POLICY ISSUE
(Information)

August 21, 1990 SECY-90-296

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: FINAL REPORTS OF THE NUREG-1150 REVIEW COMMITTEES: STAFF COMPLETION OF FINAL NUREG-1150

Purpose: To transmit the final reports of the NUREG-1150 Peer Review Committee, the American Nuclear Society's Special Committee on NUREG-1150, and inform the Commission of staff plans to complete and issue a final version of NUREG-1150.

Discussion: In June 1989, the Commission established a committee to review the second draft of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." This committee was established under the provisions of the Federal Advisory Committee Act and was chaired by Dr. Herbert Kouts. Its primary effort was directed to developing answers to five questions relating to the adequacy of the second draft's incorporation and resolution of comments on the first draft, the appropriateness of low-frequency cutoffs of PRA information, and future directions in PRA research.

The Commission was briefed by Dr. Kouts and members of the Committee on June 20, 1990. At that time, a draft version of the Committee's report was also provided to the Commission. The Committee has now completed its final report, which is provided as Enclosure 1. There have been no substantive changes to this final report from the draft version provided to the Commission.

For the past several years, a special committee of the American Nuclear Society (ANS) has also been reviewing NUREG-1150 (both first and second drafts). This committee, chaired by Dr. Leo LeSage of Argonne National Laboratory, has also completed its report; it is provided as Enclosure 2.

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

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Both the Kouts and ANS Committees recommended that NUREG-1150 be issued in final form as soon as possible. The staff has made an initial review of the comments of both committees and assessed changes needed before publication of the final version of NUREG-1150. We believe that the principal changes to be made should include the following:

- o Accounting for changes to the Zion plant committed to by Commonwealth Edison Company to reduce the frequency of certain important accident sequences identified in NUREG-1150;

- o Update of the perspectives chapters in NUREG-1150 (Chapters 8 to 13) to reflect better the detailed results of NUREG-1150 and those contained in the numerous underlying contractor reports; and

- o Update of Appendix C (which provides supplemental technical detail on key issues) to provide information on certain important issues not provided in the second draft.

By a staff requirements memorandum dated May 12, 1989, the staff is to submit a final version of NUREG-1150 for Commission approval. Given the nature of the peer review comments, the staff plans to complete this final version and transmit it to the Commission by October 31, 1990.

James M. Taylor
Executive Director
for Operations

Enclosures:
As stated

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NUREG-1420

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

Date Published: August 1990

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E. H. Adolf Birkhofer, Gesellschaft fur Reaktorsicherheit Forschungsgelande, FRG
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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 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 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 adeqyuately 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. 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.

We must 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(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(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(5). 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(6) indicate probabilities of rupture typically in the range of 10^{-6}/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.

<u>Comment:</u>

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 elicitated 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. MORE DETAILED COMMENTS

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:

	Mean Conditional Probability of Containment Failure Modes		
	Surry	Sequoyah	Zion
No Containment Failure	81%	66%	74%
Late Containment Failure*	6%	21%	24%
Early Containment Failure*	1%	7%	1%
Containment Bypass	12%	6%	1%

(*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 3.4×10^{-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 5×10^{-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 initiators, 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 situation. 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 basemat 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[15,18], 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 <u>higher</u> 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.

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* 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 ($\geq 2x10^{-4}$/yr, depending on age). Associated cutoff values in the range of $5x10^{-7}$/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 $5x10^{-2}$/Sv*, the related committed annual risk of death from cancer induced by natural radiation is about $1x10^{-4}$/yr. A lower limit to the geographical variation of that risk would be well above $2x10^{-5}$ Sv/yr; in some parts of the world this value is as high as $5x10^{-2}$ Sv/yr. Thus, modification of the risk by an amount below $2x10^{-5}$/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 $2x10^{-6}$ 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 ($\sim 2x10^{-3}$/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.

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* 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^{-4}" or "of the order of 10^{-4}", 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 adoption 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 95th 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^{-5}/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 6×10^{-5}/ry in WASH-1400 to 2.3×10^{-5}/ry in NUREG-1150 (a factor of 2.6) and for Peach Bottom from 2.9×10^{-5}/ry to 1.9×10^{-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 95^{th} 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 m[illegible] are given in NUREG-1150 for the individual accident scenarios. Howeve[illegible] [illegible]centage contributions of the individual scenarios to the total mean CDF [illegible] compared by the Committee, with the results as follows:

- In the [illegible]H-1400 Surry analysis, 72% of the CDF was associated with LOCA[illegible] [illegible]ontainment bypass events, and 28% with transients. The [illegible] observed in NUREG-1150, wherein 23% of the CDF is associated [illegible] and containment bypass, and 77% with transients and station [illegible]

- [illegible] containment bypass events account for only 8% of the CDF for Su[illegible] 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 [illegible]e-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, 5^{th}, and 95^{th} 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 95^{th} 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 95^{th} 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^{-5}/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.

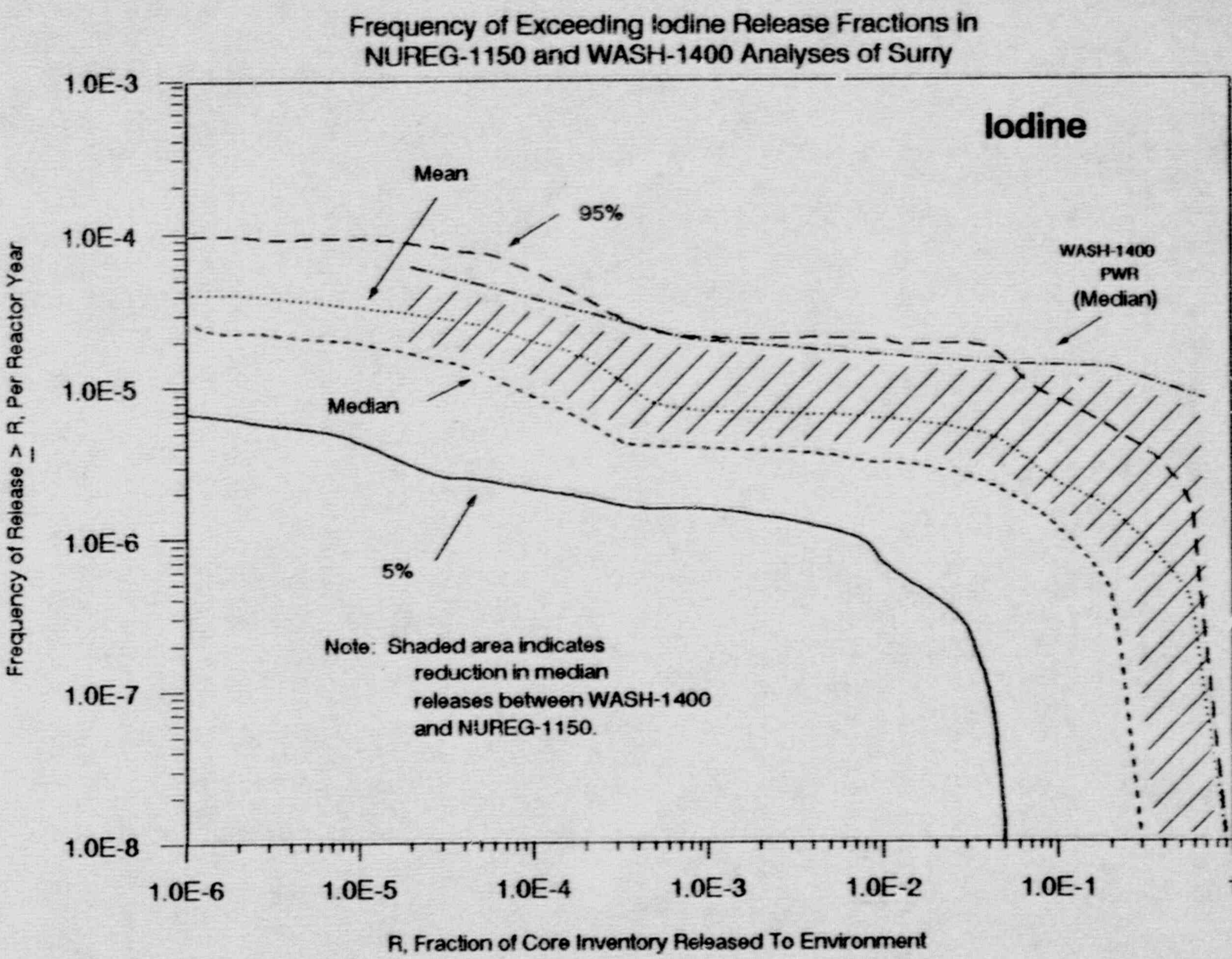

Frequency of Exceeding Iodine Release Fractions in
NUREG-1150 and WASH-1400 Analyses of Surry
Iodine
Mean
95%
WASH-1400
PWR
(Median)
Median
5%
Note: Shaded area indicates
reduction in median
releases between WASH-1400
and NUREG-1150.
Frequency of Release ≥ R, Per Reactor Year
1.0E-3
1.0E-4
1.0E-5
1.0E-6
1.0E-7
1.0E-8
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1
R, Fraction of Core Inventory Released To Environment

Figure 1

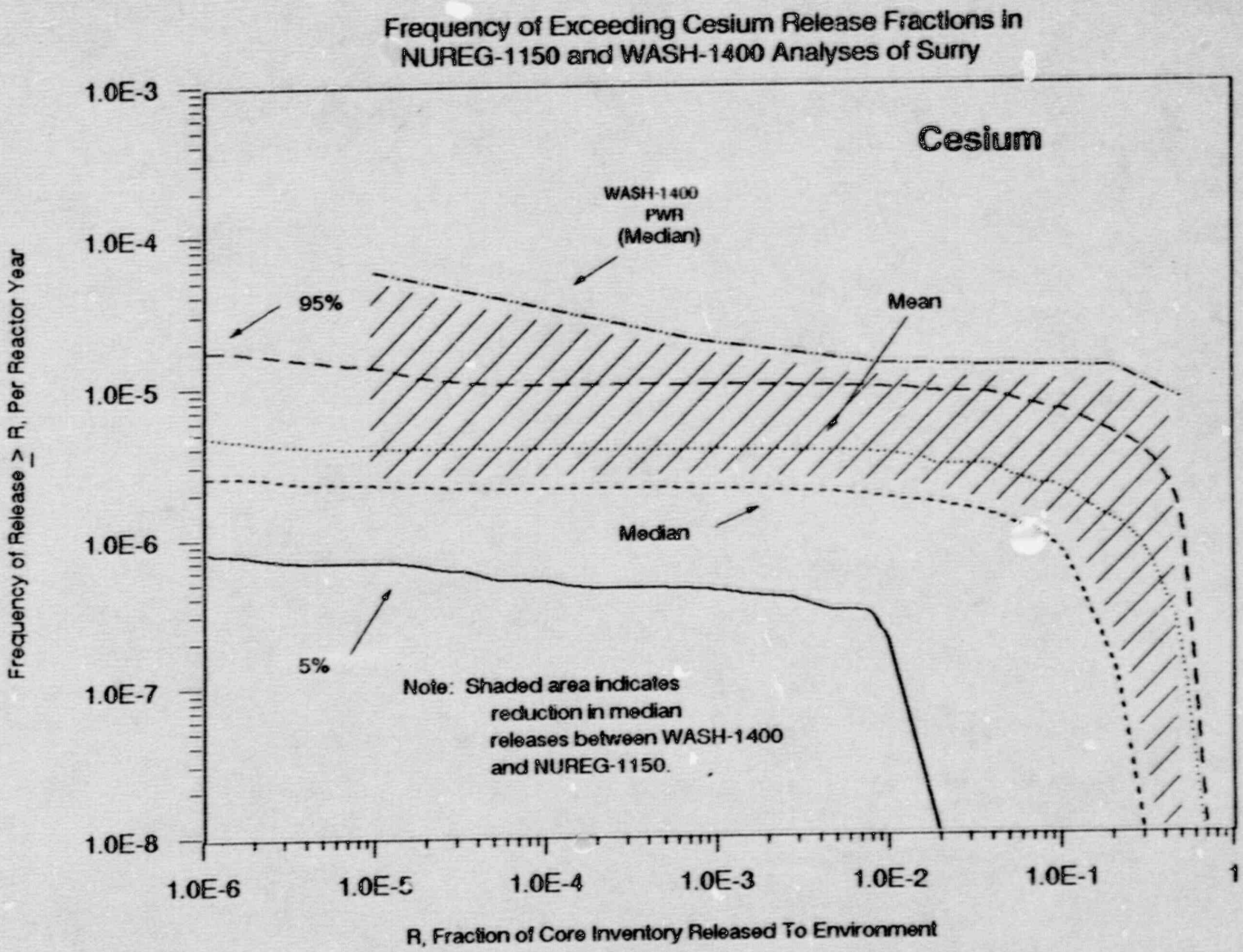

Frequency of Exceeding Cesium Release Fractions in
NUREG-1150 and WASH-1400 Analyses of Surry
Cesium
WASH-1400
PWR
(Median)
95%
Mean
Median
5%
Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.
Frequency of Release ≥ R, Per Reactor Year
1.0E-3
1.0E-4
1.0E-5
1.0E-6
1.0E-7
1.0E-8
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1
R, Fraction of Core Inventory Released To Environment

