Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)

Date Published: August 1990

Herbert J. C. Kouts (Committee Chairman), Defense Nuclear Facility Safety Board
George Apostolakis, University of California, Los Angeles
E. H. Adolf Birkhofer, Gesellschaft fur Reaktorsicherheit Forschungsgelande, FRG
Lars G. Hogberg, Swedish Nuclear Power Inspectorate
William E. 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, UKAEA
John J. Taylor, Electric Power Research Institute

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
FOREWORD

In April 1989, the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research published a draft report "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants", NUREG-1150. This report updated, extended and improved upon the information presented in the 1975 "Reactor Safety Study", WASH-1400. Because the information in NUREG-1150 will play a significant role in implementing the NRC's Severe Accident Policy, its quality and credibility are of critical importance. Accordingly, the Commission requested that the Office of Nuclear Regulatory Research (RES) conduct a peer review of NUREG-1150 to ensure that the methods, safety insights and conclusions presented are appropriate and adequately reflect the current state of knowledge with respect to reactor safety.

To this end, the Office of Nuclear Regulatory Research formed a special committee in June of 1989 under the provisions of the Federal Advisory Committee Act. The Committee, composed of a group of recognized national and international experts in nuclear reactor safety, was charged with preparing a report reflecting their review of NUREG-1150 with respect to the adequacy of the methods, data, analysis and conclusions it set forth. In carrying out its work, the Committee held a number of public meetings with NRC staff and contractors to review the details of the methods and data upon which NUREG-1150 was based. The report which follows reflects the results of this peer review.

On behalf of the NRC, I wish to express our appreciation to the members of this Committee who gave of their time and energy, without compensation, in the interests of improving nuclear reactor safety worldwide. Particular thanks must go to Dr. Herbert Kouts, Chairman of the Committee whose leadership helped bring this report to completion.

Eric S. Beckjord, Director 
Office of Nuclear Regulatory Research
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1. BACKGROUND

1.1 The WASH-1400 Report

As one of the last acts before its replacement by the Nuclear Regulatory Commission and the Energy Research and Development Administration, the Atomic Energy Commission published the report WASH-1400, entitled, "The Reactor Safety Study," (RSS), which is often called the Rasmussen Report after the director of the project that produced it. WASH-1400 was the first complete analysis of the risk of nuclear power plants, for it provided calculated values of both the probabilities of severe nuclear accidents and their consequences.

Before this it was believed that the probability of a severe accident to a nuclear plant was very small, with an occurrence expected no more often than about once every million operating years, although the consequences might be extreme, leading to widespread loss of life in nearby areas. The conclusions developed in WASH-1400 were quite different. The probability of an accident causing severe damage to the reactor core was now calculated to be much higher, but the consequences in terms of public injury or death were estimated to be much smaller.

As a basis for comparing hazards between different nuclear plants, and between nuclear plants and other hazards to mankind, there was defined a quantity termed "risk", which corresponded exactly to the concept devised by Pascal in his classic study which underlies probability theory. The risk was defined as the probability of an accident times its consequences. For this reason, such an analysis was called a probabilistic risk assessment (PRA).

WASH-1400 also led to new insights concerning the vulnerabilities of the reactor systems that were analyzed. It was found that the possibility of a severe accident started by rupture of the largest coolant pipe was not the major source of risk from the reactors. Rather, the possibility of rupture of a smaller pipe could contribute more to risk. It was also found that other events associated with such transient conditions as the loss of load were among the more important potential accident initiators. One of the most important insights was that the pressurized water reactor analyzed was subject to the possibility of an accident termed the "interfacing systems LOCA (loss of coolant accident)". This would be initiated by the failure of the check valves separating the high-pressure primary coolant system and the low-pressure emergency core cooling system. The result could be serious damage to the reactor core, with fission products released directly to the environment without intervening protection by the reactor containment building and without the possibility of restoring isolation. This last finding indicated strongly that the new technique would have high value in uncovering fundamental vulnerabilities of specific nuclear plants, with some hope of estimating the reduction in risk that could be achieved by eliminating the vulnerabilities, thereby increasing the safety of the plants. Because this kind of application of the methods is the most important one, we shall term an analysis of the type developed in WASH-1400 a probabilistic safety assessment (PSA), which is closer to the terminology used internationally.

WASH-1400 had an executive summary that presented conclusions on the safety of the plants that were analyzed. The implications of the estimated risk were
extrapolated to further conclusions on the relative safety of the entire nuclear industry. The level of safety was compared to that of other industries, and to safety as seen against the historic background of the effects of such natural phenomena as floods, hurricanes, and even meteorites impinging on the earth.

1.2 The Risk Assessment Review Group

Though the summarized conclusions were not refuted, critics pointed out that they were comments on the report and they did not really constitute an executive summary of WASH-1400 and the results it presented. The concept of evaluating risk from a theoretical analysis of the various ways by which things could go wrong was quite novel, and it was not widely accepted at the outset. Although the staff of the Nuclear Regulatory Commission had begun to use the new method of risk assessment in special applications, skepticism prevented widespread reliance on it, or its conclusions.

For these and related reasons, the Commission convened a special Risk Assessment Review Group in 1977, to advise on the validity of the method and its uses. This Group gave a qualified endorsement to WASH-1400. The Executive Summary was found to be deficient. The methodology was considered to be fundamentally sound, though some of the analysis was regarded as questionable. It was implied that the methodology would be found to have an increasingly important role in the nuclear regulatory program in the future. The Group concluded that the true risk to be attached to operation of the nuclear plants analyzed in WASH-1400 might be larger or smaller than in the "bottom line" estimates presented, but that the estimated uncertainty in these values was probably too small.

The Commission reacted strongly to the report of the Risk Assessment Review Group. A press release was issued that, in effect, rejected WASH-1400 and its conclusions. The staff of the NRC was directed to avoid using it or its methodology in regulatory applications.

1.3 Probabilistic Safety Assessment in Europe

The reaction to WASH-1400 in Europe was very different from that in the United States. There was immediately much greater acceptance of the risk assessment methods, partly because they were recognized as an extension to methods developed and used for some time in the United Kingdom by Reginald Farmer and his associates. A risk analysis then was performed in the Federal Republic of Germany for reactors in that country along lines parallel to those of WASH-1400; it arrived at comparable results. Risk studies were instituted in Sweden and other countries.

Furthermore, the results of these risk studies led to more positive action in Europe than in the United States. The designs of Swedish and German nuclear plants were changed to respond to conclusions on ways to reduce the calculated values of risk.

In the United Kingdom, PSA was applied to non-nuclear questions with important benefits. The best known of these was a study of the safety of industries on Canvey Island, in the Thames estuary, which led to improvement in safety practices there.
1.4 The Three Mile Island Accident

The accident that destroyed the core of the Three Mile Island No. 2 Nuclear Plant reversed the Nuclear Regulatory Commission's policy in the United States. The Kemeny Commission, that conducted the subsequent review for the President, pointed out that the accident was of the most probable type, according to WASH-1400's analysis of the PWR resembling the damaged reactor. In effect, the accident was a confirmation of what might be called a WASH-1400 prediction. Starting from this time, the activities of the Commission began to depend more and more on perceptions of risk as revealed by probabilistic safety analyses.

In the years that followed, the use of the risk assessment methodology has grown both in the United States and Europe. The methods find a steady application in such ways as laying the basis for generic regulatory decisions. The number of PSA's that have been done on nuclear plants has grown. At present, the Nuclear Regulatory Commission requires all nuclear plant licensees to conduct some level of PSA on their plants, as a means of ascertaining whether there are outstanding weaknesses in design that should be considered for remediation.

1.5 Effects of More Recent PSA Work

It is now widely accepted that probabilistic safety assessments are valuable for establishing the risk profiles of nuclear plants. More importantly, the application of PSA techniques can identify unrecognized deficiencies in plant design or operation. With this knowledge, nuclear plant licensees and designers have responded more effectively to safety and regulatory concerns, and have made more informed decisions on plant betterment. Examples of such improvements from applications of PSA to date are numerous. We cite four improvements\(^\text{1,2,3}\): increased redundancy in feedwater systems, improved protection of safety equipment from flooding, improved integrity of main coolant pump seals, and remedy of problems arising from subtle interfaces between instrumentation and control between the nuclear steam supply system and the balance of the plant.

PSA models also have been used to identify the most cost-beneficial modification to plants among several proposed. In doing this, they have simultaneously supported the safety analyses and prioritized the modifications. In other instances, they have identified improvements in operating procedures and have improved the bases for technical specifications.

Applications of PSA also benefit day-to-day activities. Examples are improvements in design or operation, management of the process of plant modification, and improved training of staff\(^\text{4}\). PSA is used to enhance the staff's level of knowledge of the plant's systems and their interdependencies. In addition, the PSA is used to identify those accident sequences most deserving attention in the classroom and at the simulator. The insights from plant specific PSA's will no doubt contribute importantly to the development of comprehensive accident management measures, and the associated training program.

For these reasons, the active involvement of the staff of the plant in PSA work, both in performing the PSA and afterwards, is now seen to be of crucial importance if its full benefits are to be gained.
The continued importance of PSA methodology is highlighted by an NRC policy decision that a PSA must be carried out for all new plants. This will ensure that the present enhancements in safety through the PSA will be applied to future plants and it paves the way to further enhancements. Examination of the work to date on new plants already shows such influence in advanced design characteristics, such as systems that remove decay heat at high pressure, greater protection against station blackout, and independence of instrumentation systems used for safety and control. Nuclear plants of up-to-date design have calculated core-damage frequencies of less than $10^{-5}$ per reactor year (/ry), and still systems to prevent damage to the plant from accidents are being improved. Examples are improved provision against containment bypass and more rugged containment structures, all of which were identified and evaluated through PSA.
2. THE PRESENT REVIEW

2.1 Source Term Studies

Within a year after the Three Mile Island accident, several individuals independently observed that the amount of radioactive material that had been released was far less than expected according to WASH-1400. They transmitted their observation to the Nuclear Regulatory Commission. This led to a project to reassess the source term from a severe accident to a nuclear power plant, culminating in issuance in July, 1986 of NUREG-0956, entitled, "A Reassessment of the Technical Bases for Estimating Source Terms." During the reassessment, new insights were generated on the importance of containment and containment failure modes on the source term, and a decision was made to follow the source term study with a complete reassessment of risk attached to several diverse nuclear plants. This study would draw on all that had been learned about risk assessment in the years since WASH-1400 had been issued. This project was undertaken and became the origin of the draft report NUREG-1150, which is the subject of the present review.

2.2 The First Draft of NUREG-1150

The NUREG-1150 project produced a first draft in February, 1987. The draft was extensively reviewed world-wide. There were three formal peer reviews in the United States; the most complete of these was conducted for the NRC by members of a panel chaired by Dr. William Kastenberg (discussed in Chapter 6 of this report). The peer reviews all concluded that there were defects in the methodology that had been used in the WASH-1400 analysis. Therefore, the Office of Nuclear Regulatory Research of the Commission decided that the exercise should be performed anew with certain basic changes in the methodology. This was done; the project was extensively revised, the data base was improved, new analysis was made, and a second draft was produced, which is reviewed here.

The analytical studies for both drafts and the draft preparation involved teams from several laboratories, universities, and consultant firms, with Sandia National Laboratory assigned the central responsibility and supplying the greatest effort.

2.3 The Committee Conducting This Review

In 1989, the Commission formed the present Committee, subject to the Federal Advisory Committee Act, to conduct a peer review of the second draft of NUREG-1150. The membership of the Committee is listed on the Title page of this document. The charter of the Committee is given in the Appendix.

During its reviews, the Committee heard detailed presentations by individuals who had been engaged in preparing NUREG-1150. They presented the methodology and the results, and answered numerous questions raised by the Committee. The cooperation and the responsiveness of the project staff members and the NRC staff members were excellent.

It has not been possible for this Committee to repeat the analyses, to assess the completeness or correctness of results, nor to determine what the
analysts did in all cases with respect to assumptions and judgmental matters. However, the Committee is confident that it has arrived at balanced and supportable opinions on NUREG-1150. These are presented in the remainder of this report.
3. GENERAL COMMENTS ON METHODOLOGY

3.1 General Remarks on PSA

To lay the groundwork for the Committee's remarks, we present some of the features of a PSA, and then describe, in more detail, specific features of the work done for NUREG-1150.

Probabilistic safety assessment of a nuclear plant can be done at three levels. In Level 1, the probability is calculated of severe damage to the core of the reactor, often equated to substantial or complete melting of the core. Different accident scenarios would lead to damage occurring in somewhat different ways and at different times; these are, therefore, related to different plant damage states. The results of a Level 1 analysis are, then, principally the dominant accident sequences and the probabilities of different plant damage states, each of which could arise from more than one accident sequence.

A Level 2 PSA tracks the fission products released from the different sequences or damage states, to determine the quantities, physical and chemical characteristics, and timing of their release from the containment building. These data are collectively called the source term.

A Level 3 PSA continues the calculation through the dispersion of fission products through the available pathways, and calculates the consequences in such terms as damage to human health, land contamination and interdiction, and effects on the food chain.

The analysis through Level 1 is often called the front end of the PSA, while the remainder is called the back end.

The probability of damage to the core and the release from the containment is estimated using "event trees". An event tree begins with a specific system failure or human action called the initiating event, and continues through successive failures or errors that must also occur for the accident or its resulting release to take place. An event tree constitutes a logic chain, with branch points signifying the separate failures or errors. Each branch point is associated with the probability of the contributing branch event. These probabilities may be calculated from historical data or from fault trees, which are means of estimating the failure rates of more complex devices from the failure rates of their components. In some cases, expert judgement is used to develop failure rates at branch points; the use of expert opinion in the NUREG-1150 process is discussed at length in Section 4.4 of this report.

Many of the branch point probabilities are developed as probability distributions. The features and origins of these distributions are discussed in Section 4.7. As a result, the numerical conclusions of a PSA, regardless of level, are in the form of probability distributions, the result of propagation of the branch point distributions and other distributions through the calculation.
3.2 Methods Used in NUREG-1150

It is convenient to state the Committee's comments on the specific topics of NUREG-1150's methodology immediately after these topics are discussed. More detailed comments are reserved for Chapter 4.

3.2.1 Accident Frequency Analysis

3.2.1.1 Initiating Events

In a first step, potentially important accident initiators were identified and their expected frequencies of occurrence were quantified. Generally, initiating events were considered to be potentially important if they led to a need for actuation of safety systems for rendering the plant subcritical or for removing decay heat. The identification of these initiating events and the safety systems required to deal with them were based on plant data, the results of previous PSA's, and review of unusual or unique events that might affect the specific plant. The NUREG-1150 analysis considered only events during normal power generation, and did not include initiators from the shutdown state or startup operations.

