.50 and E.51). (Chapter 13 has been updated to reflect some of the more recent uses.) As appropriately noted by the ANS comment, the plant-specific nature of the NUREG-1150 analyses should be and has been kept in mind in such applications.

Following publication of the final version of NUREG-1150, the staff intends to provide additional guidance to potential users of the report within NRC as to its strengths and weaknesses, etc.

Comment: The results of NUREG-1150 should be used only by those who have a thorough understanding of its limitations (ACRS).

Response:

The staff agrees with this comment. As noted in Section E.5.3, the staff has developed a data base and computer codes that permit the staff to modify the NUREG-1150 (and other PRA) accident frequency analyses and plans to develop similar data bases and codes for the remainder of the risk analyses. The staff intends to develop quality assurance procedures as part of this effort to minimize the potential for inappropriate calculations.

Chapter 1 has been modified to note this caution.

Comment: It is disappointing that the staff asserts that virtually no general conclusions can be drawn from a study that took almost 5 years and 17 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 (ACRS).

Response:

The staff agrees that NUREG-1150 provides a substantial body of information, much of which has not yet been "mined" for use in other staff work. It is expected that this body of information will see its principal use by the staff to support the resolution of specific issues, such as study of alternative safety goals, generic issue resolution, PRA reviews, etc. The staff also intends to commit resources to the study of more general issues (e.g., the extrapolation of results for five plants to other plants).

Comment: It is recommended that the NRC issue additional guidance on the treatment of external events in the individual plant examination (IPE) process (Kouts 7.3).

Response:

Such guidance was issued in draft form (for public comment) in July 1990 (Ref. E.51).

Comment: The NUREG-1150 methodology is of special value with respect to guiding risk-reduction and risk-management actions because it makes possible a more sophisticated approach to risk management, addressing not only major contributors to risk, taken as point values, but also contributors associated with large uncertainty bands (Kouts 4.13).

Taken together with the individual plant examinations, NUREG-1150 should help guide evaluation of accident management from a risk-reduction perspective. However, such uses of NUREG-1150 would seem to be limited due to the parametric nature of the study (ANS 6).

Response:

NUREG-1150 information is being used in the development of general accident management guidance (Ref. E.52). As with the individual plant examination process, the NRC in ensuring that each licensee has developed an adequate accident management program. Such a program will be prepared by the licensee reflecting plant-specific information from a plant's individual plant examination as well as from more generic information such as NUREG-1150.

Comment: In many European countries, safety goals and objectives are related to a low risk of releases with disruptive effects on society, typically meaning releases with a potential for long-term restrictions on land usage over large areas. The summary presentations of the results in the main report do not facilitate comparisons with such alternative safety goals. An addition of such comparisons or later documentation might enhance the value of the report, especially outside the United States, since many of these may not be calculable with data in the report (Kouts 4.14).

Response:

The staff agrees that a comparison of the spectrum of national safety goals using the NUREG-1150 plant models would be of considerable interest. Such a comparison could not be accomplished in time for inclusion in NUREG-1150 but is being considered by the staff for future study.

Comment: The limited information presented in NUREG-1150 with respect to the NRC staff's proposed large-release goal would not be particularly useful in the evaluation of implementation strategies (ANS 2.a.13.d).

Response:

The staff agrees that NUREG-1150 provides very limited information on possible large-release goals and implementation strategies. The discussion provided in Chapter 13 of the report was intended as a demonstration of how NUREG-1150 risk models could be used in assessing alternative goals and applying the then-recommended definition of large release, rather than providing a definitive study of a complex technical issue. Since that time, the Commission has provided the staff with additional guidance on safety goal implementation (Ref. E.53) and possible definitions of large releases. It is expected that the NUREG-1150 models will be used by the staff as part of the further consideration of large-release definitions.

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ATTACHMENT TO

APPENDIX E

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. 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 presenta-

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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

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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 :udy 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

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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

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