Figure 2

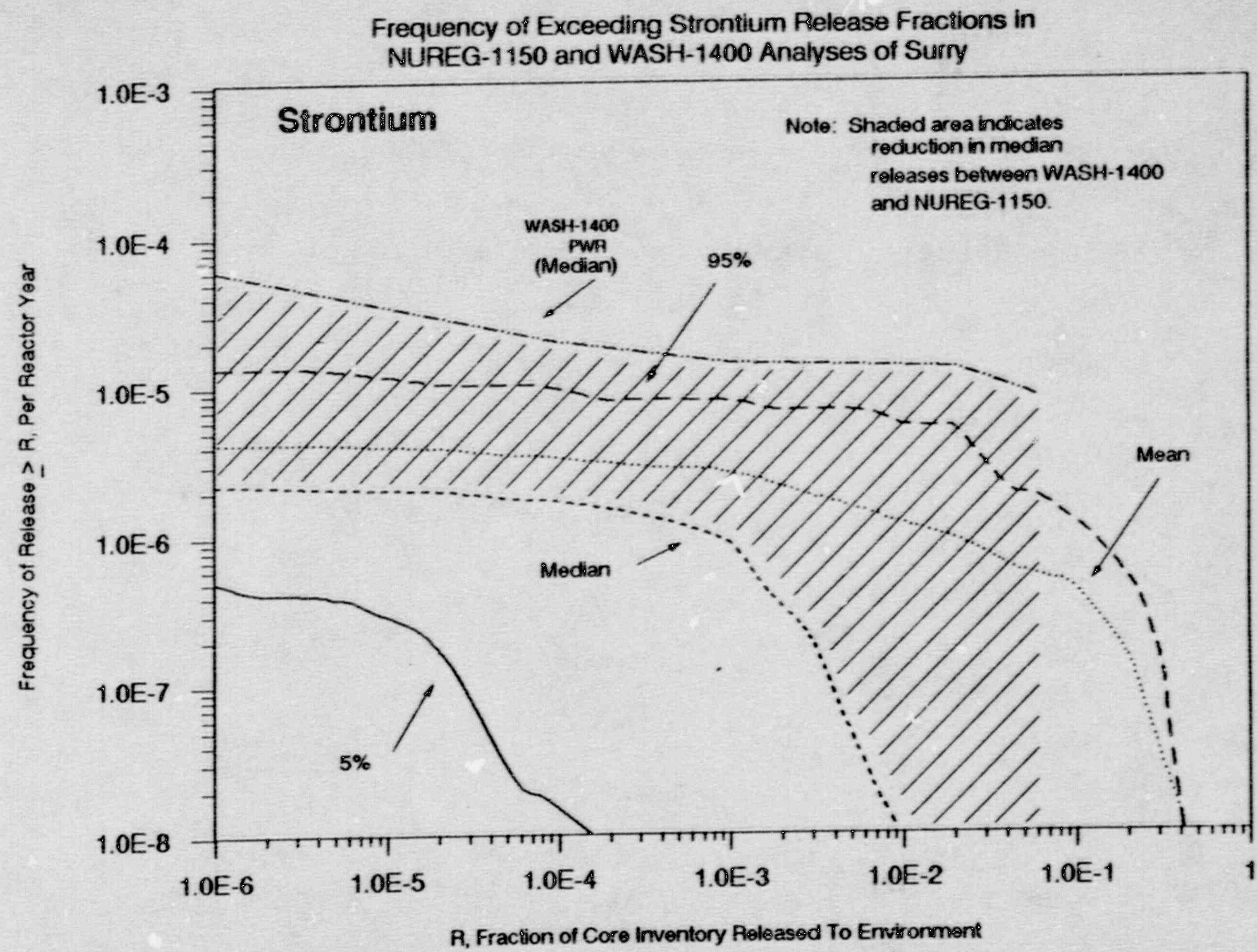
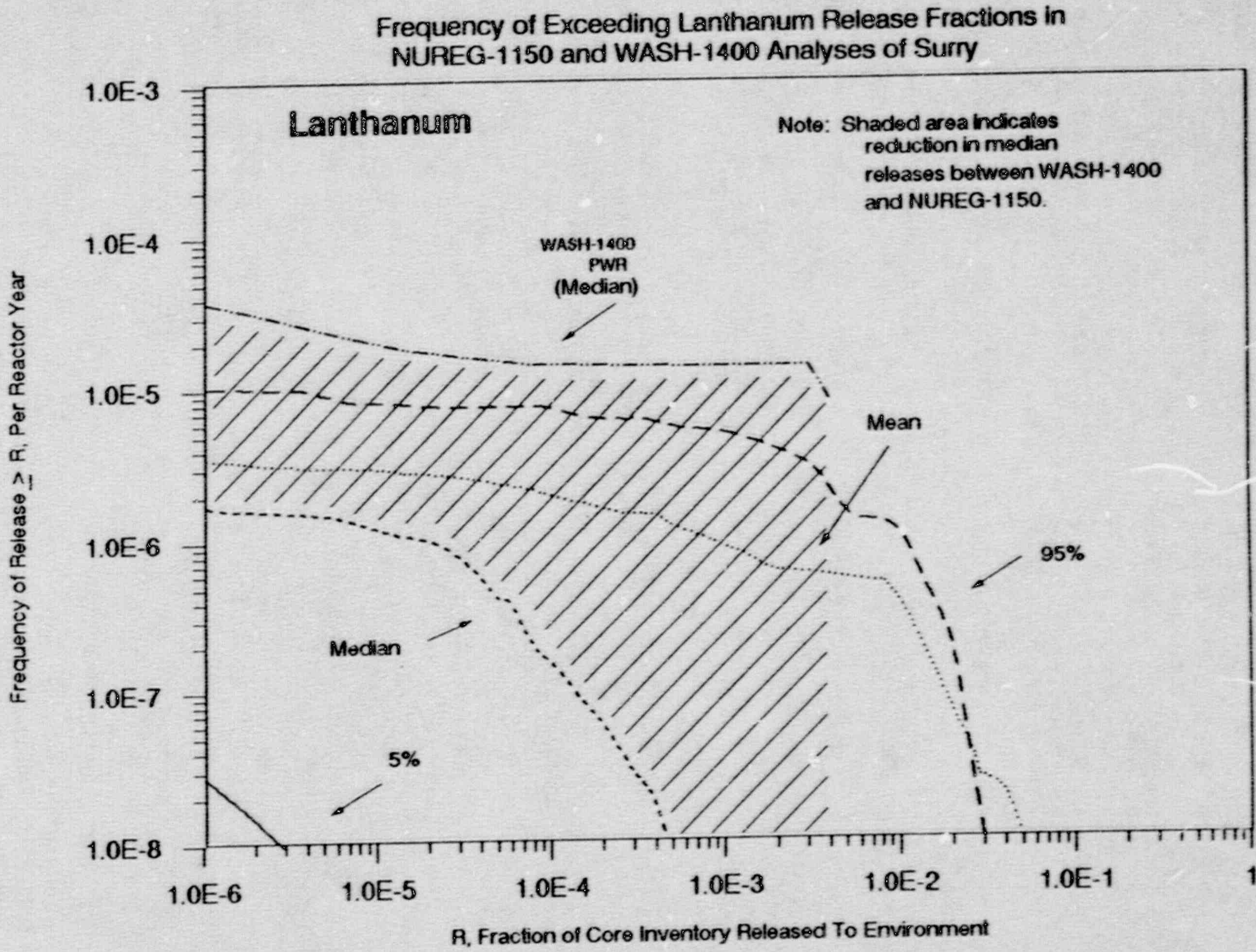

Frequency of Exceeding Strontium Release Fractions in
NUREG-1150 and WASH-1400 Analyses of Surry
Strontium
Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.
WASH-1400 PWR (Median)
95%
Mean
Median
5%
1.0E-3
1.0E-4
1.0E-5
1.0E-6
1.0E-7
1.0E-8
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1
Frequency of Release ≥ R, Per Reactor Year
R, Fraction of Core Inventory Released To Environment

Figure 3



Frequency of Exceeding Lanthanum Release Fractions in
NUREG-1150 and WASH-1400 Analyses of Surry
Lanthanum
Note: Shaded area indicates reduction in median releases between WASH-1400 and NUREG-1150.
WASH-1400
PWR
(Median)
Mean
95%
Median
5%
1.0E-3
1.0E-4
1.0E-5
1.0E-6
1.0E-7
1.0E-8
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1
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, <u>given a core-damage accident</u>. The shaded areas in the figures illustrate the reductions in median source terms between the two studies. Specific observations from these figures include:

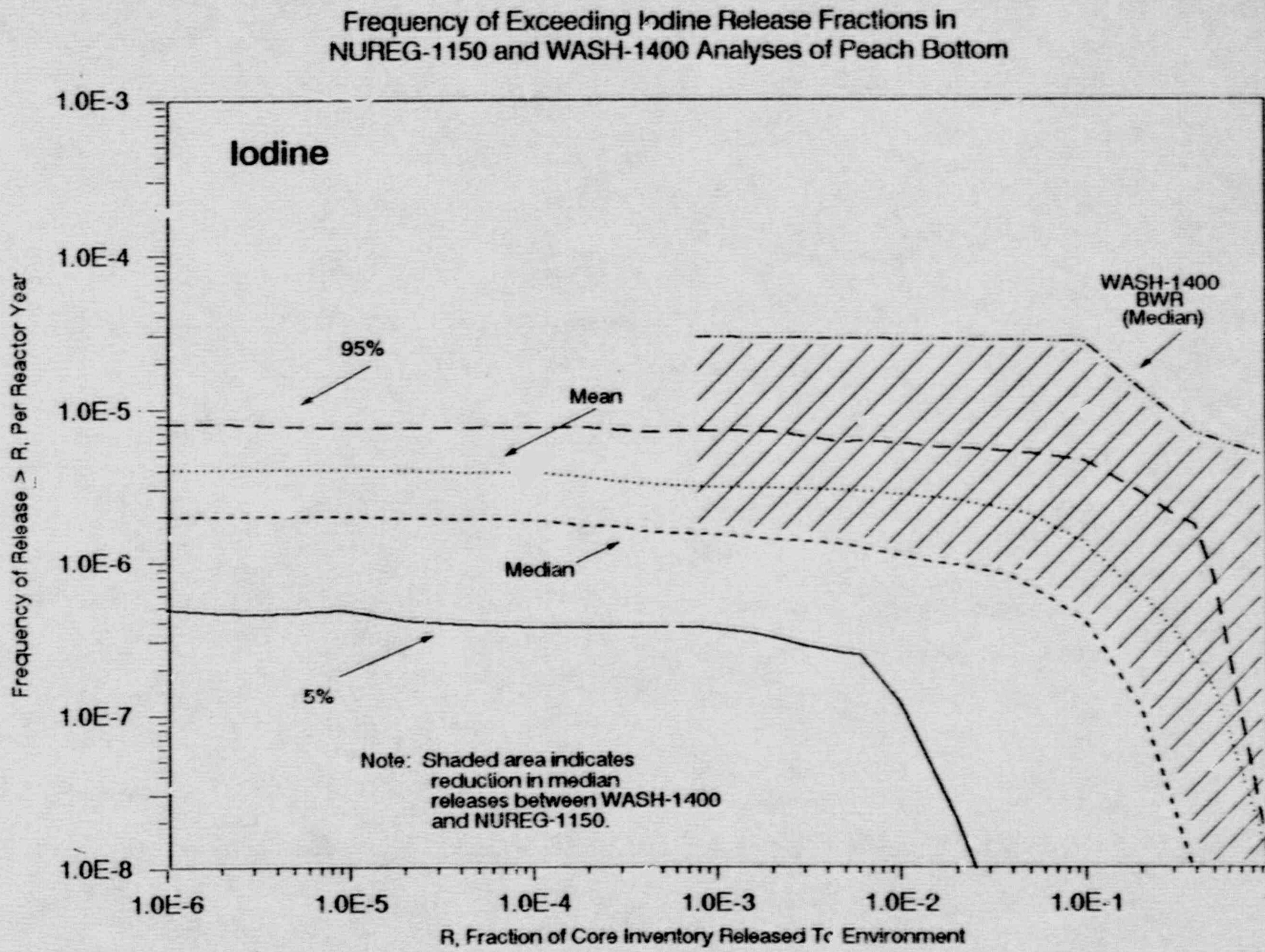
- NUREG-1150 estimates a five-fold or greater reduction in the median probability that 10% or more of the iodine or cesium inventory would be released. For the largest release fraction stated in WASH-1400 (viz. 70%), the median release fraction in NUREG-1150 has become insignificant (less than 10^{-8}/ry 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 $3\text{x}10^{-6}$/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
1.0E-3
1.0E-4
1.0E-5
1.0E-6
1.0E-7
1.0E-8
Frequency of Release > R, Per Reactor Year
WASH-1400
BWR
(Median)
95%
Mean
Median
5%
Note: Shaded area indicates
reduction in median
releases between WASH-1400
and NUREG-1150.
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1
R, Fraction of Core Inventory Released To Environment

Figure 5

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

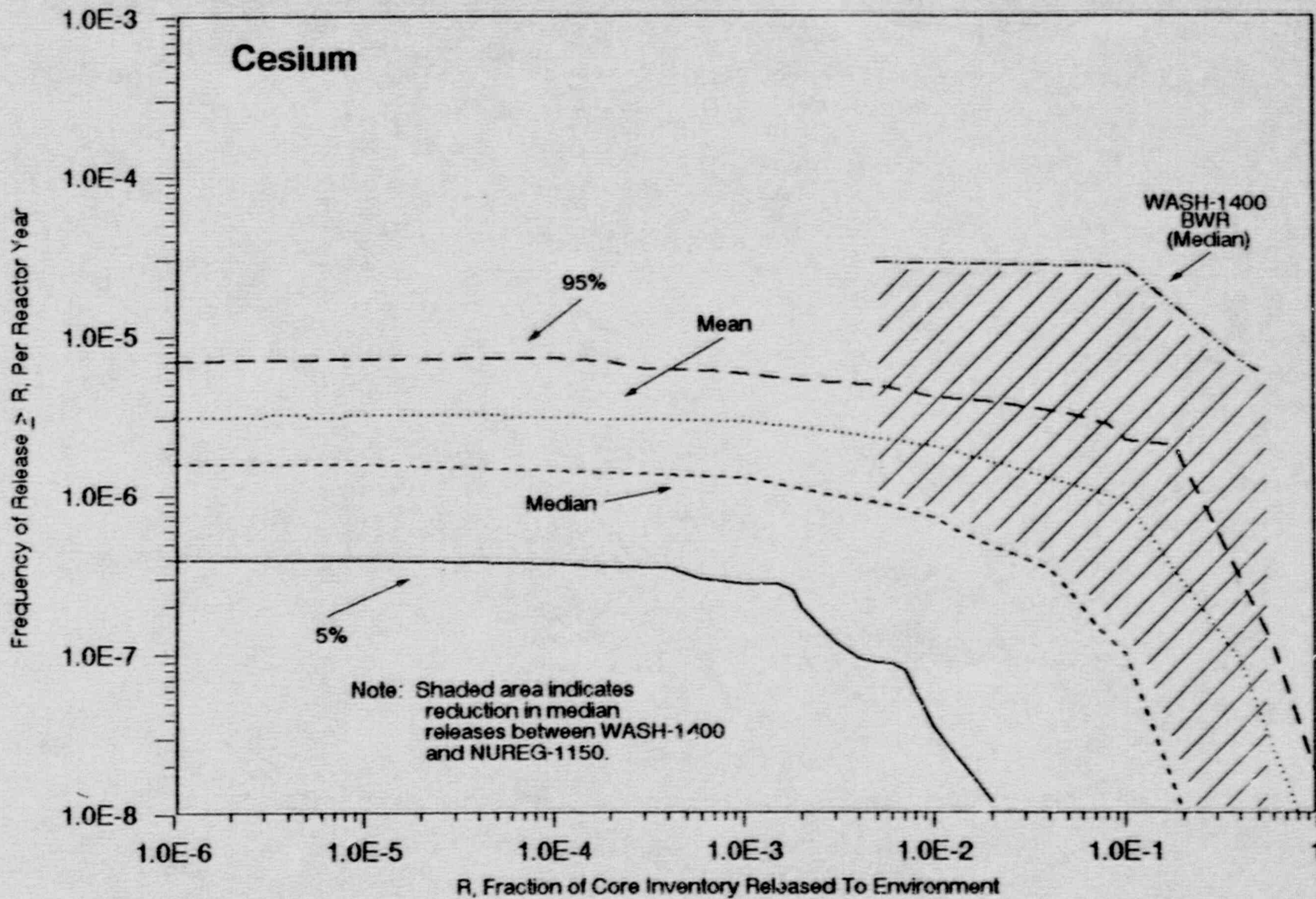


Figure 6

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

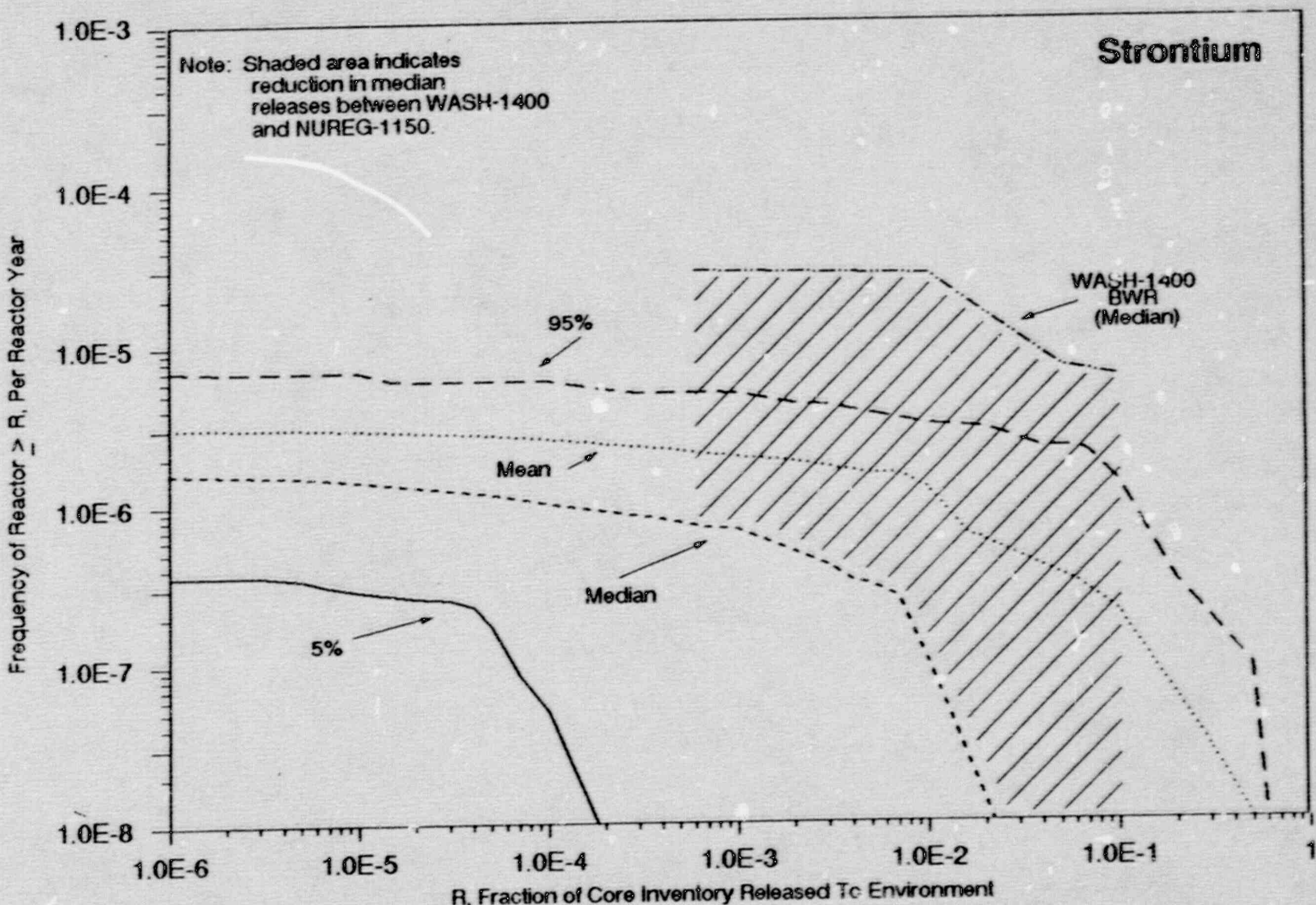


Figure 7

Frequency of Exceeding Lanthanum Release Fractions In
NUREG-1150 and WASH-1400 Analyses of Peach Bottom