The end-product of this step was a grouping of initiating events and their expected frequencies of occurrence. The grouping, which was based on similarity of system response, defined the number and types of event trees to be constructed in the subsequent steps of the analysis.

Comments:

The list of initiating events analyzed by the draft NUREG-1150 was extensive, and, in most respects, state-of-the-art, but it was not complete. As noted elsewhere, human errors of commission were not included, nor were incidents started from low-power or shutdown modes. We note that these are commonly not covered in PSA's. It is not clear as to why loss of instrument air was judged not to be important. For loss of offsite power and its recovery, the documentation does not allow a reviewer to determine how particular events contributed to the choice of the final frequency and probability of recovery, matters found important in analysis of the Millstone salt spray event. In treating loss of main feedwater events, the analysis assumed that condensate would also be lost, thereby eliminating a potential source of injection recovery. For the generic initiating event frequency, the recovery potential may be understated, because events which actually may not lead to total loss of feedwater are presumed to do so.

We note that leaks or breaks in the main steamlines of PWR's were not considered; this may be because relatively small contributions were attributed to this initiator in the PSA's of several other PWR's in the United States. In these, the contributions to the frequency of core damage ranged about a few percent. Since this value borders on being significant, and might be important if improvements are made to plants, reducing the probability of damage in other ways, the topic of main steamline breaks might more properly be addressed.
Finally, we note that damage to the plant and its safety systems through wilful human actions, i.e. sabotage, is not covered in NUREG-1150, nor in other PSA's. This is understandable in view of the methodological and other difficulties involved. However, sabotage must be kept in mind when discussing overall risk.

3.2.1.2 Accident Sequence Event Trees

In this task, event trees were constructed which defined the accident sequences leading to core damage for each of the initiating event groups. The structure of the event trees reflected the interrelationships of systems and also accounted for phenomenological aspects which determine whether the sequences lead to core damage. The structure also included potential effects on core damage to BWR's, through failures of certain containment functions and systems.

Attention was given to various methods of injecting water into the core (e.g., control rod cooling systems, fire water, and service water for the BWR's). In general, very little analysis of plant-specific thermal hydraulics was conducted. Instead, the analysts relied on the results of generic analyses and made judgements as to degree of applicability in many scenarios.

The products of this task were models of all the accident sequences to be quantified in the subsequent step.

Comments:

In respect to including the modes of containment failure, and in the level of detail, the analysis was advanced over that typically seen in level 1 PSA's performed at the time of the NUREG-1150 analysis. The insights on effects of failures of certain features are principally important to BWR's, and have been included in recent PSA's of BWR's.

Some success criteria may be too conservative, e.g., 2 of 2 PORV's required to open for feed and bleed for a PWR.

3.2.1.3 Systems Analysis

The expected frequencies of occurrence of the accident sequence groups were quantified through the success or failure probabilities at the branch points of all required safety functions, depending on the accident sequence. The important contributors to failure of each system were determined by fault tree analysis. Where an accident sequence led to an end point identified as "core damage", the fault trees corresponding to the system functions which fail along the sequence path were merged into one large fault tree. Common-cause failures and dependent and subtle failures resulting from system interdependencies were modeled directly in the fault trees, as were human errors associated with testing and maintenance, and also some recovery actions when they were included in the operating procedures or the emergency procedures. The level of detail to which fault trees were developed depended on the importance of the systems and on the data base available to quantify component failure probabilities. The interrelated tasks "Accident sequence event tree analysis" and "System analysis" were combined in this manner, using the "Small event tree/large fault tree" method.
Comments:

The effort in this task is typical of that of other PSA's. Heavy use was made of other PSA's, both for data and for fault trees.

Only in the case of Grand Gulf did the BWR ATWS event tree include the two branches of early and late closure of the main steam isolation valves. In the Peach Bottom analysis it was, probably conservatively, assumed that the MSIV's closed for all scenarios. We have found no justification for this difference, based on design data or plant operating experience.

3.2.1.4 Dependent and Subtle Failure Analysis

Dependent failures from direct functional dependencies were incorporated explicitly into the fault trees. "Miscellaneous" dependent failures resulting from less direct causes were incorporated into the fault tree analysis using a modified Beta-factor method. Common-cause failures were modeled for mechanical equipment such as redundant pumps, valves, diesel generators, and batteries.

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. However, 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 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, needs better substantiation. Electrical control and actuation circuits were not included in the analysis of common-cause failure.

3.2.1.5 Human Reliability Analysis

This very important topic is discussed in detail in Section 4.8.

3.2.1.6 Data Base on Failures

A generic data base for frequencies of initiating events, component failure rates, and their associated uncertainties was developed. If plant data appeared to differ significantly from generic data, plant-specific data were developed, and included in the data base. Yet plant-specific data were not used if they were based on no failures or one failure observed in a small population.

Comment:

A rigorous analysis would always combine the generic and the plant-specific 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.
3.2.1.7 Accident Sequence Quantification

The information produced in the preceding steps was assembled into estimates of the frequencies of accident sequences. In this process, event sequences were dropped from further consideration if their frequencies were below some value, and if no credit had been given to recovery actions. For the remaining sequences, recovery actions by the plant personnel were taken into account, and included in the analysis if they

- were directly stated in the emergency or abnormal procedures, or
- could be expected to result directly from procedural steps, and
- if sufficient time would be available for diagnosis and completion of the action.

In the latter category, some credit was allowed for "innovative recovery" actions which were not explicitly identified in the plant procedures, but which could be provided by the plant's accident response team in long-term accident sequences. The recovery actions were plant-specific. Event tree and fault tree analysis were used to incorporate them into the accident sequence quantification.

In a second sweep, event sequences were dropped from further consideration if their expected frequency of occurrence with credit for recovery action was below some value, generally $10^{-7}$/ry. Only the remaining sequences were analyzed further. For Surry, this cutoff value was $10^{-9}$/ry for all station blackout sequences.

Comments:

The inclusion of some recovery actions was state-of-the-art in PSA methodology. However, the assumptions behind actual recovery curves are not always clear. For example, in station blackout scenarios at Surry it was assumed (without explanation) that following depletion of the batteries after 4 hours, the plant could survive 3 more hours without any instrumentation and control, and then recovery could take place without core damage. These recovery actions also included some unplanned ones which normally would be included among accident management measures. Furthermore, 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.

We noted an inconsistency for PWR's: the frequency of disruptive failure of the reactor pressure vessel was assumed to be between $10^{-7}$/ry and $10^{-6}$/ry, yet the event was not treated in the analysis. Recent reviews indicate probabilities of rupture typically in the range of $10^{-8}$/ry to $10^{-9}$/ry, based mainly on considerations of probabilistic fracture mechanics which show a significant influence of plant-specific parameters such as material properties and aging, positions of welds, and inspection programs. Thus, a more extensive discussion might have been warranted in NUREG-1150.
3.2.1.8 Plant Damage State Analysis

Plant Damage States were defined to conveniently group the information that must be passed on to the subsequent analysis of accident progression and containment loads. The definitions of plant damage states provided the status of the plant systems at the onset of core damage, that included information on the status of the core cooling systems, containment systems, and support systems. The plant damage states were defined by additional questions at the end of the accident sequence event trees.

Comment:

This step 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.

3.2.1.9 Uncertainty Analysis

Estimations of the uncertainties in the calculations of core-damage frequency were included in the analysis. The uncertainties in this phase (Level 1) resulted from incomplete understanding of initiating events, reactor systems, and operator actions. The uncertainties were generated from a combination of data inputs and statistical scatter in expert opinion.

The important topic of uncertainty analysis is discussed in more detail in Section 4.7.

3.2.1.10 Display of Results of Accident Frequency Analysis

The results for the total core-damage frequency were displayed as

- the subjective probability density function of core-damage frequency
- histograms of Latin hypercube sampling observations, and
- identification of the distribution measures: mean, median, and percentile values.

The definitions of plant damage states and their estimated frequencies were presented in tables. The contributions of accident groups to the total mean frequency of core damage were displayed in piecharts. Several other importance measures were also discussed and the results presented in tables:

- risk reduction potential, which is the amount by which the total core-damage frequency would be reduced if the probability of a specified failure mechanism were zero,

- risk increase potential, which is the amount by which the total core-damage frequency would be increased if the probability of a specified failure mechanism were unity,
... uncertainty/importance, which shows the amount by which the overall uncertainty in the core-damage frequency would be affected by the uncertainty associated with a specified event or phenomenon.

- importance of common-cause failure, which shows the potential effect of eliminating all common-cause failures, and

- importance of human errors, which shows the potential effect of eliminating all human errors.

Comments:

The method of display was a substantial improvement over that used in the first draft of NUREG-1150, and was similar to that in other recent PSA's.

In the spirit of a level 1 PSA, it would have been desirable to show in a separate presentation the contributions of the unavailabilities of safety systems to the total frequency of core damage.

Additional discussion of the method of display of results can be found in Section 4.11.

3.2.2 Accident Progression, Containment Loadings, and Structural Response

3.2.2.1 Development and Quantification of Accident Progression Event Trees

This part of the analysis traced the physical progression of the accident from a plant damage state to quantification of the characteristics and magnitude of a release of radioactive substances. The analysis included the core-damage process inside the reactor vessel and outside the vessel subsequent to breaching of the primary system. The impact of these processes on the containment building structure was analyzed, emphasizing pressure buildup.

All important aspects of accident progression cannot yet be modeled on the basis of validated physical models. Therefore, all possible accident sequences resulting from each plant damage state cannot be described fully and in detail with current analytical tools.

The information used in accident progression analysis consisted of a variety of research results, including both experimental results and numerous computer calculations of specific important aspects of accident progression. Elicitation of expert opinion also played an important role. The results benefitted considerably from observations of damage at TMI. Many new calculations were performed for NUREG-1150, filling the largest gaps in knowledge of accident progression.

The accident progression analysis had four steps:

- Development of accident progression event trees (APET's),

- Probabilistic quantification of event tree issues,
Structural analysis, and
Grouping of event tree outcomes into accident progression bins.

Plant-specific accident progression event trees (APET) were constructed by posing a set of questions on the physical phenomena affecting accident progression.

Many of the questions governing the branching probabilities were related to such high-level issues as "amount of zirconium oxidized in vessel?", "amount of the core released from the vessel at breach?", and "debris bed coolable?". In general, the questions were not answered by calculations based on phenomenological models. Rather, branching probabilities, dependencies of a question on previous questions, and/or tables of values of parameters were assigned directly to the branch-points. Questions relating to operability of equipment, availability of power, and recovery actions were addressed in terms of probability distributions in a way similar to the accident frequency analysis. For some of the key issues, the knowledge base was rather poor, so expert opinion was elicited to generate these branching probabilities or probability distributions.

**Comments:**

The accident progression event tree for each plant consisted of about 100 branches, each having multiple outcomes or branches. It seemed to us that this level of detail exceeded understanding of the phenomena involved, and implied greater insight into the processes assumed to be taking place than was justified. When confronted by the need to quantify poorly understood phenomena, it is certainly necessary to dissect the problem carefully to ensure that important aspects are not overlooked. But this practice should be restricted to assisting the thought process, and the final quantification should be at a scale commensurate with the overall understanding.

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.

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 in cases where 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 when applied to issues with a weak information basis.

### 3.2.2.2 The XSOR Codes

The actual outcome of the accident progression event trees in terms of release of fission products to the environment was found with an approximate, simplified calculational procedure based on the XSOR codes. The process was an essential part of development of distribution functions and uncertainty estimates. The XSOR codes and their use are discussed in Section 4.5 of this report.
3.2.2.3 Grouping of the Outcomes of Accident Progression Event Trees

The process just described generated many alternative outcomes, that were grouped into a relatively small number of "accident progression bins". These bins were characterized by features important for the assessment of the release of radioactive substances from the containment, for example, time, size, and location of containment failure, availability of equipment and processes that remove radioactive substances from the containment atmosphere.

Comment:

Basemat melt-through could also occur, even in the presence of other containment failure modes. Therefore, a separate accident progression bin should be used for basemat melt-through because knowledge of the consequences of this form of release is useful for other purposes, though not necessarily important from the standpoint of risk to the public health and safety.

3.2.3 Elicitation of Expert Opinion

One of the distinctive features of NUREG-1150 was the extensive use of structured, formalized elicitation of expert opinion. In particular, the level 2 and, more generally, the back-end analysis rested heavily on the outcomes of elicitation of expert opinion on a number of crucial issues. The process was used to generate input values and distributions for many of the parameters in the study where reliable models and values were not available, e.g., due to the complexity of the phenomena. The procedure to elicit expert opinion used for the first draft of NUREG-1150 and the results obtained with it were extensively criticized by the peer reviews; the entire process was restructured and elicitation was redone for the second draft. Of the seven panels of experts that were assembled for the latter, only one addressed issues in the Level 1 part of the exercise.

The elicitation of expert opinion was such an important part of the NUREG-1150 methodology that it is discussed at length in Section 4.4 of this report.

3.2.4 Consequence Model

The third and final set of calculations in a PSA (Level 3) is aimed at quantifying the radiological consequences of severe accidents at nuclear power plants. Before NUREG-1150, the major tool for analysis of consequences in almost all risk assessments was the CRAC series of codes which were developed for WASH-1400. NUREG-1150 employed the MELCOR Accident Consequence Code System (MACCS), a relatively new model that is still undergoing development: its operation and results have not been tested by extended use.

Calculations with MACCS (as with CRAC) require extensive data for such things as the source term, weather, population distributions, land usage, economic factors, and health effects. They also require assumptions regarding emergency response (e.g., evacuation and interdiction).

The consequences (e.g., early and latent fatalities, economic loss) are calculated probabilistically. A typical MACCS calculation will sample 100
weather variations and a smaller number of population sectors. The results are displayed as complementary cumulative distribution functions for each source term and accident sequence.

In NUREG-1150, thousands of source terms were generated by the XSOR codes for use in the uncertainty analysis (see Section 4.5). However, it would have been too expensive (in terms of time) to run MACCS for each source term. Therefore, a clustering procedure was used to bin the source terms into a smaller number. For example, in the Peach Bottom risk assessment, 13,895 source terms were grouped into 54 bins.

A single MACCS calculation was performed for each bin, and the results used for the analysis of integrated risk and uncertainty. It is important to note that the uncertainties in the consequence analyses for each sequence were not propagated. The uncertainties shown in the risk profiles for each reactor and each consequence are due to the uncertainty in the Level 1 and Level 2 aspects of the PSA only.

**Comments:**

We realize that NUREG-1150 only estimated the numbers of early and late cancer fatalities and individual mortality risks, and did not estimate land interdiction or economic losses. The following comments are addressed more to the MACCS code itself and its prospective uses, rather than to the narrower issue of thier use in the NUREG-1150 analysis.