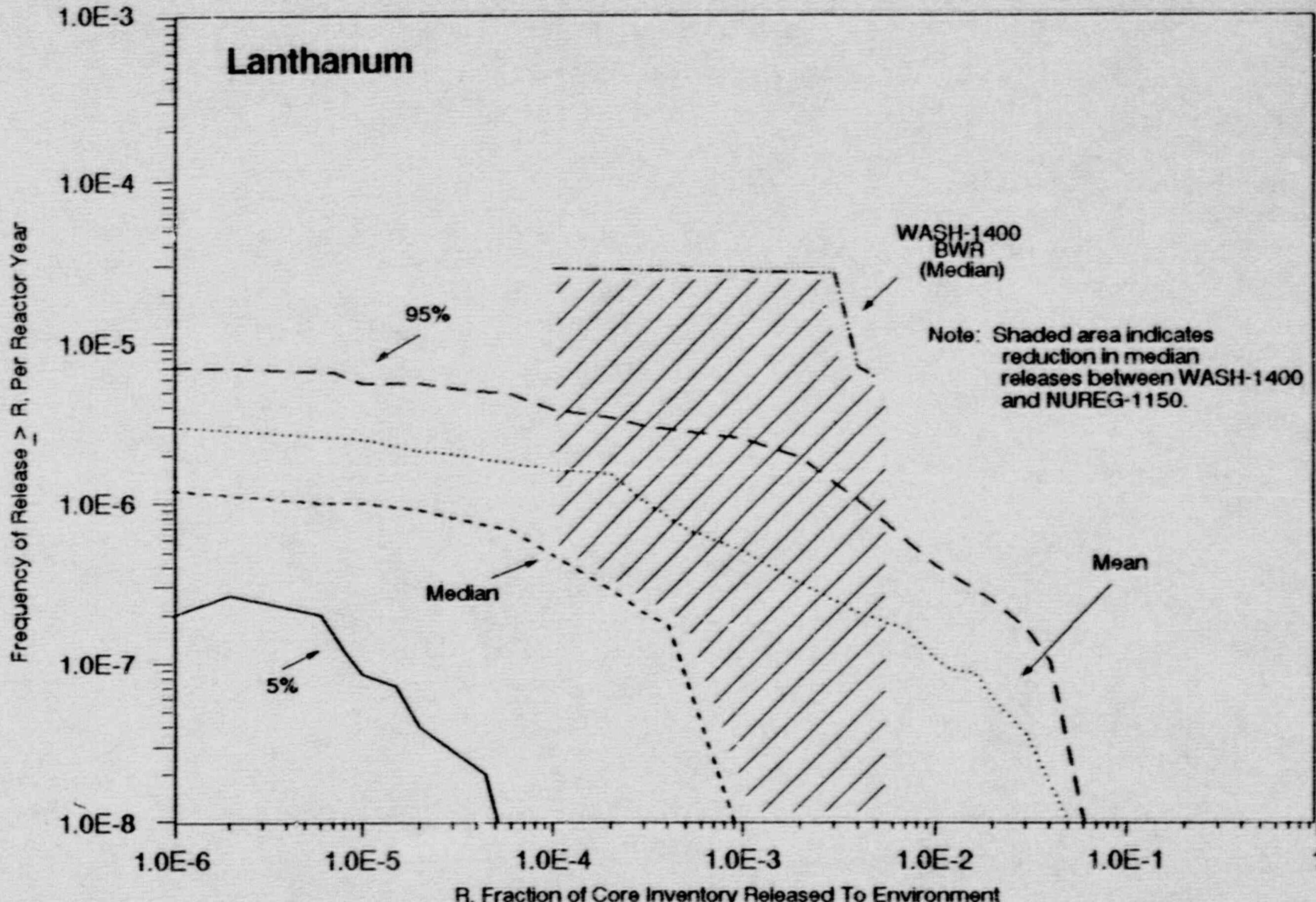


Figure 8

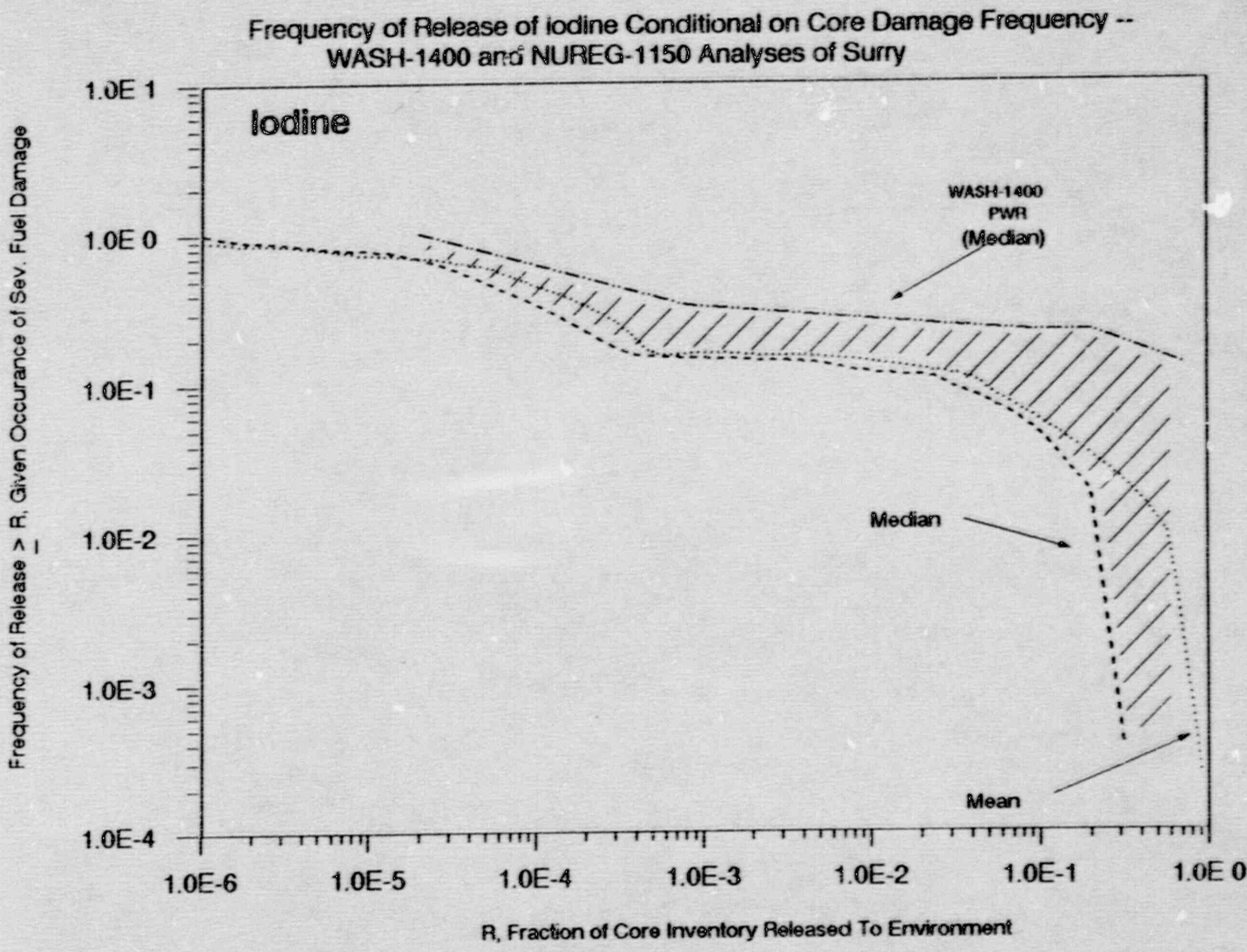

Frequency of Release of Iodine Conditional on Core Damage Frequency --
WASH-1400 and NUREG-1150 Analyses of Surry
Iodine
WASH-1400
PWR
(Median)
Median
Mean
1.0E 1
1.0E 0
1.0E-1
1.0E-2
1.0E-3
1.0E-4
1.0E-6
1.0E-5
1.0E-4
1.0E-3
1.0E-2
1.0E-1
1.0E 0
Frequency of Release > R, Given Occurance of Sev. Fuel Damage
R, 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