A recent study by Helton et al.\(^7\) focused on the sensitivity of the MACCS results to variations of important input parameters and data as well as on possible inaccuracies of the models. The study concluded that, "...the potential effects of consequence modeling uncertainties in the NUREG-1150 analyses or other integrated risk assessments could be large..."

In addition to these types of uncertainties associated with consequence calculations, there are several socio-political decisions that may have a significant impact upon the magnitude of the health and economic consequences, including the decision of when and over what region an evacuation may be ordered. Important effects could also flow from the definition of the "safe to occupy" level of contamination of homes and businesses and the setting of the contamination levels of food and water that require withdrawal from use. Most calculations assume that in the United States these decisions will be based upon the EPA Protective Action Guides (PAG). However, the experience in Europe following the Chernobyl accident strongly suggests this may not be so. After Chernobyl, several countries set acceptable levels of contamination well below values recommended by expert international bodies. This action significantly increased the economic impact of the Chernobyl accident.

For PSA's on U.S. reactors that include Level 3 calculations, the general practice is to base the socio-political levels described above by an interpretation that is consistent with EPA's PAG's, as was done in NUREG-1150. The results of the Level 1 and 2 analysis have produced quite large uncertainties, and so it is not clear whether including this effect would significantly increase the economic risk. However, recent experience in the United States and elsewhere
suggests that much lower levels than those in Protective Action Guides are sometimes set by political decisions and considerations of market acceptance of food products. This almost always results in a substantial increase in costs for a very modest reduction in health effects. In such a case, the NUREG-1150 results for the economic impacts may be biased low, and health impact biased high.
4. DETAILED COMMENTS ON SELECTED ISSUES

4.1 Introduction

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. The second draft has substantially improved documentation over the earlier draft.

4.2 Internal Events

4.2.1 Bypass Sequences

One of the major conclusions from the NUREG-1150 study is that risks for pressurized-water reactors tend to be primarily associated with accident sequences in which the containment is bypassed. They are usually followed in importance by sequences with early containment failure. (Late containment failures are calculated to have very small source terms.) These points are clearly illustrated in graphs, such as Figures 3.13 and 3.14 of NUREG-1150, which depict the major contributors to risk among the various plant damage states and accident progression bins considered. It is instructive to note the dominance of the contribution to risk by containment bypass, despite the fact that these sequences are not heavily represented among those leading to core damage, i.e., the proportion is 8% for Surry, 4% for Sequoyah, and 0.5% for Zion.

Moreover, in the NUREG-1150 study, most PWR core-damage accidents do not result in containment failure, as illustrated in the following tabulation:

<table>
<thead>
<tr>
<th>Mean Conditional Probability of Containment Failure Modes</th>
<th>Surry</th>
<th>Sequoyah</th>
<th>Zion</th>
</tr>
</thead>
<tbody>
<tr>
<td>No Containment Failure</td>
<td>81%</td>
<td>66%</td>
<td>74%</td>
</tr>
<tr>
<td>Late Containment Failure*</td>
<td>6%</td>
<td>21%</td>
<td>24%</td>
</tr>
<tr>
<td>Early Containment Failure*</td>
<td>1%</td>
<td>7%</td>
<td>1%</td>
</tr>
<tr>
<td>Containment Bypass</td>
<td>12%</td>
<td>6%</td>
<td>1%</td>
</tr>
</tbody>
</table>

(*Failure above ground)

In the Surry analysis, bypass sequences dominate risk and are 12 times more likely to result in releases to the environment than are sequences resulting in early containment failure. In the Sequoyah analysis, early containment failure and containment bypass are more nearly equal in probability, but the larger source terms attributed to the bypass sequences result in their being the dominant contributors to risk.

In the case of Zion, accident sequences resulting in early containment failure are more than twice as probable (1.4% contribution) as accidents associated with containment bypass (0.7% contribution). As a result, the risks are dominated by the early containment failure sequences for Zion. When the
Component cooling water modifications summarized in Section 4.2.2 are reflected in the analysis, the estimated probability of early containment failure will be substantially reduced for Zion, resulting in an increase in the relative contribution to risk from bypass sequences.

A recently completed study by the Electric Power Research Institute entitled "Evaluation of Consequences of Containment Bypass Scenarios" (NP-6586L), issued in November 1989, explored the effects of detailed features of the containment on the outcome of the bypass scenarios. It concentrated on containment bypass sequences for PWR's and BWR sequences in which the suppression pool is bypassed. Plant-specific features from 21 nuclear power plants were considered in detailed sensitivity analyses conducted with the Modular Accident Analysis Program (MAAP).

The range of plant-specific features included building size and compartmentalization, location of vertical and horizontal passages, and location of communication paths with the environment. Other influences were the presence and/or operability of equipment (fire sprays and ventilation equipment systems) and geometric considerations that might determine whether fission products would enter the reactor or auxiliary building under water.

The calculations showed that the magnitudes and types of estimated fission product releases to the environment are highly sensitive to the number and location of paths to the environment, to the compartmentalization, to the position of doorjambs, to the flow area to the environment, and to the scrubbing effects of water pools and sprays. This EPRI research, conducted after the completion of NUREG-1150, shows that the potential for mitigating fission product releases can be significant, although the degree of mitigation would be highly plant-specific. The work implies that in the IPE analyses underway, care must be exercised to ensure that the methods used can deal properly with the features affecting the outcome of containment bypass scenarios.

It is recognized that any study has to have a cutoff date for introducing new information and data; NUREG-1150's cutoff date was February 1988. However, this issue could have an important effect on the outcome of some NUREG-1150 calculations, and we address it among the conclusions and recommendations in Chapter 7. Citing more recent studies, such as the EPRI report mentioned above, should help guide the users of NUREG-1150 to existing analyses which provide detailed assessments of some of the most important accident sequences identified in NUREG-1150.

4.2.2 Treatment of Zion Nuclear Plant

The estimated mean core-damage frequency (CDF) for Zion stated in NUREG-1150 is 3.4x10^-4/ry, which is significantly higher than the frequencies estimated for Sequoyah and Surry. A reactor coolant pump LOCA, caused by a loss of cooling water, contributes 85% of this frequency. Commonwealth Edison Company has committed to improve the availability of this cooling water, to install new and improved seal O-rings, and to implement more effective operating procedures. The NUREG-1150 contractor has told us that these improvements, using existing NUREG-1150 methodology, could reduce the CDF to about 5x10^-5/ry, a value comparable to that of the other PWR's studied. We recommend that the final NUREG-1150 report
state the likely impact of Commonwealth Edison Company's committed modifications
on the results for the Zion plant results. This action would emphasize the fact
that the greatest importance of a PSA is in its use to improve safety by
revealing weaknesses that can be remedied.

4.3 External Events

4.3.1 General

The treatment of external events is not as complete nor as definitive in
NUREG-1150 as is the treatment of internal events. The reasons for this are:

- The "state-of-the-art" of the assessment methodology is not as refined
  as for internal events, and

- The assessment of external events (seismic and fire risks) was included
  as an appendage, rather than an integral part of the study. Thus, it
  was not practical to analyze more than two of the five plants studied
  in NUREG-1150.

4.3.2 Estimate of Seismic Hazard

A simplified approach was taken in NUREG-1150 in defining seismic initi-
ators, which leads to failure from all resulting transients, small or large.
Containment failure was based on broad assumptions rather than on structural
analyses.

Since the seismic contribution to risk is so large in cases where it has
been examined, we extend our attention to the source of uncertainty in its
estimation.

The estimates of seismic hazard use two different model sets of ground
motion attenuation, one developed by the Lawrence Livermore National Laboratory
(LLNL), and the other by the Electric Power Research Institute (EPRI). The EPRI
and LLNL models give very different estimates of seismic risk. To understand
why, it is necessary to consider the two models and their derivation.

Seismic risk is associated with large earthquakes rather than with small
ones, even though the larger seismic events may be centered at a greater distance
from the nuclear plant and will, naturally, be more rare. Therefore, the
attenuation of the ground motion over substantial distances becomes important.
Models of the modes of attenuation are important parts of seismic methodology.

Ground motion attenuation models of both EPRI and LLNL consist of two
parts:

- The basic model for estimating mean log ground motion as a function of
  earthquake size and distance from source, and

- The variability (randomness) in ground motion about the mean estimate
  caused by heterogeneous geological differences, seismic source term
  variations, and uncertainties in measurement.
Both EPRI's and LLNL's modeling of ground motion treat each of these parts as uncertain. They characterize uncertainty in the basic model by specifying alternative models (three by EPRI and eight by LLNL) to compute an average result. Each model is weighted to determine its contribution to the average.

EPRI (i.e., a group consisting of its consulting seismic scientists) assigned weights to each of its models based on a consensus of the goodness of fit to the available data. The primary model, EPRI-1, which was qualified against nearly 600 ground motion recordings in the eastern United States, was judged to yield the best fit, and therefore, was given a weight of 50%. The other two models (EPRI-2 and EPRI-3) are widely accepted in the peer-reviewed literature but are qualified with many fewer data, and, therefore, each was given a weight of 25%.

LLNL assigned weights to each of its models by averaging the independent recommendations of a panel of five seismic scientists. The eight models were weighted from a low of 6% to a high of 32%. It should be noted that a weight of 54% was given to spectral shapes typical of western U.S. earthquake sources, which have less high-frequency energy relative to eastern U.S. sources. Four of the five expert panel members gave one model (the G16-A3 model) a weight of zero; the fifth (the author of the model) gave the G16-A3 model a weight of unity and zero weight to the remaining seven models. Accordingly, the G16-A3 curve was given a weight of 20% in the LLNL hazard computation.

The weight of 20% given the G16-A3 model in the LLNL seismic hazard computations, due to the opinion of one expert, is the dominant reason for LLNL's hazard results being consistently higher and having a larger uncertainty than EPRI's. The difference is particularly large in the mean values of distributions and at rock sites. In the median, which is less sensitive to the tails of the distribution, the EPRI and LLNL predictions are reasonably consistent from site to site.

The seismic hazard analysis in NUREG-1150 shows how the final risk estimates and the associated uncertainty bands may be influenced by a single member of an expert panel, given the small number of experts on many panels. The seismic hazard analysis highlights important issues in the selection of panel members.

The uncertainties in total risk from nuclear power plants due to seismic hazards analysis may seem to be considerable. When evaluating these uncertainties, e.g., with respect to compliance with overall safety goals, the following points should be noted:

Nuclear power plants, which comply with seismic design criteria for a particular site, would most probably be damaged to the extent of giving rise to large releases only if a seismic event were to occur of such a magnitude that other societal damage in terms of loss of lives and property would be considerable. Much of the uncertainty in the ground motion models, which appear to dominate the uncertainty in the seismic hazards analysis of nuclear power plants, also applies to the estimates of risk of such other societal damage. Thus, the relation between risks to public health and safety from nuclear power plants and the corresponding risks from damage to other structures in the case of seismic
events as initiators appears to be less sensitive to uncertainties in local ground motion models than is the estimate of risk from seismic events. This finding should be kept in mind as the NRC safety goals are basically related to other types of risks through comparisons.

4.3.3 Analysis of Fire Risk

The analysis of risk from fires was limited to the Surry and Peach Bottom plants. By and large, the analytical methods were at the level of state-of-the-art. The possibility of destructive fires is important in the analysis of risk to nuclear plants because fires are potentially contributors to common-cause failures. However, most of the information on fires was in supporting documents which the Committee did not review.

The Committee believes fires are such important initiators of possible accidents that the analysis should have been extended to all five plants treated by NUREG-1150.

4.4 Expert Opinion

One of the distinctive features of NUREG-1150 was the extensive use of structured, formalized elicitation of expert opinion. This process provided input values and distributions for many of the parameters in the study for which values were not otherwise available or where the available results were incomplete, highly uncertain, or internally discrepant. The experts were generally asked to provide distribution functions for the parameters rather than point values. Latin hypercube sampling from these distributions was used to provide input values for the risk calculations, constituting one of the key steps in the generation of the uncertainties in the estimates of risk.

The expert opinion process involved several steps:

- **Selection of the expert panels.** Several expert panels were assembled. An attempt was made to include technical judgements from national laboratories, government, universities, and industry, endeavoring to include a wide range of views. This did not always succeed.

- **Training.** Professionals in the elicitation of expert opinion trained the panel members in that discipline. These same professionals provided guidance throughout the expert elicitation process.

- **Technical Presentations and Discussions.** The objective was to provide the experts with the information and relevant technical literature available on the subjects, and, consequently, to bring all the experts on a panel up to approximately the same technical background and level of understanding. The process involved presentations to the assembled experts by specialists in various aspects of the issues, and group discussions among the experts.

- **Elicitation Process.** After the training sessions, the experts were given several weeks to review the material, continue discussions, consult other experts, and make additional supporting analyses of their
own. In some cases, the groups were reassembled for additional discussions and presentations. Each expert provided his/her opinion on an individual basis in a private session with an individual trained in the elicitation process. The experts were also required to provide detailed documentation of the rationale for their opinions.

- **Results.** The values or distribution functions from the experts were averaged to provide those used in the analysis.

Expert opinion was elicited for the initial draft of NUREG-1150 but this was not the formal, professionally guided process described above, and most of the reviewers of the initial draft were critical of this first attempt at elicitation. Therefore, the elicitation was repeated using this more structured process. The comments of the Kastenberg Panel on the treatment of expert opinion in the first draft, and the views of this Committee on the changes made for this draft, are given in Chapter 6 of this report.

Expert opinion elicitation is technically less satisfactory 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. With this in mind, we comment on the expert opinion process of NUREG-1150 as follows:

- Formal, professionally structured expert opinion is preferable to the current alternative, according to which the individual PSA analysts make informal judgements which are not always well-documented. However, it is not as technically defensible as analysis using detailed, validated codes. The reproducibility of the results of expert opinion is a concern.

- Recognized professionals were employed to guide the process, with procedures that appeared to be state-of-the-art.

- 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 that for the first draft. Yet it still seemed to be unbalanced, in that the panels had 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.

- The training of the experts and their subsequent discussions were valuable in clarifying the focus on the important issues.

- The procedure for expert elicitation provided a structured method for introducing additional analytical and experimental results into the NUREG-1150 process.

- The process was well-documented. This documentation should prove valuable in future studies on the issues subjected to expert opinion.
The number of issues addressed by the expert panels was limited to those judged to be most important, due to the workload assigned the panel members, and the time available. Other issues, for which expert opinion was required, were addressed by the project staff without the same formal procedures being used. Even with the very limited number of issues presented to each panel, the workload on the individual expert was sometimes excessive. Because distributions were requested, many experts were asked to produce several thousand numbers, along with detailed supporting documentation.