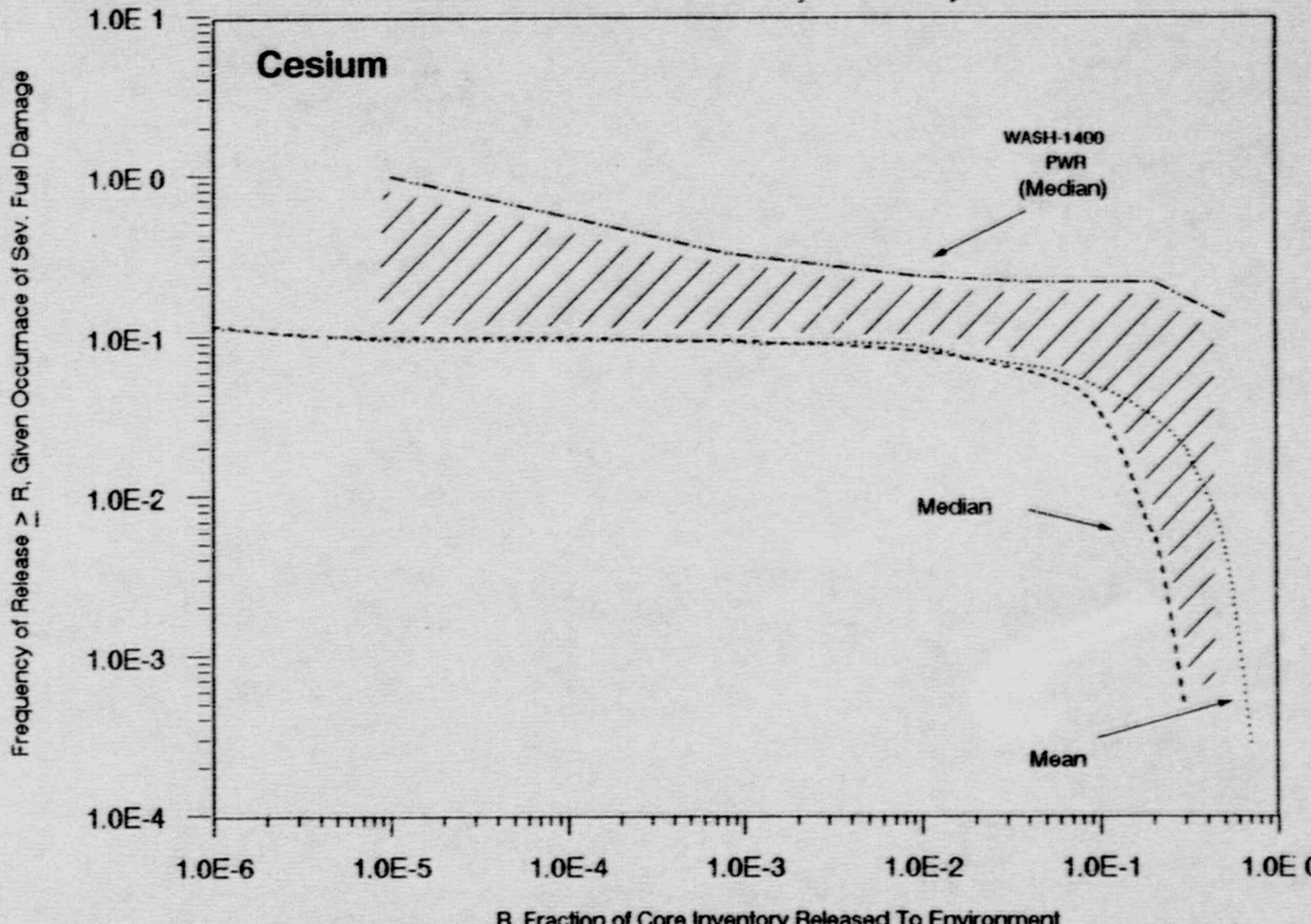


Figure 10

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

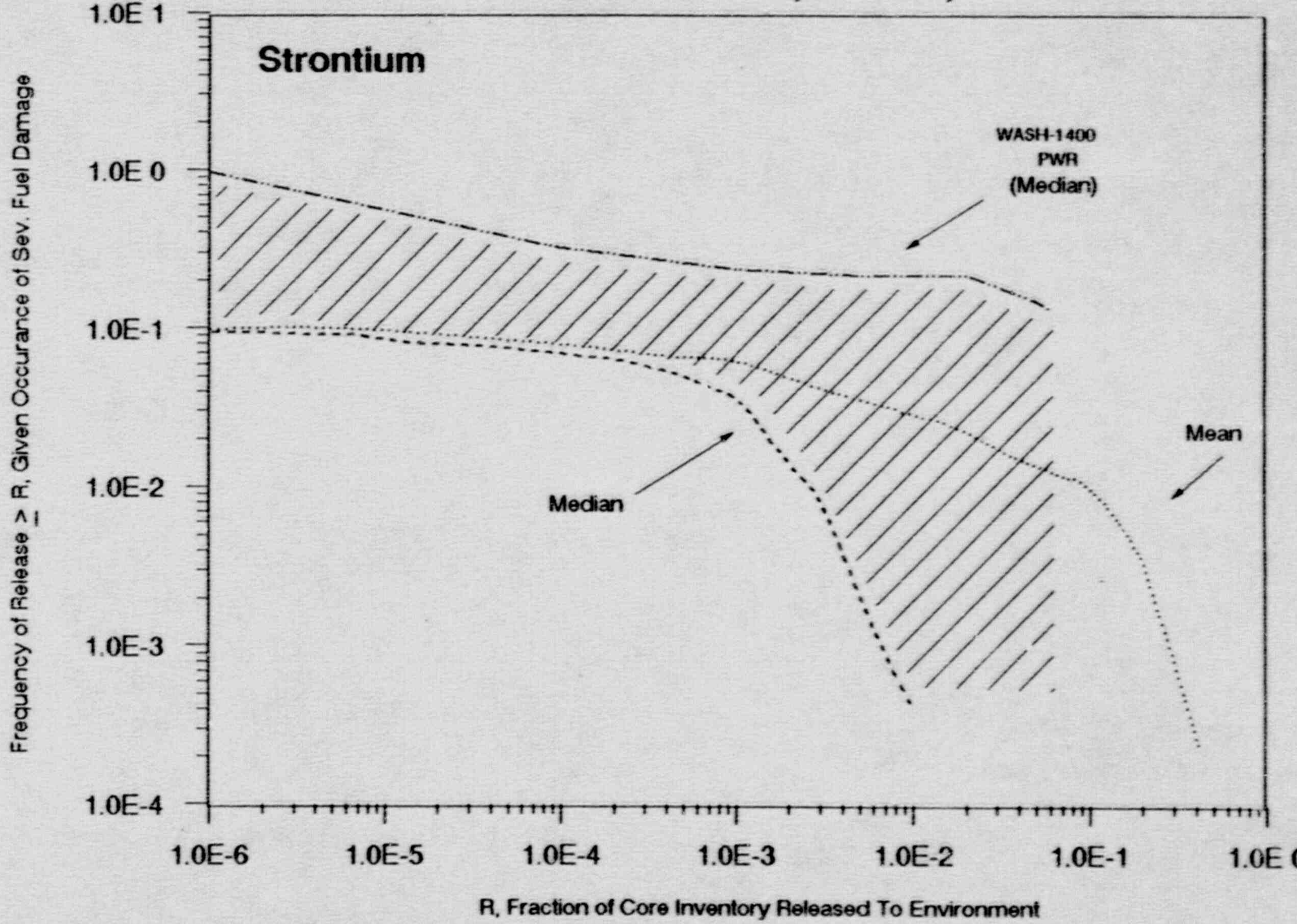


Figure 11

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

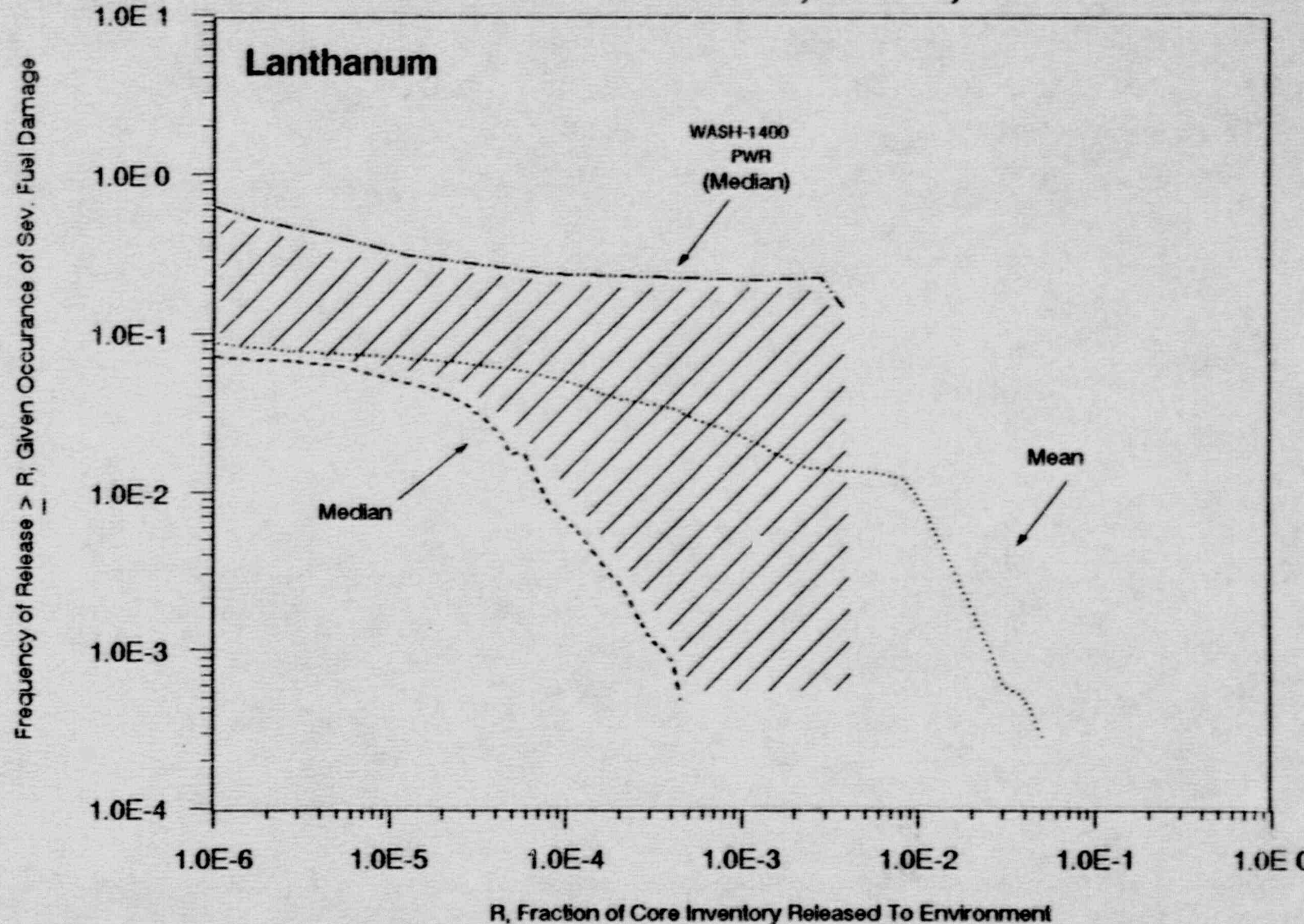


Figure 12

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

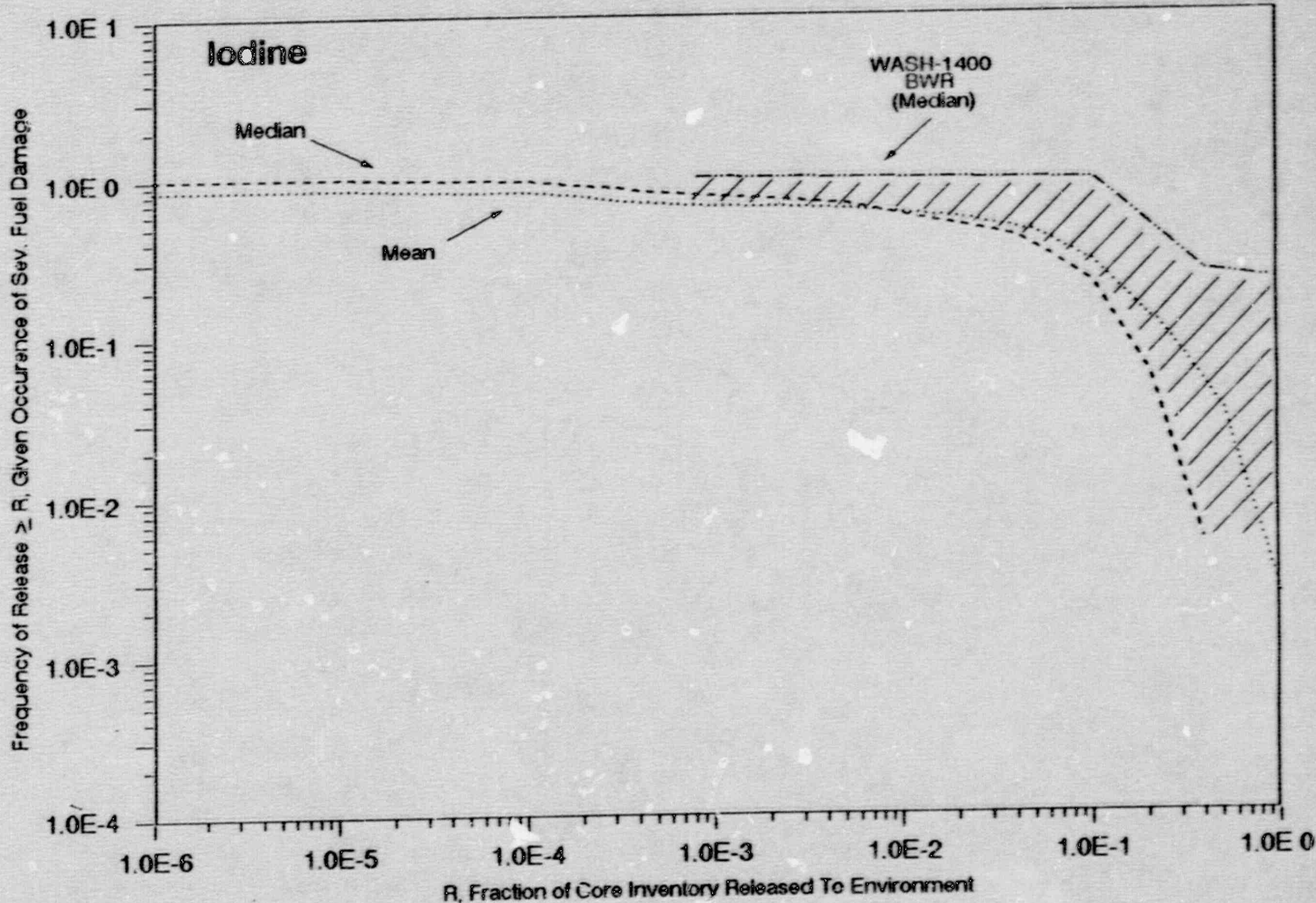


Figure 13

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

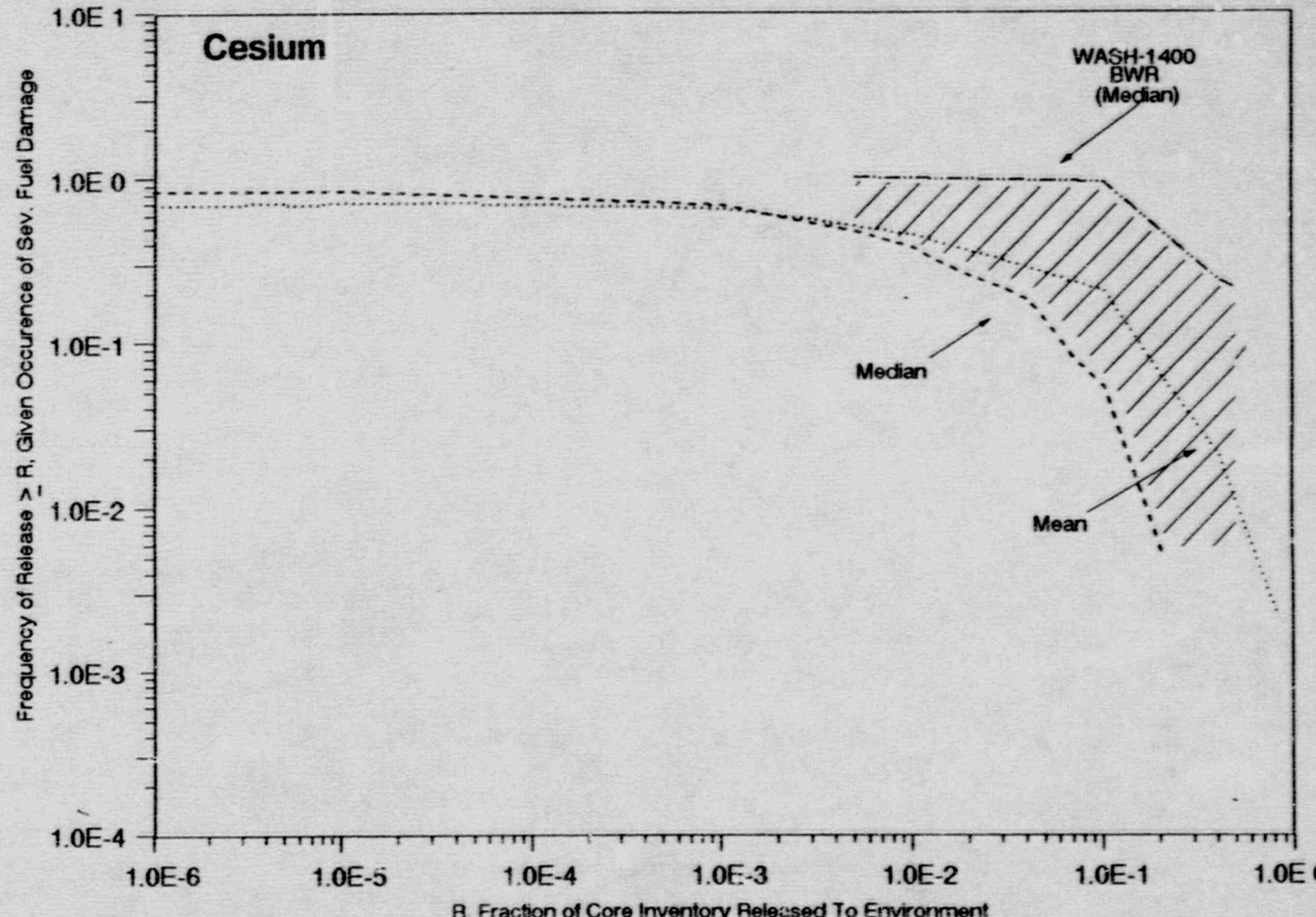


Figure 14

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

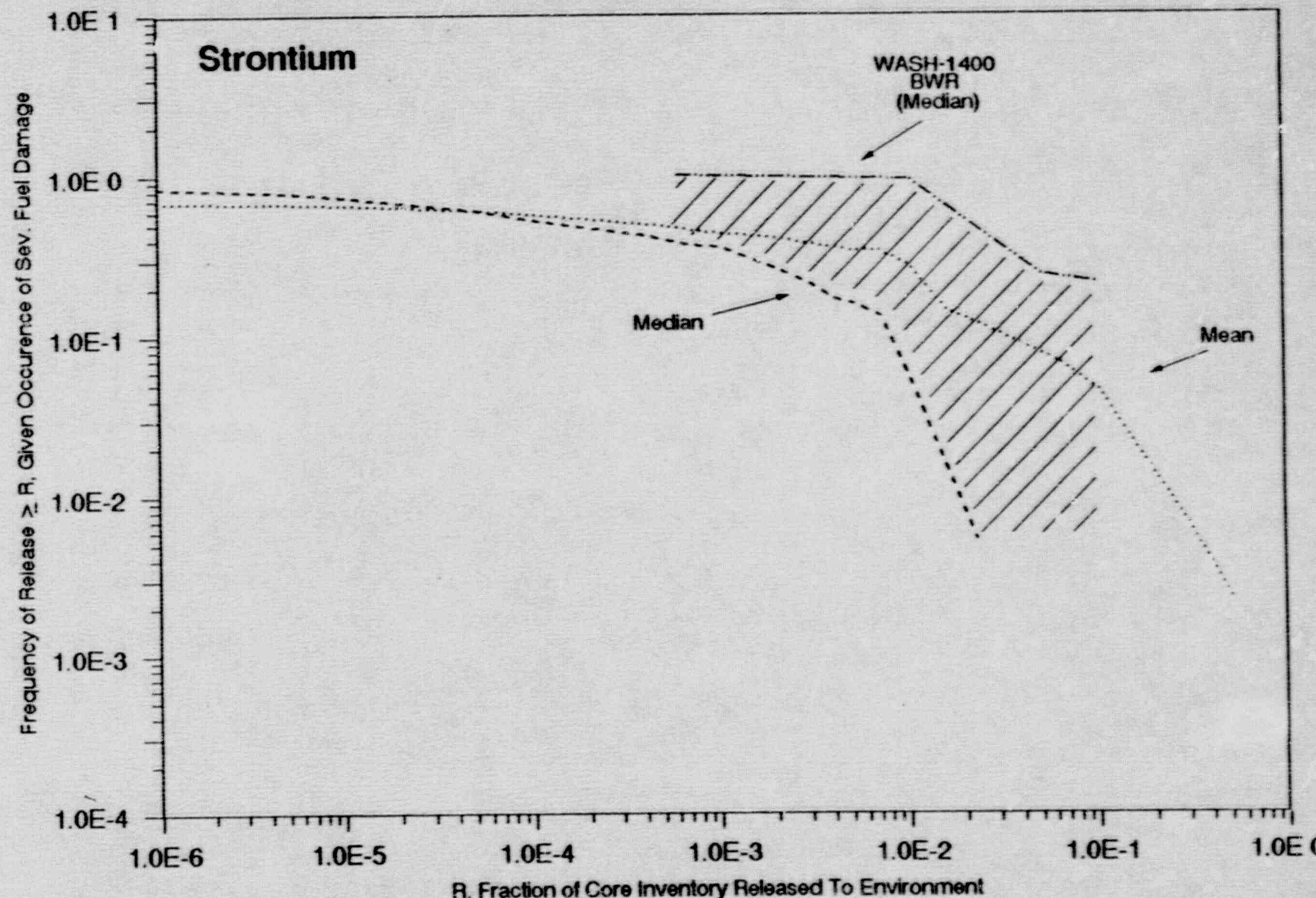


Figure 15

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

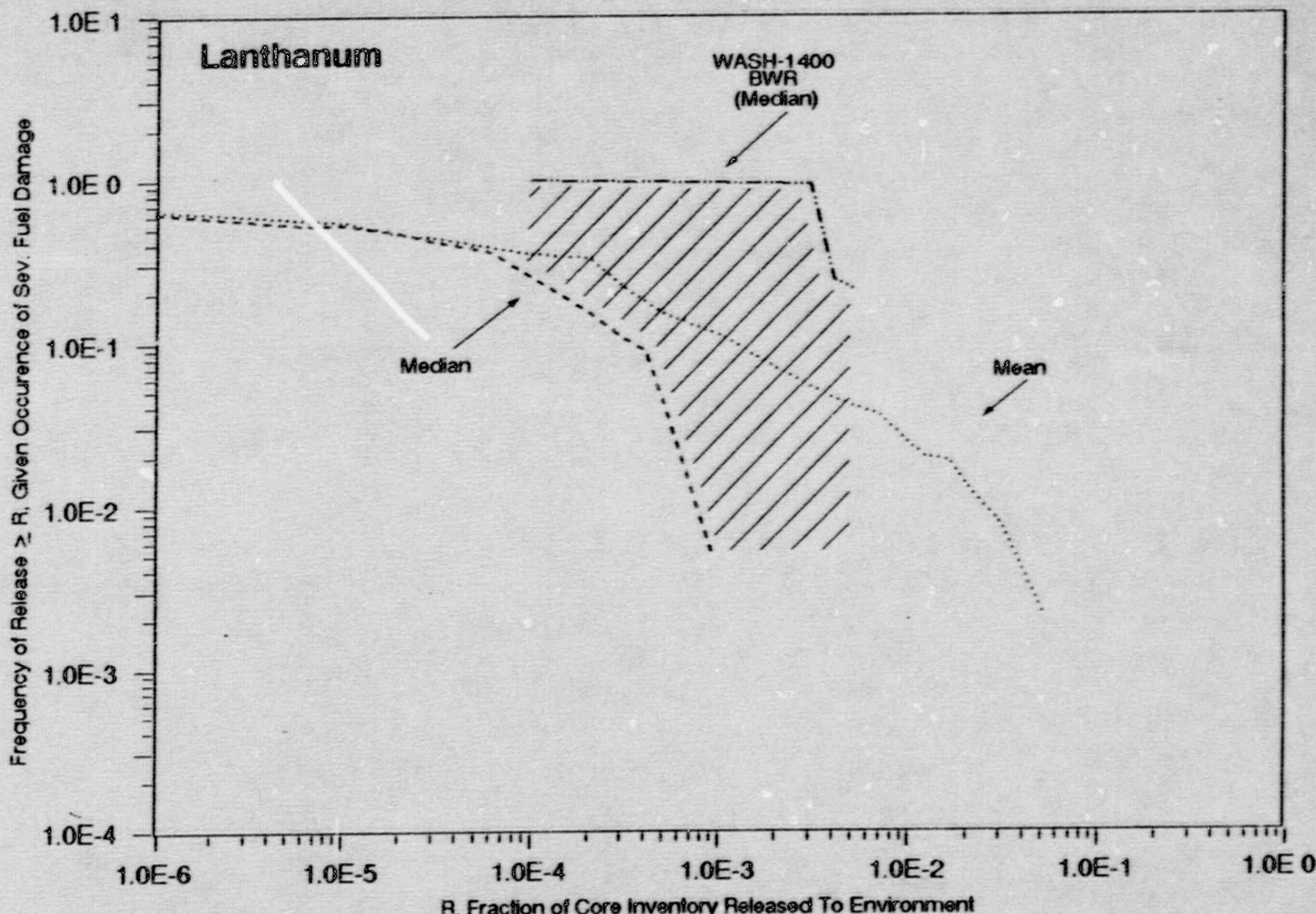


Figure 16

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 constraints, 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 uncertainty 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 comments 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 phenomona; 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 of 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 8.2×10^{-4}/ry, with recovery actions, 3.5×10^{-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. ANSWERS 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.