Expert opinion may have been relied upon too heavily in some instances. An important example is the treatment of core cooling after containment failure, where expert opinion was used to argue that equipment would fail 70 - 80% of the time if environmental temperatures exceeded EQ 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.

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 and resources were lacking to update it. In these cases, the Sandia staff modified the expert opinion to treat the redefined issue. For example, expert structural opinion was obtained about the failure pressure and mode for steel-lined concrete containments. The experts' opinions focused upon slow pressurization, i.e., a time constant of hours. As NUREG-1150 evolved, the study team realized that it also needed to consider fast pressurization, i.e., a time constant of seconds, therefore, the Sandia staff extended the expert opinion to such situations. Unfortunately, these new calculations were not reviewed with the expert panel and are not reported in the NUREG-1150 Main Report nor in other documentation available to the Review Committee.

The study assigned equal weight factors to the opinions of all experts. Some other methods, which might develop unequal weight factors, were not used.

The elicitation of expert opinion is complex, time-consuming and expensive. Therefore, the full scope of this methodology may have very limited future application. It is unlikely that a 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 form of "expert judgment" is necessary in all PSA's.
4.5 Level 2 Uncertainties and the XSOR Codes

A key objective of NUREG-1150 was to determine the uncertainties in the values of risk. The Reactor Safety Study (WASH-1400) was criticized for not giving enough attention to these uncertainties. The procedure for evaluating uncertainties in the Level 1 (front-end) PSA's in NUREG-1150 was well established in previous PSA's. This was not the case for the Level 2 (back-end) calculations, however, which have been neglected.

To generate statistically significant output distributions in the Level 2 calculations, numerous calculations were necessary, each corresponding to a different combination of input parameters. The input parameters to each of these calculations were selected by the latin hypercube method. The calculation was repeated many times, each with a new set of randomly selected input parameters, until, after a large number of calculations, reasonable distributions were obtained for the output parameters.

Unfortunately, the codes normally used to perform the Level 2 calculations are large, detailed, and very expensive to run (i.e., the Source Term Code Package (STCP) or an alternate code). To repeat thousands of calculations with these codes was impractical; in fact, these codes were used for only a few (possibly 10 or 20) of the Level 2 calculations for each plant. Very simplified parametric codes were used for the remainder of the calculations. These were called the XSOR Codes (e.g., the SURXOR code was used for the Surry calculations). The XSOR Codes were simple mass-balance equations with constants in the equations determined from detailed calculations. In a simplified sense, the XSOR codes were normalized to the detailed calculations, and were used to interpolate between the few detailed results.

Therefore, 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 calculations. The remainder were calculated using the parametric XSOR codes. This trade-off met the need to generate many results in order to evaluate the uncertainties.

The XSOR codes themselves are mathematically self-consistent since they are simply mass-balance equations. The XSOR process is not exact, however, approximations being introduced in selecting the correct input values and constants for the codes and in ignoring, or greatly simplifying, the interdependence and timing. This was the cost of approximating the very complex physical processes in the Level 2 analysis by simple parametric equations.

Sandia National Laboratory and Battelle Memorial Institute have estimated the error introduced by using the XSOR codes. The results from the XSOR codes were compared to those from more detailed calculations and showed reasonable agreement; this was regarded as validating the XSOR process.

Caution is recommended in applying the XSOR methodology and using its results directly, because of these approximations. XSOR results seem valuable in screening results to determine dominant scenarios and for generating uncertainties in distributions, as they were used in NUREG-1150, but they cannot
supplant the more accurate methods for determination of point results of specific input variables.

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.

4.6 Key Issues in the Accident Progression Event Trees.

Some key issues deserve special discussion.

4.6.1 Arrest of Core Degradation before Vessel Breach

If core degradation were to become arrested before failure of the bottom head of the reactor pressure vessel, the structural integrity of the containment could only be threatened by large hydrogen burns, whose probability, however, is small for such sequences.

Core degradation may be arrested by early restoration of the emergency core cooling function. Such restoration may be effected by recovery of electric power in station blackout sequences, or by depressurization as a consequence of passive failures of parts of the pressure retaining boundary, such as failure of the main coolant pump seals and subsequent activation of the low pressure ECCS of a PWR.

If core degradation is arrested, the sequence ends in the accident progression bin "no vessel breach". Otherwise, it can end in one of the bins associated with containment failure. The bin "no vessel breach" has a relatively high conditional probability for all plant damage states of PWR's.

The capability to model the issue is rather poor. We cannot yet judge the validity of the conditional probabilities associated with the bin "no vessel breach". If the estimate of the conditional probability of this accident progression bin had to be lowered, the results would shift towards an increase of the conditional probabilities associated with bins responsible for high offsite consequences. This effect would be more pronounced for PWR's than for BWR's.

4.6.2 Failure of Main Coolant Pump Seals

The depressurization of the primary system after the failure of the main coolant pump seal is an issue important to the arrest of core degradation. The probability of pump seal failure was generated from elicitation of expert opinion. The aggregated density function reveals large uncertainties. The distribution is bimodal with two pronounced, widely separated peaks resembling delta functions.

We feel uneasy about the large uncertainty which expert opinion assigns this important parameter, which can be determined experimentally. The result introduces large phenomenological uncertainties into the question of depressurization via the pump seal. It will also cause difficulty in determining the effect of the new Westinghouse seals on the results of the Sequoyah, Surry, and Zion analyses. While it is generally accepted that these seals will reduce the
leakage rate, 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. The answer will impact both core-damage frequency and consequences in future assessments.

4.6.3 Temperature Induced Failure of the Hot Leg in High Pressure Sequences in PWR's

Another possible mechanism for depressurization of the primary circuit of a PWR in high pressure sequences is temperature-induced structural failure of hot leg piping. It is assumed that such failures would lead to less severe containment loads than a bottom head failure. These sequences are of high importance in risk for PWR's.

The analytical and experimental bases for the quantification of this issue are weak. Therefore, expert opinion was used to generate a probability distribution.

We note that only one of the three experts whose opinions were elicited provided a distribution function. The two others 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 these two statements were incorporated into the aggregated probability distribution presented in NUREG-1150. Therefore, we cannot judge the validity of the result.

4.6.4 PWR Containment Loads During High-Pressure Melt Ejection

If the bottom head of the reactor pressure vessel were to fail with the system at high pressure, large amounts of molten core and structural material, water vapor, and hydrogen would be ejected into the containment. An attendant pressure buildup in the containment atmosphere would result from a superposition of several effects:

- Blowdown of vapor and hydrogen
- Combustion of hydrogen
- Interactions of molten core material with water on the containment floor, and
- Direct heating of the containment.

In the NUREG 1150 analysis, the pressure rise at time of vessel breach was treated as one single issue summarizing the contributions from all four sources. Several parameters are thought to be important in this analysis:
- pressure in the reactor vessel
- amount of unoxidized metal in the melt
- fraction of the molten core ejected
- initial size of hole in reactor vessel
- availability of water in the reactor cavity
- operability of containment spray system.

Some of these parameters are highly uncertain, and their combined effects on containment loading are still more uncertain. The uncertainty in the containment load curves does not seem to be important for the strong containments of the Surry and Zion plants. For Sequoyah, however, small changes in the containment loads curves cause significant changes of the probability of containment failure.

In the initial draft of NUREG-1150, direct containment heating (DCH) and hydrogen combustion were the major contributors to early containment failure (ECF) for PWR's, and ECF was the dominant contributor to risk. In the current draft of NUREG-1150, this situation has changed dramatically. The containment bypass sequences dominate risk for the PWR's (as discussed in Section 4.2.1) because of a large reduction in the probability of ECF. This reduction in ECF in the current draft is the result of three factors.

- There is a large increase in the probability that the RCS would be at a reduced pressure before melt-through of the vessel.
- Given DCH, the calculated pressures in the containment are lower.
- The estimated strength of the containment is greater.

The considerations that contribute to the increased probability of pressure reduction in the RCS prior to vessel melt-through include depressurization by the plant operators, melt-through of the hot leg, a stuck open-relief valve, and failure of the seals of the reactor coolant pumps. Unfortunately, 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 ECF.

4.6.5 Basemat Melt-through of PWR Containments

The CORCON code was used to model the erosion of concrete by molten corium. Calculations for the Surry plant suggest that basemat penetration would occur not earlier than 5 days after accident initiation, if at all. This result is derived by extrapolating from a calculation which could not be carried beyond 1.1 days of continuous computer operation. The speed and amount of erosion of the concrete strongly depend on the distribution of the decay heat into the fraction consumed to erode the concrete, the fraction consumed by evaporation of water, if present, and heatup of the containment atmosphere.
In the CORCON calculation, this division is much less in favor of concrete erosion than in other computational models, for example WECHSL, which has been validated by the BETA-experiments for dry conditions\(^8,9\). We suspect that the concrete erosion progresses faster and with greater intensity than is estimated in NUREG-1150, with a corresponding increase of hydrogen production. However, we agree with the assessment in NUREG-1150 that the melt-through per se has no important influence on health risk.

4.6.6 Hydrogen Production in the Ex-Vessel Phase in PWR's

The rate of hydrogen generation in the ex-vessel phase of a core melt accident depends on the coolability of the debris, and on the molten core-concrete interaction, if the debris is not coolable.

The coolability of the debris bed is influenced by the mode of vessel breach and the amount of water available in the cavity or in other parts of the containment building. Significant erosion of concrete by molten core material is unlikely if water is present in the cavity at time of vessel breach. However, there is insufficient information on the probability of availability of water, and on the mode and size of vessel breach.

If the debris is not coolable because there is no water, the generation rate of hydrogen essentially depends on the speed and intensity of the molten core-concrete interaction. For reasons explained in the section on basement melt-through, we believe that this process is modeled incorrectly, so that the hydrogen generation rate in the ex-vessel phase of accidents in PWR's is underestimated.

4.6.7 Drywell Shell Melt-through in BWR Mark I Containments

If a severe accident were to occur to the Peach Bottom Plant, leading to melting of the reactor core, early failure of the BWR Mark I containment might result from molten core debris penetrating the steel containment shell. This failure mechanism has the potential for severe offsite consequences. According to the NUREG-1150 analysis, the accident progression bins associated with drywell melt-through are responsible for about 90% of the calculated early and late fatalities. This result was derived from the conditional probabilities for drywell melt-through generated by an expert panel. The judgment of the individual members of the panel is nearly binary, i.e., the panelists either believe that the drywell would almost always fail or that it would fail very rarely; individual judgment is nearly independent of initial and boundary conditions. The aggregate distribution 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 is clearly demanded.

4.7 Remarks on Uncertainty Analysis

Uncertainty analysis is an integral part of PSA and one of the most controversial. NUREG-1150 has made significant contributions in at least two areas, namely, model uncertainties and the formal use of expert opinions. While most work before PSA focused almost exclusively on parameter uncertainties, NUREG-1150
recognized explicitly that our incomplete understanding of important phenomena often leads to different models that may be the dominant contributors to uncertainty. The formal use of several models in PSA requires an assessment of their credibility, and this was achieved by eliciting expert opinions, as discussed elsewhere in this report. 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.

It is important to realize that the kinds of uncertainty that are of main interest in PSA's are due to lack of knowledge. (The opening of a valve upon demand is a stochastic event whose outcome is not known; however, this is not the kind of uncertainty with which PSA's are concerned, rather, the uncertainty on the numerical value of the frequency of the valve's failure to open is the state-of-knowledge uncertainty that a PSA would typically attempt to quantify.) The distributions that express this uncertainty are often called subjective, and they are generated from expert judgment and statistical evidence, if available. Statistical information is typically available for frequencies of events that appear in the front end of the PSA. For the so-called back end, expert judgment dominates. The question, then, is whose judgment ought to be used.

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 such 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.

4.8 Human Reliability

4.8.1 Introductory Comments

Human Reliability Analysis (HRA) is recognized as a very important part of PSA, and yet one of the weakest. The TMI accident focused the attention of the industry and regulatory authorities around the world on the significance of human actions in preventing and managing incidents and accidents.

NUREG-1150 is a major study. Its methods and results will find many uses, e.g., in the resolution of generic issues, the review of Individual Plant Evaluations (IPE), and the identification of areas for further research. Therefore, we deemed it important to address the following issues in our review of NUREG-1150's HRA:

- The methods used for HRA and the associated uncertainties.
Human actions and factors that are left out of the analysis, but, nevertheless, may have a significant influence on overall estimates of risk.

To illuminate some of the points made, we end our comments by discussing in more detail the HRA of one particular sequence, namely the ATWS sequence in the two BWR's analyzed.

4.8.2 Methodology

Modeling the thinking processes of operators and their interaction with the plant systems is difficult. Several human reliability models have been proposed in the literature, and research is active in this area. NUREG-1150 has predominantly used one of these models, namely, the Accident Sequence Evaluation Program (ASEP) HRA procedure, which is based heavily on the THERP methodology and is considered as one of the state-of-the-art methods in PSA applications.

However, benchmark exercises indicate a fairly large spread in the results obtained when different methods of HRA are used, and also between the results obtained by different analysts using the same method. This was evident in the findings of the Human Factors Reliability Benchmark Exercise (HF-RBE) organized by the Ispra Joint Research Center of the Commission of the European Communities[10]. Teams from several countries used various HRA models to estimate human error rates for both pre-accident and post-accident tasks. The results reached by different teams differed significantly, and the organizers concluded that "...human reliability analysis is an art rather than a science, and it is too early to specify preferred ways of performing the analysis..."

The NUREG-1150 team, in their presentations to us, confirmed that other models could have been used and that the uncertainties are substantial. The argument has been advanced that the conservative screening procedures that were employed and the wide uncertainty ranges that were assigned to the error rates include the results that other models would have generated. However, such an approach goes against the presumed goal of a PSA, 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. Indeed, one of the observations of the Ispra HF-RBE is that THERP tends to give lower results than those of the Human Cognitive Reliability model, thus creating the suspicion that there may be systematic biases. When uncertainties are estimated, it should be kept in mind that, in reality, there may be a large variation in performance shaping factors, depending on the actual situation (e.g., time of the day or night) and the specific characteristics of the control room crew on duty. In fact, some of the factors influencing the uncertainty bands in the human error probabilities at a particular plant may be associated with the concept of "safety culture" (see section 4.9 below).

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. Our preceding comments are not intended to address the individual merits of THERP or other models. On-going research both in the
United States, primarily sponsored by the NRC and the Electric Power Research Institute, and also research abroad, may eventually answer these questions.

However, we note that NUREG-1150 has pioneered the explicit treatment of model uncertainties and the use of expert panels to weigh the relative merits of alternate methods of analysis, yet did not employ this approach for human actions. Experts were consulted for two operator-related issues, namely, #5: Innovative Recovery Actions for Long-Term Sequences Involving Loss of Containment Heat Removal, and #10: Use of High Pressure Service Water Spray in the Dry Well. However, these experts were not asked to assess the impact of using alternate models, as we discussed above. Especially notable is the fact that expert panels were not used to address the treatment of errors of commission, and the methods and data used in the HRA of some very complex situations in the control room, such as the early phase of an ATWS sequence in a BWR.

4.8.3 Errors of Commission

The only errors of commission covered by the HRA methods used appear to be those caused by deviation from proper maintenance and test procedures, though some others may be implicitly included in empirical failure rates of systems. The NUREG-1150 study itself recognizes that errors of commission emanating from misdiagnosis of a degraded safety state or of an accident in the making are not considered. We would point out that in some PSA's, e.g., those for the Oconee\(^{11}\) and Seabrook nuclear power plants\(^{12}\) an attempt was made to at least structure the problem using "confusion matrices". In our opinion, such errors of commission not included in the analysis might contribute to risk an amount comparable to that from some mechanistic initiators. This opinion is based on human factors analysis of several incidents in recent years\(^{13,14,15,16,17}\), indicating that serious errors in decision-making in the control room, driving the plant into a degraded safety state with respect to defence-in-depth capability, may have a frequency of occurrence comparable to such serious technical disturbances as rupture of steam generator tubes, on which substantial analysis efforts have been spent. We note that serious errors in the decision-making process in the control room were among the contributing factors to both the TMI and Chernobyl accidents.

PSA models assume that all the actions of operators are guided solely by the operators' desire to bring the plant to a safe state. This is not necessarily true. Conflicts of interest have been observed\(^{15,16}\), and are recognized in at least some of the HRA procedures used (e.g., in the analysis of the ATWS sequence discussed in the following Section) by introducing a "reluctance factor" among the human performance-shaping factors. Their importance is also recognized by the industry and regulators, in stressing predominance of the need to protect the public and the plant. Bearing in mind the difficulty in quantifying the effect of attitudes, which, in our opinion, is beyond the state-of-the-art in PSA, it is nevertheless important to recognize the potential significance of such reluctance factors and countervailing compliance factors when NUREG-1150 is used for risk evaluation and risk management.

Collecting field experiences and simulator data is probably the most credible way to address this issue. A start has been made through EPRI's Operator Reliability Experiments (ORE) in which a limited set of data on errors
and their causes was collected from several plant simulators, using actual operating crews and accident scenarios. Analysis of these data is underway and will be expanded to understand causes of these errors and to look into practical means for modeling, quantification, and integrating them into PSA's.

4.8.4 The ATWS Analysis as an Example

To illuminate some of the issues raised in the preceding sections, we have reviewed in more detail the HRA performed for the ATWS sequence in the two BWR's (Peach Bottom and Grand Gulf), with special emphasis on manual initiation of boron injection using the Standby Liquid Control (SLC) system. A principal reason for this choice was that the ATWS sequence is among the principal contributors to risk from internal events for the two BWR's, with a fairly high conditional probability for early containment failure. Furthermore, the sequence is characterized by complex interactions between members of the control room crew in a short interval (about five minutes) of high stress at the start of the event sequence.

While the ASEP HRA procedure is the dominant one in NUREG-1150, exceptions occur in the BWR ATWS sequences. The Grand Gulf sequence was analyzed in detail using THERP by a Sandia National Laboratories (SNL) team headed by Alan D. Swain, the principal developer of THERP. This analysis includes insights gained through plant visits, the review of training manuals and emergency procedures, as well as the performance of three ATWS scenarios on the Grand Gulf simulator. The Peach Bottom ATWS sequence was analyzed by a Brookhaven National Laboratory (BNL) team. Insights from plant visits and reviews of procedures also were used by this team; however, the quantification of human error rates is carried out using a certain set of time reliability correlations (not those used by THERP or ASEP).

The NUREG-1150 PSA team was asked to give a more detailed presentation of the HRA performed for the ATWS sequence in the Peach Bottom PSA\(^{(19)}\) while only the detailed documentation was examined for Grand Gulf\(^{(20)}\). The presentation of the Peach Bottom sequence demonstrated good traceability of the methods and data used in the analysis, as did the detailed documentation of the Grand Gulf case.

It is interesting to compare the results of these two analyses for the same human action, namely, failure to initiate Standby Liquid Control (SLC). The Peach Bottom analysis estimates that the operators must initiate SLC within 4 minutes from the beginning of the accident to prevent the temperature of the suppression pool from becoming excessive (the Main Steam Line Isolation Valves (MSIV) are assumed closed). The probability of this human error is estimated by the BNL team to be 0.02 (mean value). The uncertainty distribution is estimated using human error probabilities (HEP) from four previous studies of similar sequences ranging between 0.26 and 0.01\(^{(19)}\). The Grand Gulf analysis estimates that the operators have 2 to 7 minutes to initiate SLC and the probability of failing to do so is 0.0001.

The question that inevitably arises is how much of this substantial difference in HEP's is due to the different methodologies employed and to the different groups of analysts using them. The documentation fails to reveal any differences between the layouts of the two control rooms of major significance to the HEP in this sequence. Also, the Grand Gulf analysis cites two factors
that would contribute towards a higher HEP in Grand Gulf than in Peach Bottom. These are the necessity to fetch keys to operate the SLC switches, and the assumption that only two operators are initially available in the Grand Gulf control room to cope with the numerous tasks called for in the first minutes of this transient, versus three in Peach Bottom.

The methods and data used in the analysis of this particular situation raise several questions. Indeed, it may be questioned if the relatively simple models used in NUREG-1150 for the ATWS cases are the most appropriate ones, when analyzing a complex, high-stress situation involving communication between several persons, each with multiple tasks to perform.

In fact, records of actual behavior by the control room crew in real stress situations of a broadly similar nature (loss of all feedwater in Davis Besse (1985)\(^{18}\), loss of power to ICS in Rancho Seco (1985)\(^{21}\)) indicate that crews may initially focus all their efforts on one action strategy, which to them appears technically sound in the perceived context, but is not necessarily the strategy prescribed in the procedures. If the chosen strategy is not successful, they may easily use up the time window available for the prescribed action if this time window is as short as about five minutes.

In our opinion, it would have been valuable if the theoretical HRA’s of the ATWS sequences had been tested against real events, such as those cited above, as a basis for an in-depth analysis of uncertainties in HRA. This test could be done as part of the input of expert opinion 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 for a number of back-end issues of similar importance as measured in their contribution of overall risk.

4.8.5 Conclusions

NUREG-1150 shows that substantial progress has been made since WASH-1400 in human reliability analysis, including consideration of recovery actions. However, additional research should be devoted to errors of commission.

4.9 Management Influence

As already stated above, NUREG-1150 is a major study, and its methods and results will find many uses, e.g., in the resolution of generic issues, the review of Individual Plant Evaluations (IPE), and the identification of areas requiring further research. Therefore, it is important to have a clear picture of what is left out of the analysis. (Some areas of concern were addressed in previous sections.) Recent experience has led safety experts to the belief that the quality of plant management has a decisive influence on the safe operation of a plant, with an impact on PSA that has not yet been thoroughly investigated and understood.

The International Nuclear Safety Advisory Group (INSAG) advances the view that a fundamental responsibility of management is the establishment of a safety culture governing "...the actions and interactions of all individuals and organizations engaged in activities related to nuclear power..."\(^{22}\) Such a culture would allow "...an inherently questioning attitude, the prevention of
complacency, a commitment to excellence, and the fostering of both personal accountability and corporate self-regulation in safety matters."

While it is beyond the current state-of-the-art to identify quantitative measures of safety culture, most experts agree that recent major accidents (e.g., Chernobyl and TMI) were, in part, due to the failure of senior management to establish such a culture (see also the discussion on the use of "reluctance factors" in our comments on human reliability analysis). The available PSA models cannot account for the influence of management quality on risk and, hence, it is understandable that NUREG-1150 does not address these issues. In fact, we doubt that the concept of management quality may be factored into PSA in a quantitative way, either at present or in the near future. The impact of management quality on safety is currently addressed through other activities pursued by INPO and the NRC. However, as stated above, it is important to bear in mind that management quality is not reflected in the risk curves when the insights and results of this study are used.

4.10 Cutoff Criteria

4.10.1 General

It is important that all essential contributions to risk be taken into account in probabilistic safety assessment. On the other hand, it is not reasonable to wish to evaluate all conceivable accident sequences, nor is it possible to do so. Therefore, criteria are needed to distinguish between what is to be considered in the analyses and what is to be neglected. These are called cutoff criteria. NUREG-1150 has cut off its curves at probabilities of $10^{-10}$/ry, which appears to be on the low side. Clearly, the calculations did not include numerous natural phenomena of severe destructive capability that might have caused very serious consequences, as well as consequences from other modes of operation than full power. The following comments outline a basis for more effective cutoff criteria.

In the front-end analysis, the cutoff criterion is often based on the frequency of the sequence, with sequences neglected if their frequencies are below the cutoff. If the neglected sequences are not associated with completely new phenomena, this cutoff cannot noticeably influence the results if the chosen cutoff frequency is sufficiently low.

In the back-end analysis, the calculated distributions can also include consequences at extremely low frequencies. These low-frequency contributors are associated with large uncertainties and they do not contribute appreciably to risk. Therefore, a cutoff criterion should also be applied in the back-end analysis to eliminate them.

Meaningful quantitative cutoff criteria require considering the level to which frequencies are really needed, and to which meaningful results can be calculated in probabilistic analyses.
4.10.2 In Connection with Low Frequency Sequences

It is well-established practice in reactor safety in general, and in PSA in particular, to consider families of events and plant damage states. This practice greatly reduces the likelihood of omission of accident sequences that should be included. In the front-end analysis, initiating events are usually grouped into families based upon the similarity of physical phenomena or the response needed from plant systems. Depending on the system's failure modes, different sequences of events within a single family may finally lead to different physical phenomena and consequences. Therefore, it is appropriate to provide a different grouping at the back end of the analysis. It is helpful for that purpose to define plant damage states that include all sequences leading to a physical condition of the plant with common attendant outside consequences (source terms).

In each plant damage state, the families of events with high probability of occurrence dominate the calculated contribution to risk. Therefore, excluding low-probability sequences from the analysis will not change results significantly. Which cutoff frequency is appropriate depends on the classification of event families and on the frequency of the dominant risk contributors. 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 $10^{-5}$/yr, a cutoff frequency of $10^{-7}$/yr seems appropriate.

The situation is different if entire plant damage states are neglected. Dropping an entire plant damage state might cause an entire class of consequences to be dropped from the analysis. But then it is not reasonable to analyze in detail plant damage states whose frequency is below that of catastrophic failures like that of the reactor pressure vessel, for which the conditional probability of severe offsite consequences could be high. Present understanding sets the upper bound for the frequency of such a failure at about $10^{-7}$/per plant and year for a pressure vessel that has an acceptably low nil ductility temperature, including the region of the welds.

4.10.3 In Connection with Low Risks

PSA is increasingly used for decision making, in particular, for identifying means for further risk reduction. The consideration of small contributions to risk is not helpful in this context, in particular if their calculation is influenced by large uncertainties. Therefore, decision making normally includes a de minimis concept providing a clear-cut distinction between a substantiated real risk which is to be limited and reduced, and insignificant risks that are not reliably assured. A de minimis threshold can best be established by considering comparable risks in other areas of human action that are commonly accepted as insignificant.

* This definition is not the same as that given in NUREG-1150.
** de minimis non curat lex
Several countries have adopted safety goals associated with the risk of accidental death of individuals ($\geq 2 \times 10^{-5}$/yr, depending on age). Associated cutoff values in the range of $5 \times 10^{-7}$/yr to $10^{-6}$/yr are used in this connection. Risks below those limits constitute only a small fraction of the total fatality risk from all causes.

For individual risk of late cancer fatality that might be induced by radiation, natural background radiation provides an appropriate scale for comparison. Though the dose from natural radiation varies widely over the earth, a typical rate is 2 mSv/yr. Since the risk coefficient is about $5 \times 10^{-5}$/Sv*, the related committed annual risk of death from cancer induced by natural radiation is about $1 \times 10^{-4}$/yr. A lower limit to the geographical variation of that risk would be well above $2 \times 10^{-5}$ Sv/yr; in some parts of the world this value is as high as $5 \times 10^{-2}$ Sv/yr. Thus, modification of the risk by an amount below $2 \times 10^{-5}$/yr can be considered insignificant compared to the natural variability of this risk. The conclusion is even stronger when it is noted that there is no proof that radiation at low dose and low dose rate is harmful. Restriction of the probability of latent cancer fatalities to less than $2 \times 10^{-6}$/yr per year, as implied by the safety goals used in the United States, is far below that limit and well within the range where the contribution to the overall cancer risk ($-2 \times 10^{-7}$/yr) is negligible.

Thus, it is reasonable to neglect individual risks which are about one order of magnitude or more below the value associated with the US safety goals. A de minimis threshold of $10^{-7}$/yr would appropriately represent this reasoning. Reduction of risk to values below that level would not affect the overall risk to an individual. The results of risk analyses of consequences with lower frequencies are not meaningful for decision making, because the risk of events with probabilities below $10^{-7}$/yr is definitely dominated by large natural or other manmade catastrophes.

4.10.4 Conclusions

We believe that a realistic cutoff in both frequency of severe accidents and their resultant risk is warranted, and should be encouraged in all PSA's. The preceding considerations indicate that event families and plant damage states with frequencies below about $10^{-7}$/yr should be neglected in probabilistic risk analyses. In addition, a health risk in the range from $10^{-2}$ to $10^{-3}$ times the normal occurrence rate also seems reasonable. For curves of accident magnitude vs frequency, a cutoff at from $10^{-7}$/ry to $10^{-8}$/ry in frequency seems warranted.

4.11 Display of Results
4.11.1 General Comments

In the first draft of NUREG-1150, the numerical results on risk were presented according to a "box and whiskers" concept which gave an indication of the ranges of distribution in risk without reporting details of the distributions or the principal statistical measures of mean, median, or the percentile brackets.

* ICAP 90
The motivation was apparently to respond to criticism of the presentation of results in WASH-1400, where it was considered that insufficient attention was given to the uncertainty in results.

However, the course adopted for the first draft of NUREG-1150 was itself criticized heavily in the subsequent peer reviews, on the grounds that it had gone too far in the other direction. Essential information that should have been available because it had been generated by the analysis was suppressed by the way the results were shown.

The second draft, reviewed by this 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.