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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

John C. Hoyle
Advisory Committee Management Officer
U.S. Nuclear Regulatory Commission

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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11. ABSTRACT (200 words or less)

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.

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SUBMITTAL DRAFT

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FOREWORD

In February 1987 the Nuclear Regulatory Commission (NRC) published and distributed for public comment the first draft of NUREG-1150, a report which documented the results of its major risk assessment project encompassing five nuclear power plants in the United States. This report was, in effect, a successor to the Reactor Safety Study, WASH-1400; however, it represented a significantly expanded and updated effort relative to the earlier study. Many members of the American Nuclear Society (ANS) felt that the Society should express its views on this document considering its potential for influencing public opinion and the regulatory process. In response, the ANS President appointed a Special Committee in the Fall of 1987 to follow and comment upon the progress of NUREG-1150. Committee members were drawn from industry, universities and national laboratories, and represented a range of the Society's technical divisions. The Committee's membership also reflects the international nature of the Society with members from Canada and Switzerland and a corresponding member from Belgium.[1] The Committee's assignment from the President included developing a coordinated understanding and technical consensus on the document, making technical comments to the NRC, and reporting its findings to the ANS membership.

The Committee has maintained an active schedule since its inception, reviewing NUREG-1150 itself and many of its underlying documents, and has had numerous meetings with representatives of the NRC and personnel of the organizations that the NRC has contracted to perform the analyses which undergird NUREG-1150. In April 1988 the Committee issued an initial report, a short summary of findings and recommendations, directed primarily at the NRC, which at the time was supervising a major modification of the document. The second draft of NUREG-1150 was issued in June 1989, again for comment. The revised document was almost totally changed in its structure and text, and a large portion of the analyses whose results it reports were redone. The revised version reflected many of the comments made in the ANS Committee's initial report. The Special

[1]See list of Committee Members at the end of this Section.

Committee has been retained in its review capacity and, in this report, turns its attention to the revised version of NUREG-1150.

The review of the NUREG-1150 documents and process has not been an easy task. It has been made much less difficult, however, by the opportunities afforded to meet with staff members of the NRC and its contractors, particularly personnel of the Sandia National Laboratories. These frank, in-depth and often lengthy discussions were an immense help to the Committee as they provided access to the process, which otherwise could only have been obtained by reading the multitude of underlying documents, many of which have not yet been published. The Committee would like to express its appreciation to the personnel of these organizations. The Committee members would also like to thank their respective employers for allowing them to participate in this very time-consuming activity. Finally the Committee acknowledges the enthusiastic support of the ANS staff, particularly Marianne Mnichowski, who was an invaluable asset, serving as our liaison with the Society.

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1. INTRODUCTION

This document is the final report of the American Nuclear Society Special Committee on NUREG-1150. It contains a summary of the conclusions of the Committee regarding the revised version of NUREG-1150, published in June 1989. The Abstract of that report states:

> "This document discusses the risks from severe accidents in five commercial nuclear power plants. Information is presented on the frequencies of core damage accidents from internally initiated accidents (and from externally initiated accidents for two plants), containment performance under severe accident loads, releases of radioactive material and offsite consequences, and risk (the product of accident frequencies and consequences). This report is a second draft for peer review, modified to account for comments on a February 1987 draft from the public and three formal peer reviews of that draft. Following a peer review of this version, a final report will be issued.
>
> Volume 1 of this report provides summaries of the risk analysis results for the five studied plants, perspectives on these results, and a discussion of the role of these risk analyses in the NRC staff's severe accident regulatory program.
> Volume 2 of this report provides more detailed discussion of the methods used in the risk analyses, additional discussion on specific technical issues important in the analyses, and responses to comments received on the February 1987 draft."

The five Light Water Reactor (LWR) power plants studied in NUREG-1150 include three Pressurized Water Reactors (PWRs), Surry, Zion and Sequoyah, and two Boiling Water Reactors (BWRs), Peach Bottom and Grand Gulf. Only Surry and Peach Bottom were evaluated for externally initiated events. These are also the plants that were evaluated in the earlier Reactor Safety Study, WASH-1400.

Figure 1.1 (Fig. 1.4-1 from NUREG/CR-4550, Vol. 1, Rev. 1) shows the organization of the analysis and the flow of information in NUREG-1150. It also illustrates some of the key terminology (e.g., front-end or back-end analysis) which are used widely in the study. NUREG-1150 employed a traditional probabilistic risk assessment (PRA) approach to the analysis of risk of severe accidents; however, several new and unique features to the approach and methodology were used in the study.

- The study was as much a sensitivity analysis as it was a PRA. There was a strong focus on the calculation of the uncertainty in the risk values. This resulted in a somewhat reduced emphasis on the calculation of the best estimate values.

- Formal, professionally guided expert opinion elicitation was utilized extensively to develop information for which the analytical or experimental results were not available or considered inadequate. Seven teams of experts, representing all segments of the nuclear technology community were assembled to address specific groups of issues. However, a substantial fraction of the issues quantified in the study were developed by the project staff and contractor personnel rather than the expert panels.

- Because of the many hundreds of results obtained from the event trees, it was necessary to aggregate these results into many fewer groups at three points in the analysis before proceeding with the subsequent step. Thus, it is not possible to follow a unique accident scenario all the way through the accident from initiator to risk. The points of aggregation were after the core damage frequency analysis, after the accident progression analysis, and after the source term analysis.

In addition to the ANS Committee, three other committees were formed to review NUREG-1150. These include:

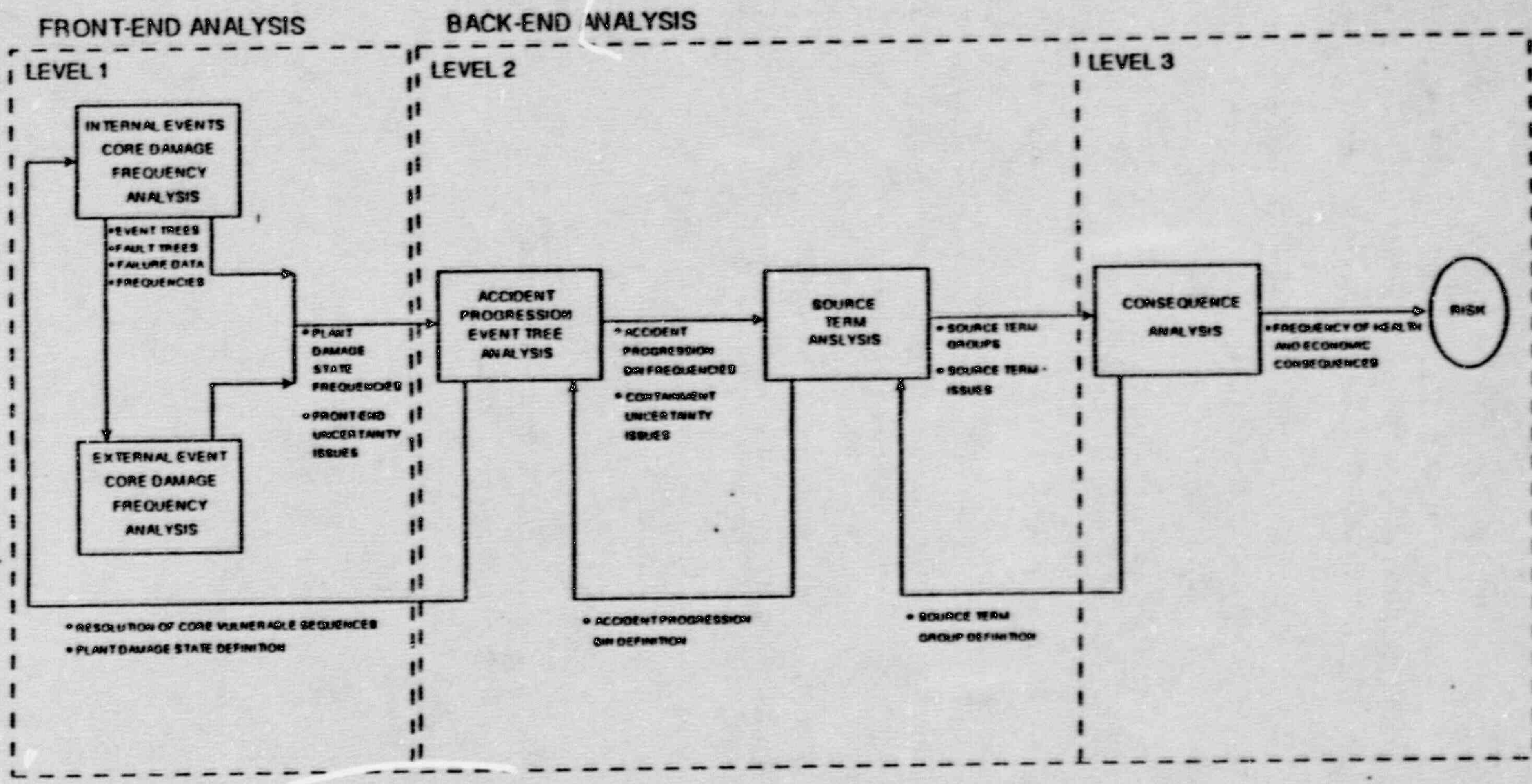


Fig. 1.1. Organization of the Analysis and the Flow of Information in NUREG-1150.

- A Committee formed by the NRC under the chairmanship of Dr. H. Kouts (BNL) to review the NUREG-1150 methodology and uncertainty analysis. A report (NUREG/CR-5000) was issued in December 1987.

- An NRC Committee formed under the chairmanship of Prof. W. Kastenberg (UCLA) to review the initial draft of NUREG-1150. A report (NUREG/CR-5113) was issued in March 1988.

- An official NRC Advisory Peer Review Committee under the chairmanship of Dr. H. Kouts formed in July 1989 to review the 2nd draft of NUREG-1150. A report is expected in mid-1990.

The NUREG-1150 Project has been a massive undertaking. When all the supporting reports are considered the total amount of documentation resulting from the Project is very large. Thus, it was not possible for the Committee to review all aspects of NUREG-1150 to the same level of detail. The approach utilized by the Committee included reviews of the two drafts of NUREG-1150 and selected supporting documents, discussions with individuals from the NRC and its contractors (primarily Sandia National Laboratories), presentations to the Committee by the NRC, its contractors and other organizations that had reviewed or evaluated the documents (e.g., EPRI), and attendance by members of the Committee at several of the expert opinion review sessions. Prior to issuing our initial report in April 1988, the Committee reviewed and utilized, as appropriate, the comments of other organizations and institutions which were submitted in response to the initial draft of NUREG-1150.

2. FINDINGS AND RECOMMENDATIONS

a. Findings

Based on our overall review and appraisal of NUREG-1150 and the supporting reanalysis, the Committee finds:

(1) ***NUREG-1150 is a Major Achievement*** - Notwithstanding a number of specific areas of concern, NUREG-1150 is a major achievement in the field of risk assessment, which pioneers the structured use of expert opinion and, to a large extent, represents a compilation of the current state of knowledge in the various aspects of severe reactor accident analysis as of early 1988. Its principal contributions are (a) the quantification of both the central tendencies and the ranges of uncertainty in the risks of nuclear power plant accidents, and (b) a compendium of current information pertaining to severe accident analysis, which is principally contained in the many volumes of the supporting series of documents, NUREG/CR-4550 and 4551.

(2) ***The Revised Draft Reports Essentially a New Study*** - The revised draft of NUREG-1150 is a substantial improvement over the initial draft not only in the format of the report itself but, equally important, because of the significantly improved analysis on which the report is based. The reanalysis of the five plants, which was performed subsequent to the initial draft, was so complete as to constitute essentially a new study. Utilization of the results of the initial draft should be discouraged. The change in title from "Reactor Risk Reference Document" to "Severe Accident Risks; An Assessment for Five U.S. Nuclear Power Plants" is appropriate and more descriptive of the work presented. Although significantly improved, NUREG-1150 is not without its shortcomings, a number of which are discussed in subsequent findings and in the body of our report.

(3) ***The Revised Draft Provides a Balanced Presentation of the Central Tendencies and Uncertainties in Risk*** - As noted in the Committee's report on the initial draft of NUREG-1150 (April 1988):

"The Lewis Commission criticized the Reactor Safety Study (WASH-1400) for emphasizing the best estimates of risk and not providing adequate emphasis on the uncertainties involved in risk estimates. The pendulum has swung too far in the opposite direction in the draft version of NUREG-1150 in that it emphasizes the quantification of uncertainty over the quantification of risk."

We find that the revised report provides a balanced representation of both the central tendencies and uncertainties in risk, thus correcting the deficiency noted in our initial report.

(4) *The Use of Expert Opinion in the Revised Study was Greatly Improved* - One aspect of the initial draft of NUREG-1150, which resulted in considerable comment and concern, was the reliance on expert opinion. Some form of expert judgment is necessary in severe accident studies because there is a paucity of data and analysis to quantify complex phenomena that must be addressed. Additionally, it is virtually impossible to undertake the quantification of the range of risk uncertainty without resorting to the use of expert opinion because of the extensive number of analyses which otherwise would be required. The Committee believes that the use of expert opinion polling involved in the revised draft is rigorous and structured, and represents not only a significant improvement over its use in the initial draft but also in comparison with previous PRAs. However, expert opinion, even when properly structured, should be applied with caution and the results treated with some skepticism since the experts are dealing with poorly understood and complicated phenomena.

(a) In NUREG-1150 the use of expert opinion included the largest pool of expertise ever assembled in support of a PRA, including representatives from government, national laboratories, industry and universities. By contrast, in the initial draft, the expert pool was essentially limited to the project staff and some contracted personnel, and

(b) a formal, structured, process under the guidance of decision theorists, trained in expert opinion elicitation, and a historical record, i.e., the underlying rationale for each expert's opinion, was developed and will be included in the supporting documents.

Notwithstanding the above comments, the Committee notes two aspects of the use of expert opinion which raise some concern, namely: (1) a substantial fracction of the data in NUREG-1150 were not developed from expert panel elicitation, but were quantified by the Project staff from available data sources and, (2) in some instances the risks are dominated by data not reviewed by the expert review panels, e.g., the steam generator tube rupture accidents for PWRs.