However, two other questions arise as a consequence of the choice and its results. The first concerns what to do in the face of distributions that are asymmetrical and very broad, covering several decades of variability, sometimes with bimodal shapes. This pattern has usually resulted from differences of opinion among individuals elicited for their expert opinion. The second question also results from the broad statistical spread in the results, which causes large differences between the statistical values of the mean and the median values of distributions. These questions are addressed in turn.

4.11.2 Wide, Asymmetric, and Bimodal Distributions

At first appearance, the unusually wide distributions in risk generated by the NUREG-1150 analysis are surprising and confusing. They are, however, a natural result of the elicitation of expert opinion on phenomena that occur under very unlikely conditions, and that are poorly understood.

Individuals who analyze the effects of events with low probability are accustomed to thinking in terms of powers of ten. When such people use an expression such as "approximately $10^n$" or "of the order of $10^n$", they have in mind variability of the exponent rather than the coefficient. Since expert opinion is sought on rare and poorly understood events, the distributions that are proposed typically range over decades.

Furthermore, the information base on such events is, by its nature, sparse. Therefore, these experts have little by way of experience to guide them, so that various elements of bias may be introduced in the opinions of individual experts.\(^{(23)}\)

The sparseness of data, combined with the poor understanding, causes the wide probability distributions seen in the NUREG-1150 inputs and outputs. The same situations also account for the bimodal shapes, because the inadequacy of information is commonly accompanied by polarized views and overconfidence in personal judgement.

Little can be done by way of methodology to improve this situation. As long as the analysis aims to incorporate the breadth of informed opinion, wide and even occasional bimodal distributions will be generated. The best that can
be done is to recognize their origins, and to make allowances for them. In this setting, as in many others, prudent advice would seem to be to scrutinize carefully all extremes of viewpoint.

It can be hoped that, in the long term, the accumulation of experience will 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.

4.11.3 Means or Medians?

It has been said many times that the "bottom line" results of a PSA should not be used in regulatory decisions. By this it is meant that the uncertainty distributions attached to risk are so wide that a judgment as to whether a particular plant is safe enough should not rest on a single value of risk, as calculated from its PSA.

Yet it is sometimes necessary to approach doing just that. Three examples come to mind. The first is encountered at the design stage of a plant, when there are design choices to be made, the preference being determined, in part, by safety considerations. The comparative influence on safety of the alternatives is determinable, in part, from a PSA type of analysis. Though this analysis may often be very rough and incomplete, in some modern applications the process can sometimes be continued to an essentially complete product. Then, a single number from the PSA for each alternative is the basis for comparison.

A second and similar example is attached to exercises such as the IPE now underway. A major objective to identify weaknesses in design or operations of a nuclear plant. This will be done by determining the effect of a design or operational feature on risk as it is determined by point values.

The third example is a result of adoptions of safety goals, which are usually expressed qualitatively but are interpreted quantitatively in terms of point values of risk, such as short-term or long-term health effects of accidents. It is made clear that they are not to be used as measures of whether individual plants are safe or unsafe. Yet if single statistics on risk for an individual plant greatly exceed the values used for the safety goal, strong pressure is felt to improve the situation. And if the point values indicate conformance to the safety goals, the tendency is to accept the situation and move on to the next question.

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.

The matter assumes substantial importance because asymmetries in distributions cause means and medians to be well apart in value. In formation of the mean value of a variable as
\[ \int _{x}f(x)\,dx \]

the larger values of \( x \) tend to dominate the averaging process, especially because of the typical spread of the probability distributions over some orders of magnitude. The contribution to the integral at smaller values of the variable \( x \) is, generally, not very great. As a result, the mean values of input and output variables tend to be located near the upper ends of the distributions, while the medians are found naturally at the midpoint, with equal areas of the distribution on either side. Some of the distributions of risk derived in NUREG-1150 have mean values outside the 95\textsuperscript{th} percentile ranges.

In engineering circles, where expert opinion is sometimes obtained from several competent persons, engineering reality is generally thought to lie in the region where there is a preponderance of agreement among the experts. An outlier in the form of a dissenting opinion on the side of pessimism might alter an engineering decision to cause it to lean more toward the conservative side, but this would be regarded more as prudence than a change of opinion as to where realism lay. Generally speaking, in determining answers to straightforward engineering questions, the tendency would normally be to settle for the predominant weight of engineering judgment, or an answer near the median, of course after the introduction of a safety factor.

On the other hand, if the question is motivated by safety considerations, greater weight would have to be given to the conservative, more pessimistic estimates. This would lead to a preference for the mean, which has that character, or an even higher point on the distribution.

From these considerations we conclude that the current form of display of the results in NUREG-1150 is preferable to that in the first draft. Presentation of the means and the medians along with the distributions allows readers to extract the information most suited to their purposes.

4.12 Completeness and Uncertainties in Overall Risk Estimates.

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. Of course, such caution is also needed when more conventional deterministic methods are used. Such caution becomes especially relevant when discussing overall probability estimates of catastrophic events of the order of \( 10^{-6} \) per reactor year or less. In our review of NUREG-1150, we identified such reservations in the following areas:

- Certain potentially important effects are not explicitly or fully covered: events starting from the low power and shutdown modes, sabotage, and aging which may not be fully covered by current inspection and maintenance programs.

- Completeness of modeling of interdependencies of technical systems, including detailed modeling of auxiliary systems, formally regarded as not safety-related. The contribution to overall risk from the
component cooling water and service water systems identified in the Zion PSA is one example. A similar risk contribution was found in the Swedish Ringhals 1 PSA from some electrical protection circuits, which turned out to be common to both safety-related and non-safety related equipment.

- Completeness and uncertainties in the area of HRA, especially with respect to the treatment of errors of commission.

- Completeness and uncertainties associated with the analysis of external events.

- Uncertainties associated with probabilities mainly based on expert judgment, especially where considerable divergence of opinion existed.

- The impact of "safety culture" and management quality is not included. Although this impact cannot yet be factored into the PSA, it is important to bear in mind such impacts as overall decisions are made on plant safety.

There are also uncertainties in the modeling of consequences due to decisions that would be made only during, or after, a severe accident. These decisions are of a socio-political nature and include such things as evacuation, interdiction of land and foodstuffs, and the valuation of real property. These uncertainties were not included in the NUREG-1150 analysis of consequences, although they have been discussed elsewhere. (24, 25)

Nevertheless, NUREG-1150 is a substantial step forward in clarifying various contributors to risk and in developing PSA methodology, not least in the exposure of uncertainties.

Taking into account the remaining uncertainties in the PSA methodology, e.g., with respect to completeness in the treatment of human factors and external events, estimated core-damage probabilities much below 10^{-3}/ry should be regarded with some caution. Taking into account that the resilience of a well-designed containment is largely independent of the particular type of core-damage sequence, this indicates that a risk figure for a large release based on a core-damage frequency of 10^{-4} to 10^{-5}/ry and a conditional probability for containment failure of 10^{-2} might be assigned a higher credibility than a risk figure based mainly on a low core-damage frequency. (See also Section 4.10 on cutoff criteria.)

Many of the limitations and uncertainties mentioned above may be reduced by improved PSA methodology and by improved experimental and empirical data. Such improvements should be made part of the IPE program, but not delay it. We note that many improvements in methods and data have become available since the closure date for the NUREG-1150 analysis.

In particular, 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 with quantitative safety goals.
4.13 A Tool for Risk Reduction and Risk Management

Some wide uncertainty bands (and associated contributions to the mean value of risk) may be reduced by proper application of risk reduction and risk management techniques, using the insights gained from the PSA to modify the plant and its procedures. The NUREG-1150 methodology is of special value in this respect, because it allows a more sophisticated approach to risk management, addressing not only major contributors to risk, but also contributors associated with large uncertainty bands. Cases of special interest include sequences where the risk of high consequences is mainly driven by the overlapping tails of probability distributions for two events, e.g., the probability that the containment pressure exceeds a certain value and the probability of containment failure at that pressure.

One approach to risk management in such cases would be to consider as tolerable a small increment in the probability of an event with small or moderate consequences as a trade-off for a substantial reduction of a large uncertainty band associated with a high-consequence event, even though this event has a low point value estimate of probability. This has been the case, for instance, with the risk from sequences leading to early containment failure of the Mark I BWR containments. The filtered containment venting systems installed in PWR nuclear plants in some countries exemplify such an approach, where the issue of uncertainties in failure of the containment from overpressure is resolved by accepting a possible small increment in the probability of a minor radioactive release by unwarranted operation of the filtered vent system.

4.14 Presentations of Additional Results

The presentation of the final risk results is much influenced by tentative safety goals of the USNRC, expressed as individual and societal health risks from accidental exposure to radiation. 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. Such safety goals as those used in Europe do not require an elaborate level 3 PSA with evacuation modeling. 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 their later publication might especially enhance the value of the NUREG-1150 study outside the United States, since many may not be calculable from the data in the report.
5. COMPARISON WITH WASH-1400

5.1 Introduction

Major progress has been made in severe accident technology and risk assessment methodology since the publication of the pioneering Reactor Safety Study, WASH-1400. NUREG-1150 is a comprehensive statement of the use of these new capabilities in updating the risk assessments of nuclear power plants. It is of interest, therefore, to examine the changes which have occurred in the results of those risk assessments. The comparison must be limited to the Surry PWR and Peach Bottom BWR, the only two plants evaluated by WASH-1400. In addition, the comparison is limited to median results and internal events since WASH-1400 did not compute the mean results nor explicitly treat external events.

The changes have resulted from two broad categories of progress:

- There has been a major increase in data on equipment reliability and in the analytical methods for the transient behavior of systems, which give greater insight into accident initiators. These data resulted from continuation of pre-WASH-1400 R&D and from increased attention to understanding, avoiding, and mitigating the possible small-break, loss-of-coolant accidents whose importance was shown by WASH-1400. Many of the programs were jointly sponsored by the NRC, the suppliers, and the utilities through the Electric Power Research Institute; several were conducted by organizations in other countries or jointly with such organizations.

- A radical infusion of experimental data and an extensive development of analytical methods have improved mathematical analysis of engineering questions pertinent to severe accident progression, containment performance, and the severe accident source term. R&D in these areas was relatively sparse before WASH-1400, but was accelerated greatly after the TMI accident.

5.2 Core Damage Frequency

The median core damage frequency (CDF) for Surry is reduced from 6x10^{-5}/ry in WASH-1400 to 2.3x10^{-5}/ry in NUREG-1150 (a factor of 2.6) and for Peach Bottom from 2.9x10^{-5}/ry to 1.9x10^{-6}/ry, (a factor of 15).

Modifications of the Surry plant since WASH-1400 provided cross-connection of the high-pressure safety injection systems, auxiliary feedwater systems, and the refueling water storage tanks for the two units, measures which have substantially reduced the probability of core damage from loss-of-coolant accidents. Although NUREG-1150 added reactor coolant pump seal failures as a new initiator

*Edward A. Warman, Sr. Consulting Engineer of the Stone and Webster Engineering Corp., has provided us with the extracted information from which comparisons have been drawn.
to the small break LOCA sequence, thereby increasing the probability of a small break loss of coolant by a factor of ten, plant modifications offset this increase, leading to the overall decrease in core-damage probability for Surry.

Ninety-six percent of the median CDF estimated in WASH-1400 for Peach Bottom was due to ATWS sequences and failure of long-term decay heat removal. The risk from core damage estimated in NUREG-1150 as resulting from failure of long-term decay heat removal is substantially reduced because Peach Bottom was modified to permit venting of the containment. The CDF from ATWS sequences was reduced in NUREG-1150 because the plant implemented ATWS fixes, and modern neutronic and thermal-hydraulic analyses of the ATWS sequences have resulted in lower calculated core power levels during the events, allowing more opportunity for mitigation. As a result, station blackout has become the largest contributor to core damage.

The range of uncertainty in CDF as estimated by the ratio of the median to 95th percentile is not greatly different in WASH-1400 and NUREG-1150 for either plant, i.e., a factor of 5.8 and 5.6, respectively, for the Surry analyses, and a factor of 4.5 and 6.8, respectively, for the Peach Bottom analyses.

5.3 Accident Progression and Containment Performance

The median cumulative containment failure pressure of the Surry reinforced-concrete containment was estimated for NUREG-1150 to be 130 psig rather than the 80 psig estimated for WASH-1400. This revision results from empirical data that became available after issuance of WASH-1400, and analytical methods improved since then. The increase is especially important to PWR dry containments. The failure pressure of the Peach Bottom steel-shell containment is estimated as 150 psig for NUREG-1150, close to the estimate for WASH-1400.

A direct comparison of CDF assigned to individual accident progression scenarios cannot be made, because only median data are given in WASH-1400 and only mean data are given in NUREG-1150 for the individual accident scenarios. However, the percentage contributions of the individual scenarios to the total mean CDF were compared by the Committee, with the results as follows:

- In the WASH-1400 Surry analysis, 72% of the CDF was associated with LOCA's and containment bypass events, and 28% with transients. The reverse is observed in NUREG-1150, wherein 23% of the CDF is associated with LOCA's and containment bypass, and 77% with transients and station blackouts.

- Although containment bypass events account for only 8% of the CDF for Surry in both studies, these sequences dominate off-site risk due to the large releases. Although considered in both studies, containment bypass, i.e., interfacing systems LOCA's, is not a significant risk contributor for Peach Bottom.

- All core-damage accidents were assumed in WASH-1400 to result in containment failure. For Surry, 24% of the severe core-damage sequences resulted in early containment failure or bypass and 76% resulted in basemat melt-through. In NUREG-1150, the containment remains intact.
in 82% of the Surry core-damage sequences, and early containment failure is calculated to occur for accident sequences constituting only 0.4% of the core-damage frequency. In the WASH-1400 analysis of Peach Bottom, 100% of core-damage sequences were assumed to result in early containment failure. By contrast, the NUREG-1150 analysis concludes that 26% of the core-damage sequences result in an intact containment, 4% in late containment failure, 13% in containment venting, and 57% in early containment failure or bypass.

Thus, accident progression and containment performance is seen to be substantially different in the WASH-1400 and NUREG-1150 analyses of Surry and Peach Bottom. This difference highlights the importance of developing realistic estimates of the contributions to risk as the basis for safety evaluation.