(5) *NUREG-1150 Should Supplant WASH-1400* - Although the analysis was carried out for only five specific U.S. nuclear power plants, some of the results and the methodology appear to the Committee to have generic implications, and NUREG-1150 should supplant the Reactor Safety Study (WASH-1400), which was limited to the analysis of two of the plants included in the present study (Surry and Peach Bottom) and is now outdated. Comparison of the results of the two studies indicates:

(a) The core damage frequency for Surry is only slightly lower than reported in WASH-1400. Core damage frequency reductions due to plant modifications and procedural improvements have more than offset increases, such as a tenfold rise in the small loss of coolant accident (LOCA) frequency, due to the inclusion of pump seal LOCAs, and improvements in techniques of probabilistic risk assessment. However, for Peach Bottom, the median core damage frequency has been reduced by a factor of fifteen, which NUREG-1150 attributes to a combination of plant improvements, procedural modifications and improved assessment of initiating events. The Committee is in general agreement with this assessment but questions the reliance on generic data in arriving at plant-specific conclusions (see Section 3.B).

(b) Great advances have occurred in accident progression and containment performance analysis since WASH-1400. Two of the results of the PWR analyses that are noteworthy are: (1) containments fail late, if at all, and (2) the substantial reduction in early containment failure has resulted in containment bypass sequences becoming the dominant contributors to the newer reduced levels of risk.

Some of the dramatic changes since WASH-1400 in containment failure probability, conditional on severe core damage occurring, are illustrated in the following tabulation.

Conditional Probability of PWR Containment Failure

	WASH-1400*	NUREG-1150**		
	Surry	Surry	Sequoyah	Zion
No Cont. Failure	0	0.81	0.66	0.74
Late Cont. Failure	0.76	0.06	0.21	0.24
Early Cont. Failure	0.16	0.01	0.07	0.014
Containment Bypass	0.08	0.12	0.06	0.006

Although containment bypass was a significant contributor to risk in the Surry analysis in WASH-1400, the mean frequency of this mode of containment failure in NUREG-1150 is twelve times as likely as early containment failure, and, as a result, bypass sequences completely dominate offsite risk in the present study. (Note. Steam generator tube rupture accidents account for roughly one half of the mean frequency and over 90% of the median frequency of bypass sequences in the NUREG-1150 analysis of Surry. Steam generator tube rupture sequences were not significant contributors to risk in WASH-1400.)

Changes in conditional containment failure probability for BWRs are not as pronounced as for PWRs, as illustrated in the following comparison.

Conditional Probability of BWR Containment Failure

	WASH-1400*	NUREG-1150**	
	Peach Bottom	Peach Bottom	Grand Gulf
No Cont. Failure	0	0.26	0.23
Late Cont. Failure	0	0.04	0.28
Cont. Venting	---	0.13	0.38
Early Cont. Failure	1.00	0.57	0.45

* Only median values were reported in WASH-1400
** Only mean values are available in NUREG-1150

(c) Significant reductions in the calculated release of fission products to the environment are evident in NUREG-1150. This is particularly true for accident sequences involving large release fractions, e.g., greater than 10% of the available core inventory of iodine and cesium. These reductions are due to a combination of reduced core damage frequency, substantial decreases in early containment failure, and recognition of increased retention of fission products within the plants' systems and structures.

(d) Predicted offsite radiological consequences are lower in NUREG-1150 than in WASH-1400. However, direct comparisons between the results of the two studies are meaningless for a number of reasons.

- NUREG-1150 uses site-specific meteorological and population distribution data, whereas WASH-1400 utilized averages from a number of sites,

- The emergency response measures assumed in the two studies differ substantially, and

- Substantial differences exist between the CRAC computer model used in WASH-1400 and the MACCS model used in NUREG-1150.

(6) ***The NRC Safety Goals are Shown to be Met for All Five Plants Studied*** - Specifically, those consequence analyses reported in NUREG-1150 (consequences for seismic event initiators were not reported) show that:

(a) The calculated risks, for all five plants studied, are substantially below the NRC's safety goals for both individual early fatality risk and individual latent cancer fatality risk, even when the range of uncertainty up to the 95th percentile is included.

(b) The central tendencies of the frequency of a large release are below the NRC staff recommendation of 1×10^{-6} per reactor year for all five plants studied and are substantially below that value in the case of the Surry, Peach Bottom, and Grand Gulf analyses. The ranges of uncertainty, including the 95th percentile of risk, slightly exceed 1×10^{-6} in the Sequoyah and Zion analyses. (Note. Although described as a "tentative goal" in NUREG-1150, the large release frequency proposed by the staff is not a part of the safety goals approved by the Commission.)

(7) ***The NUREG-1150 Documentation is a Useful Compendium of Current Severe Accident Analysis Information and Data*** - NUREG-1150 and its supporting documents, particularly the multivolume series NUREG/CR-4550 and 4551, constitute an extensive compendium of information and data for use in severe accident analyses and probabilistic risk assessments. They should prove to be particularly useful as reference sources in the Individual Plant Examination (IPE) activities presently underway in support of each operating nuclear power plant in the U.S., and they may be used as a teaching resource on severe accident analysis.

The Committee does not endorse or take any position relative to the technical accuracy or adequacy of the information and data in these voluminous documents. We have not had an opportunity to review most of them, and such a review is beyond the Committee's scope.

(8) ***Specific Findings with Respect to Analytical Methods and Their Application*** - The Committee notes several concerns about analytical methods and their application.

(a) Selection of Issues for Quantification by Expert Review Panels - Because of project limitations, the number of issues subjected to expert opinion polling was limited. Other issues requiring expert judgment, but deemed to be less important, were quantified by the project staff or contractor personnel. Thus, not all the expert judgment included in NUREG-1150 was subjected to the disciplined, structured process described above. The fraction of the numbers not generated by the expert review panels is not reported but may be quite large. The Committee believes that the discussion of issue quantification could be substantially improved, with much clearer indication of what probability distributions were developed by the staff and which specific issues were quantified by the expert review panels in each segment of the study for each of the five plants studied.

(b) Core Damage Frequency Analysis - Although NUREG-1150 is described as being "a set of modern PRAs, having the limitations of all such studies," the level of modeling in the front end (Level 1) analysis, in some areas, is not as detailed as that found in other current PRAs. Resulting perspectives based on models that lack sufficient detail may not be adequately supported (see Section 3.B).

(c) Containment Performance Analysis - Major changes have occurred affecting the importance of certain phenomenological issues in containment performance analysis as reported in the initial and revised drafts of NUREG-1150. However, the bases for these changes and insights that would provide an understanding of why these shifts have occurred are not discussed in the report. For example, direct containment heating (DCH) was identified in the initial draft as a major contributor to PWR early containment failure and risk. In the revised version this phenomenon goes virtually unmentioned, yet its disappearance reflects major changes in: (1) the potential for primary system depressurization prior to vessel meltthrough and (2) the magnitude of the contribution to containment loading resulting from high pressure melt ejection if the reactor pressure vessel fails at high pressure. Chapter 9, which purports to provide perspectives on containment performance mentions DCH only in passing, with no reference to its reduced importance. The Committee finds that the development of many of the important phenomenological issues and insights gained from the containment performance analysis results are not well documented in the revised draft. While the Committee finds that the large containment event tree methodology used in NUREG-1150 is a satisfactory representation of the appropriate phenomenological issues and accident progression time regimes, it notes that some key events for some plants are quantified based on opinion elicited from the experts on controversial phenomenological issues for which there was a paucity of data. As a result, panel members may have held widely divergent views that led to bi-modal uncertainty distributions, thus possibly affecting the mean value of the distribution. An example is the issue related to drywell meltthhrough in the Peach Bottom analysis. The Committee cautions users of the containment performance analysis results to consider and understand the underlying basis for the expert judgments, as reported in the NUREG/CR 4551 volumes for each plant, in order to use the results intelligently.

(d) Source Term Analysis - The development of severe accident source terms utilizing the XSOR parametric source term models is perhaps the most inexact aspect of the NUREG-1150 study. Although these models are mathematically correct, in the sense that they are simple mass balance equations, it is their very simplicity that raises concerns. The use of the XSOR codes requires the representation of many complex, and frequently interdependent, processes with a series of single numbers for a given accident, e.g., the fraction of iodine retained in the reactor coolant system. These numbers were frequently applied to many accident scenarios. In all, very large numbers of individual source terms were estimated, e.g., 20,000 for Surry and 75,000 for Grand Gulf.

Such a simplified representation of complex processes and phenomena does not permit quantification of a number of small effects, which, when taken together, may substantially affect the source terms. The Committee finds the analysis of source terms utilizing these codes to be very approximate, but nonetheless necessary, in the evaluation of the uncertainties and in studies directed at identifying which accident sequences and parameters are important from a risk perspective. Source terms for the risk dominant sequences, thus identified, should be subjected to more detailed analysis, after publication of NUREG-1150.

Two additional Committee findings with regard to source terms are:

- A time cutoff of 24 hours after the onset of core degradation for the release of radionuclides was used throughout NUREG-1150, although no mention of this fact is contained in the report. The rationale for this cutoff was stated to the Committee to be that accident management actions would be taken to mitigate releases after that time. The Committee does not agree or disagree with the use of this cutoff, however, it points out that its use should be indicated and discussed in the report. The significance of this issue is that several analyses, particularly for BWRs, have shown that late revaporization of volatile species, such as iodine, (e.g., 30 to 60 hours after the initial release from the core and deposition in the reactor coolant system) may be substantial if no operator actions are taken.

- The source terms for containment bypass sequences, which are the dominant contributors to risk in the PWR analyses, may be characterized as conservative approximations. In the case of interfacing system LOCAs (V-sequences), no analysis of fission product behavior in multicompartmented auxiliary buildings containing fire protection spray systems was carried out. Other recent studies have indicated that inclusion of such physical features has the effect of increasing retention within buildings and decreasing releases to the atmosphere.

  With respect to source terms for steam generator tube rupture (SGTR) accidents, simplified conceptualizations were utilized in lieu of detailed analyses. Experience has shown that such simplified approaches tend to be conservative and overstate releases. The SGTR sequence descriptions and XSOR input were estimated by a group of Project personnel and were not subject to review by the Expert Review Panels.

(e) *Consequence Analysis* - The consequence analysis presented in NUREG-1150 is practically impossible to reproduce without access to the MACCS computer program and detailed information about the many input variables, which are not available at this time. Information important in understanding the results is missing, such as the fact that inhalation doses reflect lifetime dose commitments. (Note: By contrast, NUREG-0396, which addresses the technical basis for emergency planning zones, utilized a one year dose commitment.)

(9) *External Events Analyses* - The bulk of the NUREG-1150 study considers risks from accidents resulting from internally initiated events. An analysis of the risks due to external events was carried out for two of the plants, Surry and Peach Bottom. Only fires (defined as an external event) and seismic events were significant contributors to risk.

(a) Fire Analysis - The core damage frequency and risk due to fires at Surry are somewhat less than for internally initiated events. On the other hand, at Peach Bottom core damage is more likely, and risks from fire initiators are greater than those from internal events, due primarily to common mode failures caused by fires.

(b) Seismic Analysis - The prediction of the magnitude and frequency of seismic activity is undoubtedly a demanding and uncertain exercise. However, the seismic hazards curves used in NUREG-1150, depicting a relatively high likelihood of very intense ground motion in the eastern United States, are particularly unexpected. Their use in the analyses yields significant core damage frequencies due to ground acceleration in excess of 0.5g, a result which is intuitively disturbing for an eastern site. Because of large differences between the seismic hazards curves utilized in the study and the fact that the radiological consequences could be overwhelmed by the direct offsite consequences of the seismic event, the Committee agrees with the authors' decision not to include the radiological consequences of seismic events in NUREG-1150.

(10) *Analyses of accident Prevention and Mitigation Features Have Been Deleted* - The initial draft of NUREG-1150 included cost/benefit analyses of plant features intended for accident prevention and accident mitigation. These subjects were treated superficially, and the Committee agrees with their deletion in the revised draft. These are complex subjects which involve subtle plant-specific considerations that, in the opinion of the Committee, are beyond the scope of NUREG-1150.

(11) *Discussion of Emergency Response Has Been Retained* - The Committee agrees that discussion of emergency response should be included in the revised draft; however, the discussion would be more useful if data on median and mean dose levels as a function of distance from the plant were included. This would facilitate comparison of offsite consequences for the five plants studied, independent of population distribution differences and emergency response measures from site to site, and make the data more generally useful for comparisons with WASH-1400 and other studies.

(12) *The Quality of the Report is Substantially Improved* - In its overall configuration the revised version of NUREG-1150 is an improved document over the first draft; however, some shortcomings in the quality of the report remain.

(a) Following a concise introductory chapter in which the objectives of the project and the report are stated, there is a chapter in which the methodology is presented, including explanatory examples of how the results will be displayed. Appendix A amplifies the discussion of methodology, Appendix B presents a sample calculation, and provides a description of how a number of difficult decisional issues involving uncertainty were evaluated. These Appendices and the first two chapters provide a detailed and clear explanation of the process.

(b) The five chapters in which results for each of the plants are presented are also quite clear and readable. Because essentially the same format is used in each chapter, with the same explanatory remarks, these chapters are somewhat repetitious. However, some substantial differences in the Zion analysis, performed by the Idaho National Engineering Laboratory and the analysis of the other four plants, performed by Sandia National Laboratories, are not highlighted.

(c) A major criticism of the original NUREG-1150 was the manner in which results were portrayed. The current version does a much better job of presenting the results. Probability distributions, including medians and means, are shown for core damage frequency, containment failure probability and individual isotope releases. A particularly helpful form of the results are the matrix-like figures in which mean values of accident progression bins (e.g., containment performance) are combined with mean plant damage states (and their associated core damage frequencies). Pie charts are used effectively to display qualitatively the contributions of various initiating events and accident progression scenarios.

(d) The last part of the main report, six chapters of "perspectives" on the calculated results and their use, is the least effective and hard to follow. Certain of the material here is very worthwhile (e.g., some aspects of the comparison with WASH-1400) but much of the discussion seems forced, and the observations range from the obvious to those for which the analysis provides no apparent basis. These chapters could be shortened considerably with no loss of their impact.

(13) *The Adequacy of the Report* - In Section 6 of our report we address the adequacy of NUREG-1150 with respect to the intended uses stated in the revised draft. Inasmuch as the report points out that in any of its intended applications NUREG-1150 will be used only as one of a number of information sources, we find that it is adequate for its stated uses. We offer, however, certain cautions or encouragements with respect to the intended uses, including:

(a) To Develop Guidance for the Conduct and Review of IPEs - In using NUREG-1150 in this context we caution that its limitations should be kept in mind and note its use for a specific plant should be justified.

(b) To Assist in the Consideration of Improvements to Containment Performance - Because the final determination of containment adequacy is plant-specific, those responsible for performing these analyses must make the final judgment regarding the applicability of information from NUREG-1150.

(c) To Assist in the Identification of Plant Operational Features and Practices That Have an Adverse Impact on Plant Safety - As long as the plant-specific nature of the models is recognized, we concur that the NUREG-1150 results can be used in this way.

(d) To Assist in Safety Goal Implementation Strategies - There is no basic reason why the NUREG-1150 results for the five plants studied, as well as other risk assessments, cannot be used for this purpose. The Committee does not believe the limited information presented in NUREG-1150 with respect to the NRC staff's proposed large release goal of less than 1 x 10-6 per reactor year would be particularly useful in the evaluation of implementation strategies.

(e) To Assist in Evaluation of Research Priorities and Prioritization and Resolution of Generic Issues - Although the NUREG-1150 study certainly can be useful in this regard, the Committee believes an opportunity to do so within NUREG-1150 has been missed. Risk assessment results, especially the results of many risk assessment studies, taken together, can and should be used for the prioritization of safety issues and the resolution of others. The Committee believes that such use of NUREG-1150 should be considered a priority application and a principal benefit of the substantial resources expended in this multiyear study.

b. Recommendations

(1) We recommend that the members of the American Nuclear Society become familiar with NUREG-1150 and its supporting documents and avail themselves of their useful features. While the Committee believes NUREG-1150 represents a valuable and extensive compendium on severe accident analysis, this recommendation is not an endorsement of the accuracy and reliability of all the

information and data contained therein. Caution is urged in the use of some results as discussed in our findings, in other recommendations, and elsewhere is our report.

(2) As noted in our findings, the Committee believes that NUREG-1150 should supplant the Reactor Safety Study (WASH-1400) which is now outdated.

(3) Since data from WASH-1400 were relied on heavily in the development of existing emergency planning requirements (see NUREG-0396), we recommend that data on the conditional probability of exceeding specific doses as a function of distance (e.g., similar to Figure I-11 of NUREG-0396) be included in the final draft of NUREG-1150, in order to provide more up-to-date information for assessing this important area.

(4) The Committee urges the NRC staff to issue the supporting NUREG/CR-4550 and 4551 volumes for each of the five plants studied as soon as possible. Issuance of these reports should not be tied to publication of the final draft of NUREG-1150. We believe the users of NUREG-1150, such as individuals and groups involved in Individual Plant Examinations (IPEs), should have the benefit of these supporting documents.

(5) Our findings, noted above, and observations elsewhere in our report should not be interpreted as identifying the need for a major rewrite of NUREG-1150. Rather, we recommend publication of the final report as soon as possible. However, we do recommend that some sections be substantially modified. For example, Chapters 8, 9, and 10, which discuss "perspectives," should be revised to more accurately reflect the results of the study.

(6) The final version of NUREG-1150 should clearly state that it should be viewed as a new study and as a replacement for the initial draft of the report. This is important because: (a) the earlier draft was widely circulated, it received extensive comments both domestically and internationally, and many issues were raised which have subsequently been addressed, and (b) some of the results and conclusions of the reanalysis are substantially different than those of the initial draft.

(7) Because the readership of NUREG-1150 is expected to be much larger than for the supporting documents, a summary tabulation, or matrix, should be included in the final draft which lists the specific issues studied in each segment of the analysis, along with an indication as to how each issue was quantified, e.g., by expert review panels. Such a tabulation should clearly delineate ho' each issue was treated for each of the five plants studied. Some simple indication of the relative importance of each issue should be included with reference to where more complete discussions can be found.

(8) The use of the XSOR Codes in NUREG-1150 has resulted in estimated source terms for the major contributors to risk, such as interfacing system LOCAs and steam generator tube rupture accidents for PWRs and station blackout and ATWS sequences for BWRs. For those sequences which NUREG-1150 has shown to be significant contributors to risk, more sophisticated analyses of the source terms for these particular cases should be undertaken in follow-up studies.

(9) As presently reported, the source terms and concomitant offsite risks for PWR accident sequences involving containment failure are largely obscured by the effects of containment bypass sequences, which are believed to be overstated due to simplified modeling, and need to be subjected to more sophisticated analysis. Therefore, we recommend that the source terms and offsite consequences of these two separate and distinct classes of accidents, i.e., containment failure and containment bypass, be reported separately, as well as the combined data presently reported. This is important not only because we believe the results for the bypass sequences are more likely to change in time, but other considerations logically apply to the many so-called "in-containment" ccident sequences considered in NUREG-1150. These are essentially unaffected by the outcome of investigations of containment bypass sequences.

(10) To help overcome some of the difficulties encountered in comparing offsite consequence analyses among the five plants studied, and with other studies, we recommend that data on dose as a function of distance be included in the final report (similar to Table 10.1 of the initial draft of NUREG-1150). This would provide radiological consequence results independent of population distribution and make the results more generally useful.

(11) We recommend that resolution of the seismic risk estimates be addressed outside the scope of NUREG-1150, in keeping with our Recommendation #5, that publication of the final draft of NUREG-1150 not be delayed.

3. REVIEW OF NUREG-1150

This section summarizes the Committee's review of the NUREG-1150 methodology and the results from application of the internal event methodology to five (5) plants and the external methods to two (2) of the five plants.

a. Overview of Methodology and Use of Expert Opinion

Each part of the methodology (accident sequence analysis for example) is divided into a description of the methodology, a summary of the results, and a discussion of the merits and shortcomings of the methodology as well as a discussion of the results and conclusions. The Committee's evaluation reflects the information contained in the top level documents, NUREG-1150 summary reports, and briefings by the NRC and the NUREG-1150 Project staff. The Committee did not have the benefit of many of the supporting NUREG/CR 4550 and 4551 documents which provide the details of the expert elicitation and specific plant analyses. NUREG-1150 is an attempt to develop comprehensive risk information on U.S. nuclear power plants through the analysis of five plant types. The analysis performed for each plant includes the risk assessment elements needed to estimate the off-site risk. These elements are shown and illustrated in Figure 3.1 (from Sandia National Laboratories), and collectively are referred to as a level 3 probabilistic safety assessment. In addition to the estimate of off-site risk, a major part of the NUREG-1150 effort was expended in the estimation of the uncertainties in the risk estimates.

The revised draft of NUREG-1150 includes an assessment of the risks due to external events for two nuclear power stations, Surry (3-loop PWR) and Peach Bottom (BWR with Mark I containment). These are the plants that were used to typify the two major reactor designs for the Reactor Safety Study, WASH-1400. External events were not included in the initial draft of NUREG-1150.

As illustrated in Figure 3.1, the key risk elements are the estimate of accident (core damage) frequency, accident progression through containment response to phenomena leading to possible failures of the containment, estimates

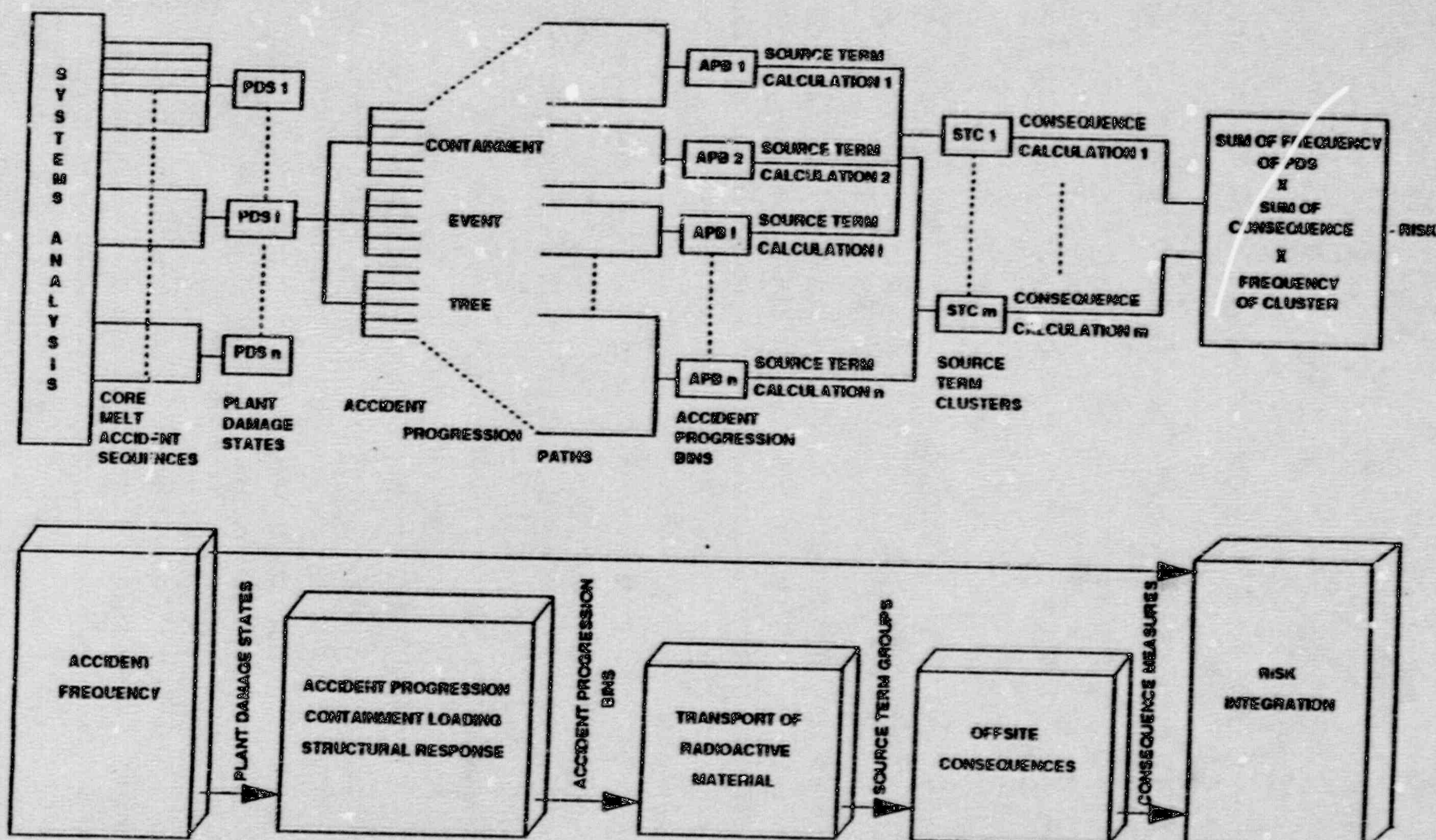


Fig. 3.1. Overview of NUREG-1150 Risk Analysis

of the source terms resulting from the various accident progressions, and estimates of the consequences resulting from each of the source term estimates. The core damage frequency analysis leads to a number of core melt sequences having similar safety system and support system failures. These are grouped into "plant damage states" according to similar failures of equipment function and reactor system conditions as vessel failure approaches. Each of these plant damage states is carried through a containment event tree (one for each plant analyzed) to develop the phenomenological conditions and containment response for each accident progression path which determine the timing and failure mode of containment and influence the transport and release of fission products.

Since there are tens of plant damage states resulting in hundreds of accident progression paths, similar paths are grouped into accident progression bins so that the number of source terms required to be developed can be limited. Even so the number of source terms developed for some plants reached several thousand. This magnitude of data required the use of a simplified mass balance code rather than a detailed time dependent physics based computer code. Thus each of the accident progression bins was analyzed using factors in a simplified expression to represent complex fission product transport phenomena. Again, a large number of source terms were developed, and these also had to be grouped by similar source term characteristics.

Source terms tended to naturally group into separable patterns according to type and timing of release, and thus they were grouped according to these "clusters" for calculation of the off-site consequences and risk.

The expert opinion elicitation process used by the NRC's project staff as a means to establish uncertainty of major issues deserves special note. While the application of expert polling as used in the revised draft of NUREG-1150 is not unique, it is rarely used correctly and rigorously. In fact, the initial draft of NUREG-1150 was very severely and extensively criticized for the way in which the elicitation process had been carried out. In response to these public comments, the NRC initiated a major and very thorough expert polling process. Professionals in the field of opinion polling were brought in to guide the program, train the expert panels, and ensure that the polling was properly conducted. The issues that are important to the core damage frequency, containment response, or source terms were screened to identify those for which sufficient information existed to adequately characterize the uncertainty distribution and those for which relatively little information was available. Experts were chosen from all sectors of the nuclear industry, trained in the expert elicitation method, and presented with all of the available data on the particular subject.

Experts were assigned to small groups to consider particular issues and were given several months to develop their individual analyses of the issues and defend them to the group. In the final polling each member submitted a probability distribution for the particular issue, and these distributions were averaged. In all, ten accident sequence issues (NUREG-1150, Table A-1) and fourteen containment phenomenological and source term issues (NUREG-1150, Table A-1) were addressed by the experts as well as several secondary issues.

In addition to probability distributions developed by the expert panels, a number of distributions for specific issues were developed by the Sandia project staff. All distributions were sampled by a specialized Monte Carlo sampling routine (Limited Latin Hypercube Sampling) repeatedly to provide sets of input values. Each of these input value sets was propagated through the system fault trees, the containment event trees, and analytical models. The containment or accident progression events that were treated as uncertainties were assigned the values corresponding to the input set. All others were held fixed. The corresponding output values yielded distributions in the output containment failure and source term input parameters. In this way (repetitive sampling and propagation) results in the form of probability distributions were developed for core damage frequency, containment response, and source terms. Each of these areas is discussed below.

b. Core Damage Frequency Analysis

(1) Core Damage Frequency Methods for Internal Events

The methodology employed to determine core damage frequency (CDF) contains the essential elements required to perform a probabilistic risk assessment study of the plant systems, procedures, and operations (generally defined in the literature as a level one analysis, but sometimes called a "front-end" analysis). These essential elements include:

- Accident Sequence Initiating Event Analysis
- Accident Sequence Event Tree Analysis
- Dependent Failure Analysis (Modified Beta Method)
- Human Reliability Analysis (THERP Method & sometimes reliability methods)
- Data Base Analysis (Generic data and plant specific data when available)
- Accident Sequence Quantification Analysis
- Uncertainty Analysis

The NRC utilized two national laboratories to perform the analyses of the five plants evaluated in NUREG-1150. The Surry, Sequoyah, Peach Bottom, and Grand Gulf studies were performed by the Sandia National Laboratories (SNL). The Zion Nuclear Plant was analyzed by Idaho National Engineering Laboratory (INEL) using a fundamentally different approach. INEL employed different techniques in their study in order to take advantage of the extensive PRA already performed by the Commonwealth Edison Company and its contractors. This different approach for Zion may have led to differences between the Zion results and the other studies.

In the SNL studies, Sandia used a fault tree linking approach to quantify accident sequence frequencies. With this approach, event trees were used to define the various combinations of initiators and system failures that can lead to core damage. The system fault trees were combined to represent the functional failures (e.g., loss of high pressure injection) identified in the event trees for specific initiators. The composite fault trees that represent accident sequences are then solved to determine the combinations of initiators, component failures, and/or human errors (i.e., cutsets) that lead to core damage. Quantification of the events that comprise these failure combinations results in an estimate of accident sequence frequencies.

In the Zion study, INEL used a large event tree approach to calculate accident sequence frequencies. With this approach, a large event tree is used to define a series of conditional probability of events. Accident sequence frequencies were calculated by multiplying an initiator by the conditional events that lead to core damage. As shown in Figure 3.1, inputs to the front end phase of the risk study are initiating events and outputs are plant damage states. The latter are defined in terms of the status of the reactor coolant system, the emergency core cooling systems, and the containment safeguards systems at the onset of core damage.

(2) Summary of Core Damage Frequency Results for Internal Events

A summary of internal event core damage frequency estimates for each plant is provided in Table 3.1. The principal core damage contributors for each plant are summarized in Table 3.2. Contributions to core damage are expressed in percent of the total mean core damage frequency.

(3) Discussion of Internal Event Core Damage Frequency Analysis

The NUREG-1150 results (Table 3.1) are consistent with an apparent trend that has developed with recent PRAs, a gap between the core damage frequency of PWRs and BWRs. Since the publication of WASH-1400, the CDF of BWRs have dropped substantially relative to those for PWRs. The results of NUREG-1150 show that the CDFs of the BWRs (Peach Bottom and Grand Gulf) are about an order of magnitude below the CDFs of the PWRs (Surry and Sequoyah). A plant-unique support system dependency (which the owner-operator has committed to correct) caused the Zion CDF to be out of line with the other PWR results (see Table 3.1).

Table 3.1 Summary of NUREG-1150 Core Damage Frequency Estimates[a]

	5%[b]	Median	Mean	95%[c]
Zion	1.1E-4	2.4E-4	3.4E-4	8.4E-4
Sequoyah	1.2E-5	3.7E-5	5.7E-5	1.8E-4
Surry	6.8E-6	2.3E-5	4.1E-5	1.3E-4
Peach Bottom	3.5E-7	1.9E-6	4.5E-6	1.3E-5
Grand Gulf	1.7E-7	1.2E-6	4.0E-6	1.2E-5

Notes: [a]Per reactor year.
[b]This value (5th percentile) represents the confidence limit at which there is only one chance in twenty that the actual core damage frequency is less than the corresponding frequency.
[c]The 95th percentile represents the confidence limit at which there is only one chance in twenty that the actual core damage frequency is higher than the corresponding frequency.

Table 3.2 Principal Contributors to Internal Core Damage Frequency

Accident Type	Zion	Sequoyah (Values in Percent)	Surry	Peach Bottom	Grand Gulf
Station Blackout	1.9	26	67	49	98
Loss of Coolant (LOCA)	93	63	15	5.8	<1
Failure to Scram (ATWS)	<1	3.3	3.9	42	2.8
Transients	4.2	4.4	5.1	3.1	<1
Interfacing LOCA	<1	1.1	3.9	<1	<1
Steam Generator Tube Rupture	<1	3.0	4.4	N/A	N/A

A detailed discussion of the perspectives drawn from the core damage frequency analysis is found in Chapter 8 of NUREG-1150. Comments are summarized below on some of the perspectives in Chapter 8.

(a) NUREG-1150 Perspective: ATWS sequences in BWRs are not dominant because the plants studied have implemented the anticipated transient without scram (ATWS) modifications.

Comment: In both the Peach Bottom and Grand Gulf studies no explicit modelling of the reactor protection system, the alternate rod insertion system, or the recirculation pump trip system was performed. The unavailabilities of the above systems were taken from generic data sources or were estimated using engineering judgment. Plant specific dependencies (e.g., DC power for the alternate rod insertion system) were not captured in the analysis. In view of this ob_ ~vation, it is difficult to justify this NUREG-1150 perspective about ATWS sequences.