5.4 Severe Accident Source Terms

The substantial reductions in source terms between WASH-1400 and NUREG-1150 analyses for Surry are illustrated in Figures 1-4, which depict the frequency of release of iodine, cesium, strontium, and lanthanum to the atmosphere in excess of given amounts. The median, mean, 5th, and 95th percentile data from the NUREG-1150 analysis are included. However, only median data are available from WASH-1400. The shaded areas in the figures illustrate the reductions in median source terms between the two studies. The results include the effects of changes in mitigation features which have been added to the plants as well as the increase in estimate of the pressure capability of the containment. Specific observations from these PWR source-term data are:

- The NUREG-1150 median frequency of release of 10% or more of the iodine or cesium inventory in Surry is lower than that reported in WASH-1400 by a factor greater than ten, and the median frequency of release of magnitude similar to the PWR-2 release category, e.g., 70% iodine release, is insignificant (less than 10-8 per reactor year).

- The NUREG-1150 median frequency of release of 1% or more of the Surry core inventory of strontium is three orders of magnitude below the comparable WASH-1400 value. Mean and 95th percentile probabilities of releases of greater than 6% of the core inventory of strontium (the largest release reported in WASH-1400) are observed in NUREG-1150, but at very low frequencies.

- There is a reduction in the median probability of release of lanthanum (0.04% of the lanthanum core inventory) by three orders of magnitude compared with WASH-1400. Mean and 95th percentile releases greater than 0.5% (the largest lanthanum release reported in WASH-1400) are observed in NUREG-1150, but at very low frequencies.

An alternative way of looking at Figures 1-4 is to consider the fraction of inventory released for a given probability. For example, for a probability of 10^-3/r-y, the fraction of iodine released from an accident to Surry is reduced from 33% according to WASH-1400 to 8x10^-3% according to NUREG-1150; a 4000-fold reduction.
Frequency of Exceeding Iodine Release Fractions in NUREG-1150 and WASH-1400 Analyses of Surry

Iodine

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Figure 1
Frequency of Exceeding Cesium Release Fractions in NUREG-1150 and WASH-1400 Analyses of Surry

Cesium

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Figure 2
Frequency of Exceeding Strontium Release Fractions in NUREG-1150 and WASH-1400 Analyses of Surry

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.
Frequency of Exceeding Lanthanum Release Fractions in NUREG-1150 and WASH-1400 Analyses of Surry

Lanthanum

WASH-1400 PWR (Median)

Mean

5%

95%

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Frequency of Release ≥ R, Per Reactor Year

R, Fraction of Core Inventory Released To Environment

Figure 4
These frequencies or magnitudes are lower for three reasons. First is the reduction in the core-damage probability cited earlier (a factor of 2.6 for Surry). Second is the higher failure pressure ascribed to concrete containments (130 psig vs. 80 psig for Surry). Third is the greater retention of fission products within containment due to the recognition that iodine would combine with cesium as cesium iodide, which is soluble in water, rather than remaining as insoluble elemental iodine vapor as was assumed in WASH-1400. As stated earlier, the first factor is largely due to plant modifications. The latter two factors reflect a restatement of the WASH-1400 source term, due to factors of the type that prompted the restudy of the source term and risk analysis after the TMI accident.

A similar comparison of probabilities of exceeding specific fractions of core inventory is given in Figures 5-8 for the Peach Bottom plant.

The results in NUREG-1150 also substantially reduced the WASH-1400 values of source terms for Surry and Peach Bottom, as illustrated in Figures 9-12 and 13-16, respectively, which show the median probabilities that the release to the atmosphere of iodine, cesium, strontium, and lanthanum exceed specific fractions of the core inventory, given a core-damage accident. The shaded areas in the figures illustrate the reductions in median source terms between the two studies. Specific observations from these figures include:

- NUREG-1150 estimates a five-fold or greater reduction in the median probability that 10% or more of the iodine or cesium inventory would be released. For the largest release fraction stated in WASH-1400 (viz. 70%), the median release fraction in NUREG-1150 has become insignificant (less than 10^-6/ry on an absolute basis).

- NUREG-1150 estimates that the median probabilities of release of 1% or more of the strontium inventory from Surry and Peach Bottom would be lower than the WASH-1400 values by factors of 100 and 10, respectively. The reductions in the median release probabilities for lanthanum are comparable.

- An alternative way of looking at Figures 3 and 4 is to consider the fraction of inventory being released at a given probability. For example, for a conditional probability of 0.1 (which translates to an absolute probability of about 3x10^-6/ry), the fraction of iodine released from Surry has been reduced in NUREG-1150 to 3%, from about 70% in WASH-1400, about a 20-fold reduction. For higher probabilities, the reduction is much larger, and for lower probabilities it is smaller since there would always be some probability, albeit infinitesimal, of releasing the entire inventory. Similar observations can be made for other radionuclide groups and for the Peach Bottom plant.

5.5 Off-Site Consequences

Detailed comparison of off-site consequences reported in WASH-1400 and NUREG-1150 is not possible, because there are many differences between the two studies, such as: the use of the CRAC computer code in WASH-1400 and the MACCS code in NUREG-1150; the use of site-specific meteorological and population data
Frequency of Exceeding Iodine Release Fractions in NUREG-1150 and WASH-1400 Analyses of Peach Bottom

Iodine

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Figure 5
Frequency of Exceeding Cesium Release Fractions in
NUREG-1150 and WASH-1400 Analyses of Peach Bottom

Cesium

Frequency of Release ≥ R, per Reactor Year

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Figure 6
Frequency of Exceeding Strontium Release Fractions in NUREG-1150 and WASH-1400 Analyses of Peach Bottom

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

Figure 7
Frequency of Exceeding Lanthanum Release Fractions in NUREG-1150 and WASH-1400 Analyses of Peach Bottom

Lanthanum

WASH-1400 BWR (Median)

Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.

5% Median

95%

R, Fraction of Core Inventory Released To Environment

Figure 8
Frequency of Release of Iodine Conditional on Core Damage Frequency --
WASH-1400 and NUREG-1150 Analyses of Surry

![Graph showing frequency of iodine release vs. fraction of core inventory released to environment.

Figure 9]
Frequency of Release of Cesium Conditional on Core Damage Frequency -- WASH-1400 and NUREG-1150 Analyses of Surry

Figure 10
Frequency of Release of Strontium Conditional on Core Damage Frequency --
WASH-1400 and NUREG-1150 Analyses of Surry

Figure II
Frequency of Release of Lanthanum Conditional on Core Damage Frequency --
WASH-1400 and NUREG-1150 Analyses of Surry

Lanthanum

WASH-1400
PWR
(Median)

Median

Mean

Figure 12
Frequency of Release of Strontium Conditional on Core Damage Frequency --
WASH-1400 and NUREG-1150 Analyses of Peach Bottom

Figure 15
Frequency of Release of Lanthanum Conditional on Core Damage Frequency: WASH-1400 and NUREG-1150 Analyses of Peach Bottom BWR (Median) and WASH-1400 BWR (Median)
in NUREG-1150 whereas WASH-1400 used composite averaging from many sites; and different assumptions as to emergency evacuations. Different health physics coefficients were used. However, and generally speaking, the off-site consequences reported in NUREG-1150 are substantially lower than those reported in WASH-1400.

To facilitate a comparison between estimates of off-site consequences in WASH-1400 and NUREG-1150, we suggest that the final version of NUREG-1150 might include comparisons of estimated probabilities of exceeding whole-body or thyroid doses as a function of distance from the site, e.g., Figure I-11 and I-13 in NUREG-0396. This comparison removes the effect of differing population distributions, which was treated differently in WASH-1400 and NUREG-1150. Other helpful comparisons might use selected figures in NUREG/CR-1131. These data are available from calculations already completed, so no delay in issuance of the report should be caused by incorporating such comparisons.
6. RESPONSIVENESS TO THE KASTENBERG PANEL REVIEW

6.1 Introduction

During the period June 1987 to March 1988, a Peer Review Panel chaired by Professor William Kastenberg, University of California, Los Angeles, reviewed the entire breadth of the risk analyses documented in the first draft of NUREG-1150. Each member of the Kastenberg Peer Review Panel wrote an individual section of the report; the panel was not asked to provide a consensus opinion. The results of this peer review have been published in Reference 26 and can be summarized in six major criticisms and twenty-one specific comments. As part of our review, we discuss here the adequacy of the second draft of NUREG-1150 in meeting the criticisms and comments of this panel.

6.2 The Kastenberg Panel Review

The six general criticisms made in the first peer review are as follows:

- The draft and the supporting contractor documents were difficult to follow, uneven in their presentation, and sometimes inconsistent with one another. Many of the key technical assumptions and management decisions were either omitted from the text or difficult to find.

- The front-end analyses were dictated by an unreasonably short schedule, resulting in several shortcuts, potentially serious omissions, and lack of thorough quality assurance.

- There was an unevenness in the overall approach, as well as in the robustness of the results.

- There was disregard for technical rigor and/or state-of-the-art in many facets of both the probabilistic and mechanistic analyses that make up the PSA.

- Where new analytic tools were used, they were ill-documented, largely unvalidated, and used to excess with little benchmarking against prior knowledge or data.

- The expert polling process was seriously flawed.

The twenty-one specific comments encompassed the following categories:

- PSA Methodology and Core Damage Frequency
- PSA Methodology and Phenomena
- Containment Response, and
- Consequence Analysis and Value/Impact Assessment.
Appendix D of the second draft of NUREG-1150 includes the NRC Staff's response to the comments made by the Kastenberg Panel, as well as others, including comments by the Kouts Committee\(^{27}\) and the American Nuclear Society's Committee\(^{28}\). The response is grouped into seven major topics, and because the Staff dealt with a number of reviews, there is not a one-to-one correspondence with any one particular review. However, the Staff attempted to respond to each category of comment. Following is our assessment.

6.3 Adequacy of the Second Draft in Meeting the Comments

The first major criticism of the first draft involved the documentation: the volumes that make up NUREG-1150 and the supporting contractor documents. Extensive restructuring and rewriting have generated the present version of NUREG-1150. This new document is an improvement over the previous one in its completeness, scrutability, and presentation of results. The contractors' supporting documents are in various stages of completion; hence, we cannot comment on them.

The second major criticism involved the front-end or systems analysis. The original Draft relied on previous analyses supported by the NRC Staff, performed in an attempt to construct a so-called "Smart PSA". Since the first draft was issued, considerable effort was devoted to making the front-end analysis more robust. These efforts included a strengthened internal review process (quality assurance) for the fault and event trees, including the analysis of common-cause failures (CCF) and the human reliability analyses (HRA). On the other hand, the NRC Staff recognizes that the state-of-the-art with respect to CCF and HRA is imperfect and that further improvements in the PSA can be made in these crucial areas as new models and methods develop. We noted in Section 4.8 that in the front end the human reliability issues, especially common-cause failures and human reliability analysis, have not been treated as a top-level issue in the elicitation of expert opinion.

The third major criticism focused on the unevenness of the approach and of the results. The NRC Staff believes that, with the exception of the Zion Plant, the NUREG-1150 methods have now been applied consistently, and that different levels of detail are necessary because plant-specific issues dictate where and when additional consideration need be given. The Zion PSA was performed by Brookhaven National Laboratory (BNL) and the others by Sandia National Laboratory. (BNL's approach was based on an Industry PSA and a Staff/Contractor review which was updated to reflect recent design and operational changes.) We concur with the NRC Staff's assessment of consistency insofar as it applies to the accident frequency analysis or "front end". There is still a level of inconsistency in the "back end"; i.e., the evaluation of the Accident Progression Event Trees (APET's). This is, in part, because a) the state of knowledge as regards severe accident phenomena in BWR's versus PWR's is different, b) the use of expert elicitation for severe accident issues was not the same for all plants, and c) there was a large uncertainty in recovery actions by operators after core melt was estimated to begin.

The fourth major criticism involved a disregard for technical rigor and/or state-of-the-art in many facets of both the probabilistic and mechanistic aspects of the PSA. This comment referred to diverse matters, including the use of
probability and statistics, the treatment of common-cause failures, and the use
of unreviewed and undocumented computer codes. The latter point was also the
thrust of the fifth criticism. The NRC Staff and its contractors have attempted
to address this issue as far as possible. Within the budget and time con-
straints, efforts were made to validate and/or benchmark some of the new computer
codes. The XSOR computer codes (for the source terms) fall in this category.
Several other codes used in the analysis are being examined, such as the MACCS
computer code (for the consequence analyses). There are still difficulties with
the "averaging" process regarding the results of the expert opinion. The display
of the results of the uncertainty analysis also have been improved.

The last, and probably most controversial and yet important, issue is the
expert opinion elicitation process for dealing with uncertainty. A number of
significant modifications were made to improve the process itself between the
first and second drafts. Yet several problems persist, two of which are
inherent, and will always persist.

- Although more diverse groups of individuals were chosen for the various
  expert panels, there is always the question of "who is an expert on a
given issue?"

- Even if an elicitation process were "perfect", the result could not be
  better than the state of knowledge itself.

Hence, there is always the question of the adequacy of the knowledge base
for expert opinion, with respect to several crucial phenomenological issues.

There are still significant questions regarding the manner in which the
judgments of the experts were aggregated or averaged, and then used in uncer-
tainty propagation. This issue was particularly acute in instances when the
experts had widely divergent views (e.g., the development of seismic hazard
curves, the BWR liner melt-through problem, and the issue of direct containment
heating). If one expert gives an opinion that is an order of magnitude larger
than those of the remaining experts, that opinion will dominate risk, especially
with regard to the mean. The matter is discussed in some detail in Section 4.11.

6.4 Concluding Comments

The NRC Staff/Contractors have addressed the issues, criticisms, and com-
ments made by the Kastenberg Peer Review Panel within the time, resource, and
knowledge constraints placed upon them. Those noted above that are not addressed
adequately are due, to a large degree, to lack of knowledge and the ability to
deal adequately with this limitation, which must be considered when using the
results of NUREG-1150 in the regulatory process.
7. CONCLUSIONS AND RECOMMENDATIONS

7.1 General

All critiques of work done elsewhere have a tendency to dwell at greater length on the weakness of the work rather than on the strengths. The present review is no exception. If we have seemed to concentrate on shortcomings perceived in NUREG-1150, the reader should not draw a conclusion that we regard the study to be fundamentally flawed. It is not. As we state in the conclusions below, we consider the present draft of NUREG-1150 to be a major step forward in risk assessment in several areas, deserving recognition as the best current update of WASH-1400. We found points where we believe improvements could have been made, and where there are shortcomings, and we have recommendations for some alterations to the draft and for future work. Some of the major conclusions and recommendations are summarized below. Others are provided in the comments sections of the text.

We do not believe that issuance of the final version of NUREG-1150 should be held up for further research or analysis. Some of our recommendations propose relatively simple changes in the exposition, or the clarification of points by including results already available from the analysis but not brought out by the text. We believe that these minor improvements could easily be made for the final version of report.

7.2 Conclusions

Our conclusions are ordered, with the overall supportive views stated first, and the shortfalls following. Several of these latter are not so much problems of NUREG-1150 as they are of the current status of PSA, which requires more development in some areas.

- 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 beyond WASH-1400, mainly in the United States but also in other countries. In most respects, it represents the state-of-the-art in 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 PSA's for individual plants. Like its predecessor, WASH-1400, it should help to show the path for future PSA developments for some time to come.

Even so, the study is not perfect, and we turn now to some of the blemishes.

The most vulnerable parts of the methodology used in the study are the treatment of human reliability and the estimation of parameters by expert opinion elicitation, both of which require more research.

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 had 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 might have been preferable.

The expert opinion procedure is complex, time-consuming, and expensive. Therefore, the full scope of this methodology may have very limited future application. It is unlikely that a procedure of this magnitude will be repeated for several years, although expert elicitation on single or narrow issues may be practical. It should be remembered, however, that throughout the study analysts had to decide how to use technical information of all kinds. This form of "expert judgement" is necessary in all PSA's.

If phenomenological models of processes are not provided and directly used, the dependence of the results of the accident progression analysis on governing physical phenomena is hidden. The generality of the structure of event 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 there is little information 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 little information.

The failure modes and characteristics of containments, as well as the conditional probabilities for typical failures of the containment structure were largely determined from expert opinion. This indicates that there are limitations to the state-of-the-art ability to calculate the containment loads directly, taking into account all the relevant phenomena that would prevail during a loss of coolant accident, especially during the ex-vessel phase.
The methods used to analyze human reliability and human error do not reflect the range of variability encountered in HRA models. Systematic error may have been introduced through the exclusive use of selected methods. Though the treatment of effects of human reliability and human error presents problems, these are mainly rooted in the state-of-the-art, and the analysis may be as good as could have been done at the time.

Several kinds of accident initiators were not included in the study. Among these are pressure vessel failure, main steam line failures in PWR's, errors of commission, and sequences beginning from shutdown or low power. They should have been included, or reasons for their omission given in more depth.

Of the five plants analyzed in NUREG-1150, only two (Surry and Peach Bottom) have been analyzed for external events. The results indicate that the contributions to risk of external events must be considered, for at least some plants. The lack of analysis of external events for the other three plants is a deficiency of the report.

Certain potentially important effects are not explicitly or fully covered: events starting from low power and shutdown modes, sabotage, and aging, which may not be fully covered by current inspection and maintenance programs. Electrical control and actuation circuits were not explicitly covered in the analysis of common-cause failure. Although it is recognized that the impact of "safety culture" and management quality cannot be factored into the PSA at the present time, it is important to bear in mind such impacts as overall decisions are made on plant safety.

The Committee believes that fires are such important initiators of possible accidents, that the analysis should have been extended to all five plants treated by NUREG-1150.

The accident progression event tree for each plant consisted of about 100 branches, each having multiple outcomes or branches. It seems to us that this level of detail exceeded understanding of the phenomena involved, implying greater insight into the processes assumed to be taking place than was justified.

It would have been valuable if the theoretical HRA's of the ATWS sequences had been tested against real events, such as those cited above, 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 is more appropriate and consistent with the use of expert panels for a number of back-end issues of similar importance, as measured in their contribution to overall risk.

The uncertainties in the consequence analyses for each sequence were not propagated. The uncertainties shown in the risk profiles for each
reactor and each consequence are due to the uncertainty in the Level 1 and Level 2 aspects of the PSA only.

As a neutral observation, we point out that a strategy for reducing the concern over the uncertainty bounds in risk estimates is to eliminate from designs and operating practices those features that lead to the wide uncertainty bounds. Where these options are impractical, the desired level of risk reduction might be achieved instead by improvements in systems indirectly related to the uncertain risk issue under evaluation, or in appropriate severe accident management measures. In fact, the "best" risk management strategy may involve an appropriate mix of some or all of these approaches.

7.3 Recommendations

- The NRC staff should now move toward early publication of NUREG-1150 in final form. We have suggested some changes or additions assuming that these can be made speedily without delaying the report. If appreciable delay would be necessary, our view is that later, separate publication should be called for, without change to NUREG-1150. Timely publication is important to provide guidance to the individual plant evaluations (IPE's) being prepared by the utilities. As for the particular plants analyzed in NUREG-1150, their IPE's will be a vehicle to complete the seismic and fire hazard assessments in sufficient depth and with accurate descriptions of the plants as they are presently configured.

- As a more general point, plant-specific analysis of external events should be included in PSA's. We recommend that the NRC issue additional guidance on the treatment of external events in the IPE program. In particular, such guidance seems warranted for the types of seismic hazard curves to be used in different parts of the United States.

- Research in seismic modeling is warranted, with the object of improving the basic model to predict attenuation and ground motion and for developing a consensus on 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.

- Special attention should be paid in the NRC's research program to further development of Human Reliability Analysis and to calibrating methods used to analyze human reliability, to facilitate comparison between plants and comparisons with safety goals.

- Large uncertainty contributions associated with some phenomena indicate the need for further research. We particularly single out the thermal-hydraulic phenomena associated with accident management strategies, such as depressurization and water addition to the primary system of a PWR, and improvement of understanding of the ways in which the primary system boundary may fail during high pressure sequences in PWR's. Another important issue deserving increased attention is the
assessment of threats to the integrity of the containment and the identification of means to ensure its integrity in case of a core melt accident with failure of the pressure vessel.

- Because plant-specific information is becoming increasingly important in PSA, such information should be collected and placed on file in a future program.

- While the expert opinion process was carefully structured and professionally guided, there were still a number of issues where the technical information available to guide the expert panels was limited. For this reason, the Committee urges caution and intelligence in the use of these results by others outside the scope of NUREG-1150. The results of sampling of expert opinion are well documented, and one should be fully aware of their limitations before using them.

- Likewise, the Committee recommends considerable caution in the use of the results obtained with the approximate XSOR codes without confirmation by more detailed calculations.

- The following are changes that are recommended be made to the final version of NUREG-1150, that we believe can be done without further analysis.

  * Where recovery actions were important, they should be discussed and their scope defined in the summary report in Chapter 2 of NUREG-1150. Their effects should be quantified in Chapters 3-7, e.g. for Surry: core-damage frequency without recovery actions $8.2 \times 10^{-4}$/ry, with recovery actions, $3.5 \times 10^{-5}$/ry (from Table 4.10-5, NUREG/CR 4550, Rev. 1, vol. 3).

  * The contributions to the core melt probabilities of the unavailabilities of safety system functions should be displayed among the results of the analysis of frequency of core damage.

  * Because of the approximate nature of the XSOR codes, the final draft of NUREG-1150 should note the need for a more exacting analysis of risk significant accident sequences, such as the interfacing systems LOCA's and steam generator tube rupture accidents for PWR's, and station blackout and ATWS sequences for BWR's. The more detailed analysis should be published in a supplement to NUREG-1150. This analysis should concentrate on best estimate modeling, and the results compared with the source terms published in NUREG-1150.

  * Some issues requiring the input of expert opinion were addressed by the project staff rather than the expert panels. It should be clearly indicated which were so treated and the values of the parameters used in the study; some indication should be made of the importance of the parameter to the values of risk.

- NUREG-1150 represents an enormous investment of resources which should be put to good use, not simply be made available as a resource
document. NUREG-1150, along with the other risk assessments and recent work in the field of severe accident analysis, should be used to: (1) close out as many open issues as is reasonable, and (2) help prioritize the limited resources to focus research on the remaining safety-related issues. A definitive program to use NUREG-1150 and its supporting documents should be developed and implemented.
8. RESPONSE TO COMMISSION QUESTIONS

In the Charter of the Committee, reproduced in the appendix, the Nuclear Regulatory Commission posed some specific questions for which a response was particularly desired. In many places in the preceding text we discussed areas covered by these questions. At this point, we repeat the questions and assemble specific answers to them.

Does NUREG-1150 adequately reflect the comments made by the Kastenberg review group (NUREG/CR-5113), given the uncertainties in data and models?

As stated in Section 6.4, the NRC Staff/Contractors have addressed the issues, criticisms, and comments made by the Kastenberg peer review panel within the time, resource, and knowledge constraints placed upon them. Those issues noted above that are not addressed adequately are due, to a large degree, to limitation in the state of knowledge and the ability to deal adequately with this limitation, which must be considered when using the results of NUREG-1150 in the regulatory process itself.

Have the uncertainties associated with both front- and back-end analyses been adequately described in NUREG-1150? Is the use of expert elicitation appropriate in developing these uncertainties?

This question is discussed more fully in Section 4.12. There we concluded that in general, NUREG-1150 represents state-of-the-art methodology in uncertainty analysis, where uncertainty estimates were made. These estimates, concerning the Level 1 and Level 2 analyses, were mainly the result of elicitation of expert opinion. Formally eliciting expert opinions to develop the uncertainties is appropriate; however, caution is required, since this process has not been widely used for safety issues, and some parts, especially the selection of experts, are critical to the process and may cause controversy if not properly done. However, the state-of-the-art does not yet provide a complete view of the uncertainty in the results. At this point, we believe that the major factors still to be settled are the treatment of human error, including errors of commission, and the uncertainty in consequences as derived in the Level 3 analysis. Lesser questions as to uncertainty analysis are found throughout our Report.

As discussed in Section 4.9, it is important to bear in mind that management quality introduces uncertainty because it is not reflected in the results of PSA. Since we doubt that it can be quantitatively factored into PSA at present or in the near future, that element of uncertainty must be assessed by the management evaluations being pursued by NRC and INPO.

To what extent should probabilistic risk assessment focus on the low-probability tails of the accident frequency distributions? Is there an appropriate cutoff in terms of reportable accident frequencies?
This question is discussed in detail in Section 4.10. We believe that a realistic cutoff in both frequency of severe accidents and their resultant risk is warranted, and should be encouraged in all PSA's. The preceding considerations indicate that event families and plant damage states with frequencies below about $10^{-7}$/yr should be neglected in probabilistic risk analyses. In addition, a health risk in the range from $10^{-2}$ to $10^{-3}$ times the normal occurrence rate also seems reasonable. For curves of accident magnitude vs frequency, a cutoff of from $10^{-7}$/ry to $10^{-8}$/ry in the frequency seems warranted.

Do the methods, models, and data used in NUREG-1150 suggest they could be used as standardized methods for preparing probabilistic risk assessments?

Some of the features used in the NUREG-1150 program will have definite value in conducting future probabilistic risk assessments. The generic aspects of the data base can be mined for specific application. Some of the computer codes may find their way into more common use as they are tested out. Some of the results of expert opinion elicitation may be used more generally. The elicitation process itself was very involved and required a substantial investment of time and resources. It is unlikely that this particular aspect of the NUREG-1150 methodology will be extensively repeated in the near future.

Does the committee have any recommendations to make on the need for further improvement in probabilistic risk assessment methods?

The Committee's conclusions and recommendations are given in Chapter 7.
9. REFERENCES


APPENDIX

UNITED STATES NUCLEAR REGULATORY COMMISSION

CHARTER

SPECIAL COMMITTEE TO REVIEW THE SEVERE ACCIDENT RISK REPORT

1. The committee's official designation: Special Committee to Review the Severe Accident Risks Report

2. The committee's objectives and the scope of its activity:

The committee shall report to and advise the Director of the Office of Nuclear Regulatory Research and through him the Commission, on the adequacy of the methods, insights, analyses and conclusions set forth in the April 1989 draft of NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants. The Commission requires this information to ensure that the proposed regulatory uses of the information set forth in NUREG-1150 are appropriate. In particular, the committee shall provide its views on the following specific questions:

1. Does NUREG-1150 adequately reflect the comments made by the Kastenberg review group (NUREG/CR-5113) given the uncertainties in data and models?

2. Have the uncertainties associated with both front and back end analyses been adequately described in NUREG-1150? Is the use of expert elicitation appropriate in developing these uncertainties?

3. To what extent should probabilistic risk assessment focus on the low probability tails of the accident frequency distributions? Is there an appropriate cut-off in terms of reportable accident frequencies?

4. Do the methods, models and data used in NUREG-1150 suggest they could be used as standardized methods for preparing probabilistic risk assessments?

5. Does the committee have any recommendations to make on the need for further improvement in probabilistic risk assessment methods?

3. The period of time necessary for the committee to carry out its purposes:

The Special Committee to Review the Severe Accident Risks Report is expected to complete its work within twelve months of the filing of its charter.
4. The agency or official to whom the committee reports:

The committee will report to the Director of the Office of Nuclear Regulatory Research and, as appropriate, through the Director to the Commission.

5. The agency responsible for providing the necessary support for the committee:

The U.S. Nuclear Regulatory Commission. The NRC's Office of Nuclear Regulatory Research will provide the necessary administrative support through a contract with the Brookhaven National Laboratory.

6. A description of the duties for which the committee is responsible:

The committee shall provide the Director of the Office of Nuclear Regulatory Research with a written consensus report of its views and recommendations regarding the adequacy of NUREG-1150 focusing on the objectives described in paragraph 2 above.

7. The estimated annual operating costs in dollars and FTE staff years:

The estimated operating costs for this committee will be approximately $300,000 and 0.5 FTE.

8. The estimated number and frequency of committee meetings:

It is estimated that the committee will hold four or five meetings.

9. The committee's termination date:

The committee will terminate one year from the date this charter is filed, subject to renewal by the Commission.

10. The date this charter is filed: July 7, 1989.

John C. Hoyle
Advisory Committee Management Officer
U.S. Nuclear Regulatory Commission
### BIBLIOGRAPHIC DATA SHEET

**Title and Subtitle:** Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150)

**Author(s):**

**Performing Organization:**
Office of Nuclear Regulatory Research  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555

**Sponsoring Organization:**
Same as above

**Abstract:**
In April 1989, the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) published a draft report "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150. This report updated, extended and improved upon the information presented in the 1974 "Reactor Safety Study," WASH-1400. Because the information in NUREG-1150 will play a significant role in implementing the NRC's Severe Accident Policy, its quality and credibility are of critical importance. Accordingly, the Commission requested that the RES conduct a peer review of NUREG-1150 to ensure that the methods, safety insights and conclusions presented are appropriate and adequately reflect the current state of knowledge with respect to reactor safety.

To this end, RES formed a special committee in June of 1989 under the provisions of the Federal Advisory Committee Act. The Committee, composed of a group of recognized national and international experts in nuclear reactor safety, was charged with preparing a report reflecting their review of NUREG-1150 with respect to the adequacy of the methods, data, analysis and conclusions it set forth. The report which precedes reflects the results of this peer review.

**Keywords/Descriptors:**
- NUREG-1150
- severe accidents
- PRA, risk assessment
- peer review

**Availability Statement:**
Unlimited

**Security Classification:**
[This Report] Unclassified

**Price:**