(b) NUREG-1150 Perspective: Station blackout contributes a high percentage of the core damage frequency for both BWRs and PWRs.

Comment: Station blackout may not be a major contributor if special initiators (e.g. loss of service wUnater (SW) or loss of DC power) were assessed in more detail and uniformly across all the NUREG-1150 studies. For example, in the case of the Zion study, a common discharge header in the component cooling water system was assessed to be a major vulnerability for reactor coolant pump seal LOCAs. This vulnerability was explicitly modelled in the Zion study. In the case of the Peach Bottom study, potential vulnerabi.,ties in the service water system (e.g., common SW header for two units) were identified but not explicitly modelled. A qualitative analysis was used to exclude potential SW initiators in the Peach Bottom study. The treatment of the DC power failure initiator in the Peach Bottom study did not account for battery dependencies in the offsite power supply to the emergency busses. These omitted dependencies could result in an underestimate of the impact of special initiators or support system initiators.

(c) NUREG-1150 Perspective: Grand Gulf is better equipped than Peach Bottom to respond to a station blackout event because it has an extra diesel which is dedicated to drive the high pressure core spray system.

Comment: It would be very difficult to support this perspective since the station blackout mean core damage frequency contribution for Grand Gulf (3.9×10^{-6}/yr) is twice the frequency for Peach Bottom (2.1×10^{-6}/yr). The highest station blackout sequence for Peach Bottom, which involves the depletion of station batteries and resulting failure of all high pressure makeup systems, has a mean core damage frequency of 1.6×10^{-6}/yr. The highest station blackout sequence for Grand Gulf, which involves failure of the dedicated diesel for the high pressure core spray system, has a mean core damage frequency of 3.6×10^{-6}/yr.

(d) NUREG-1150 Perspective: Although Peach Bottom is a two-unit site with four diesels, any one of which has sufficient capacity and the appropriate cross-ties to power both units in the event of a loss of offsite power, support system dependencies (i.e., DC power and service water) offset the diesels.

Comment: The four diesels at Peach Bottom are shared between unit 2 and unit 3. Each unit has four divisions of AC power (i.e., A,B,C, & D). For example, diesel generator A provides emergency AC power to the unit 2 division A bus and the unit 3 division A bus. There are no apparent cross-ties between diesel generators and no credit for a cross-tie capability is taken into account in accident sequence recovery analysis. If support system dependencies are very significant in the Peach Bottom study, it is not clear why support system initiators (or special initiators) were qualitatively excluded from the analysis as indicated in NUREG/CR-4550.

(e) NUREG-1150 Perspective: LOCA and transient sequences are more significant at Peach Bottom than at Grand Gulf because the steam-driven high pressure injection systems at Peach Bottom are less reliable than the diesel-driven core spray system at Grand Gulf.

Comment: The dominant LOCA and transient sequences in the Peach Bottom study involve failures of the low pressure injection system, whereas the dominant transient sequence in the Grand Gulf study involve the failure of the diesel-driven high pressure core spray system (Grand Gulf LOCA sequences were less than 1.0×10^{-8}/yr). Since the dominant LOCA and transient sequences for the Peach Bottom study do not involve failures of the steam-driven high pressure injection systems, it would be very difficult to justify this NUREG-1150 perspective.

c. <u>Containment Performance Analysis</u>

(1) Methods for Accident Progression and Containment Event Tree

The containment event tree (CET) as shown in the overview of NUREG-1150 (Figure 3.1) accepts the plant damage states determined from the core damage sequences and generates accident progression paths collected into accident progression bins. In turn, these bins are used as the basis for source term calculations.

The containment event tree is used to display and track the various phenomena associated with core melt progression and containment response. These events are defined in a logical way that provides insights into features which control containment performance and allow efficient quantification of the accident progression paths. The containment event trees used in NUREG-1150 contain many more top event questions than other recent studies (About 100 detailed questions in each compared to about a dozen more general questions in many other PRAs). A tree was developed for each plant for use with all plant damage states. A formalized process (case structure) was used to insure consistent treatment of phenomena throughout each sequence. Physical quantities were reportedly tracked through the accident sequences to insure conservation of quantities (e.g., Zirconium mass).

The containment event tree was constructed by dividing the accident sequence into time regimes. These time regimes include:

- start or initiation of the accident resulting in containment systems or other support system failures which may lead to containment failure,
- the period prior to core melt leading to containment challenge,
- core melt,
- the period immediately prior to vessel breach,
- vessel breach and immediate containment response (first two hours following vessel breach), and
- the late containment challenge period following vessel breach (greater than two hours after vessel breach).

A series of questions or branch points were then developed specific to each of the time regimes identified above and for each plant. These questions were quantified by sampling the probability distributions provided by either the expert panels elicited for certain issues or by the project staff. Of the many containment event tree nodes, fourteen major questions (issues) were identified for quantification by the expert panels (out of hundreds of basic issues). The quantifications of several of these issues are included in Appendix C.

The quantification of each issue elicited of the expert panels was done by asking the experts to develop the likelihood of an event occurring with varying outcomes (e.g., the probability of 40 percent of the zirconium oxidizing prior to vessel meltthrough, 60 percent and so on). Each expert developed a probability distribution for the issue, and these were averaged to obtain the probability distribution to be used for the issue. The project analysts also developed probability distributions for issues which were considered to have an adequate basis of supporting datave and single values for some questions. To quantify the sequences and develop the output uncertainty distributions (e.g., a distributions for early containment failure), the analysts utilized a process for sampling the probability distribution known as Limited Latin Hypercube Sampling. This method forces sampling across the probability distributions for the specific issues. With relatively few samples one can develop an uncertainty distribution in the CET output when each set of values from sampling the issue distributions are propagated through the event trees. In the Limited Latin Hypercube approach, about two hundred samples were taken as compared with thousands which would be required for complete random sampling.

The outcomes of the containment event tree were examined by a post-processor computer code that examined the characteristics of the branch such as timing of release, containment failure location, etc. for grouping similar outcomes into accident progression bins that could be used directly by the source term analysts to predict fission product releases. Quantitatively, the CET product consists of a matrix of conditional failure probabilities, with one probability (mean value) for each combination of plant damage state and accident progression bin. Also included as a product is the probability distribution of early containment failure for each plant damage state. Measures of this distribution include the Mean, Median, 5th percentile value, and 95th percentile value.

(2) Summary of Accident Progression and Containment Performance Analysis Results

Figure 3.2 (a composite figure made up from several figures in NUREG-1150) presents the conditional early containment failure (ECF) probability distributions of the five plants studied. This figure shows a dramatic difference from the large dry containment represented by Surry (fails early less than one percent of the time) to the small BWR Mark I containment represented by Peach Bottom (mean early failure probability is approximately 60 percent). Figures 3.3 and 3.4 (Figures 3.5 and 4.4 from NUREG-1150) present the mean conditional containment response probabilities in matrix form (accident progression bins probabilities based on the mean core damage frequency for each of the major plant damage states) for the Surry and Peach Bottom plants respectively. Similar figures may be found in NUREG-1150 for the other plants. Notable among the various containment responses is that containment bypass

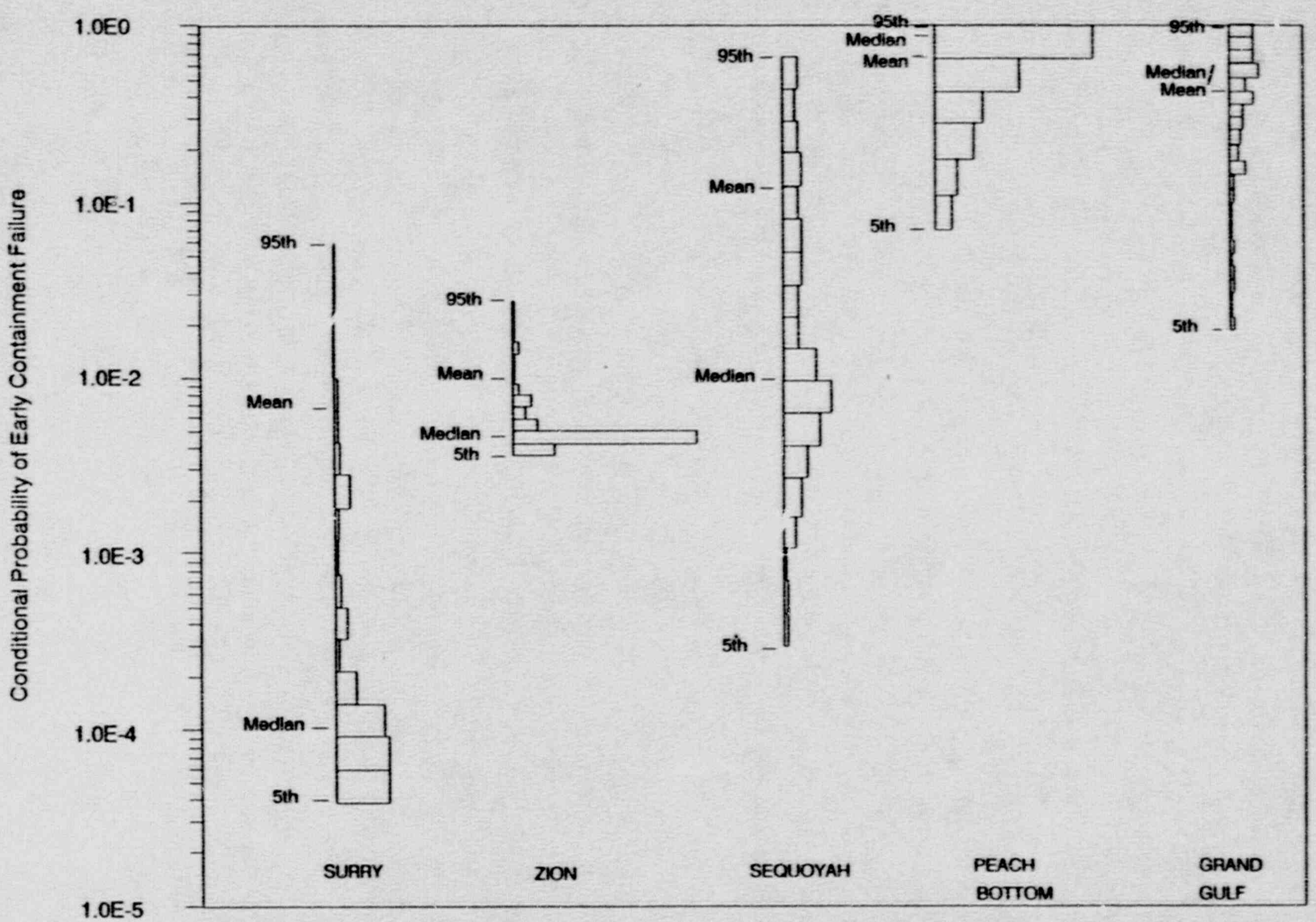


FIGURE 3.2 CONDITIONAL PROBABILITY DISTRIBUTIONS FOR EARLY CONTAINMENT FAILURE

dominates the Surry plant's potential release with only a very small fraction of early or late containment failures predicted. However for Peach Bottom, EC occurs for several failure modes a large fraction of the time. In general, ECFs for the PWR plants studied are associated with station blackout (SBO) damage states, since there are no sources of containment cooling continuously available in these sequences and vessel breach at high pressure is possible. For the BWR plants studied the susceptibility to containment challenges was greater because they have smaller containments.

(3) Discussion of Accident Progression and Containment Performance Analysis

Several important factors have emerged from the work performed for the revised draft of NUREG-1150 that have affected the differences in the PWR and BWR results. In the initial draft of NUREG-1150, direct containment heating and hydrogen burns were the major contributors to ECF for PWRs. Containment shell meltthrough and hydrogen combustion were the major contributors for the BWR plants.

For Surry, Sequoyah, and Zion evaluation of the recirculation patterns within the primary system along with an examination of the emergency procedures and other features of the PWR reactor systems has led the experts to conclude that the primary system would be depressurized before vessel breach could occur. This has substantially reduced the importance of direct containment heating (resulting from meltthrough of the reactor vessel under high pressure and dispersal of the debris into the containment atmosphere) as a containment failure contributor. Other phenomena that can cause ECF in large dry PWR containments are in-vessel steam explosions, and hydrogen combustion. These are observed to be very small contributions for Surry and Zion. For the ice condenser plant, Sequoyah, the probability of ECF is larger than for the dry containments because the ice condenser containment is relatively small. In addition to being more sensitive to hydrogen combustion than dry containments, Sequoyah was found to be somewhat sensitive also to containment wall meltthrough as a result of a postulated failure of the seal table plate during high pressure vessel breach and blowdown.

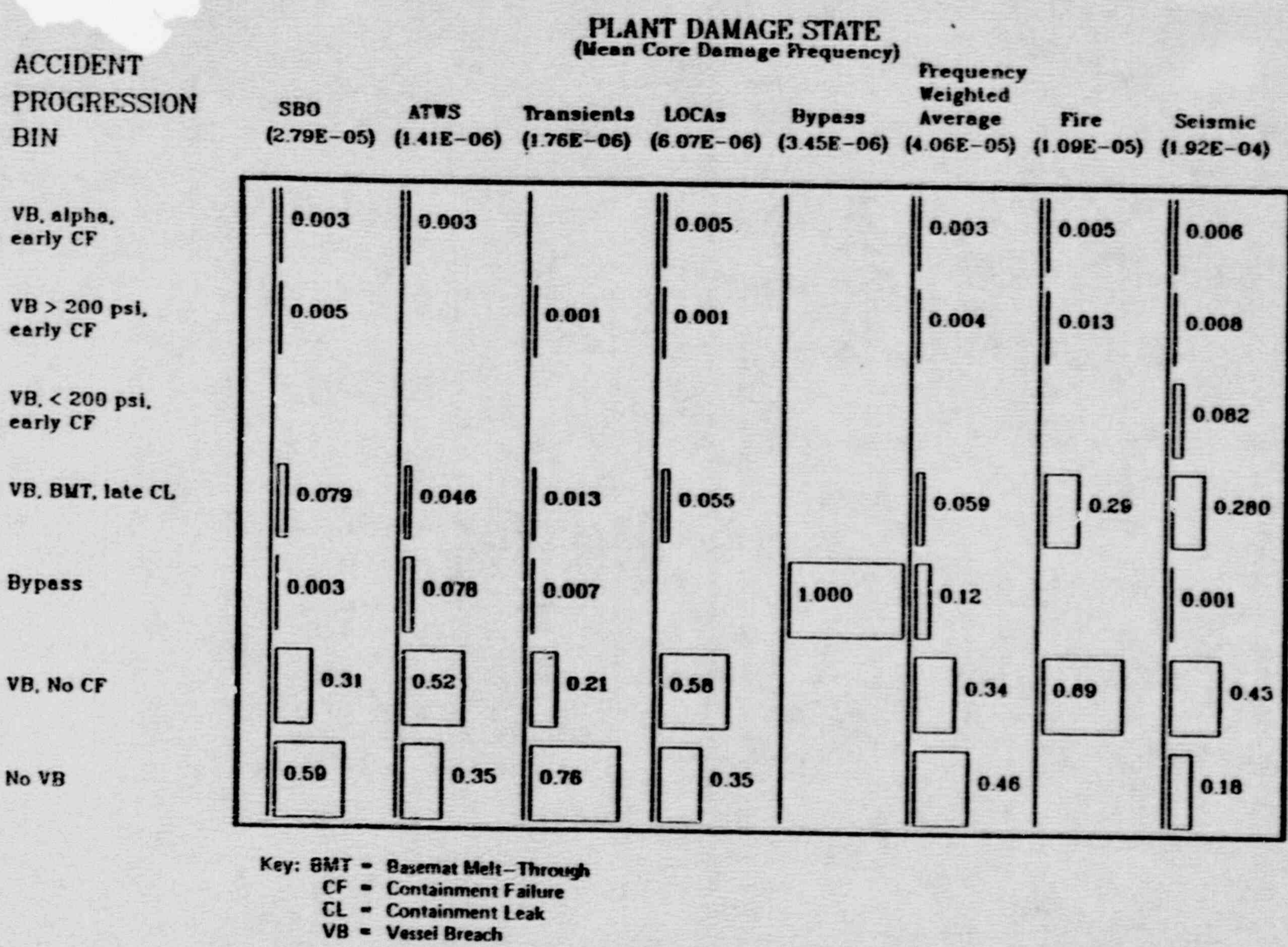


Figure 3.3. Conditional Probability of Accident Bins at Surry

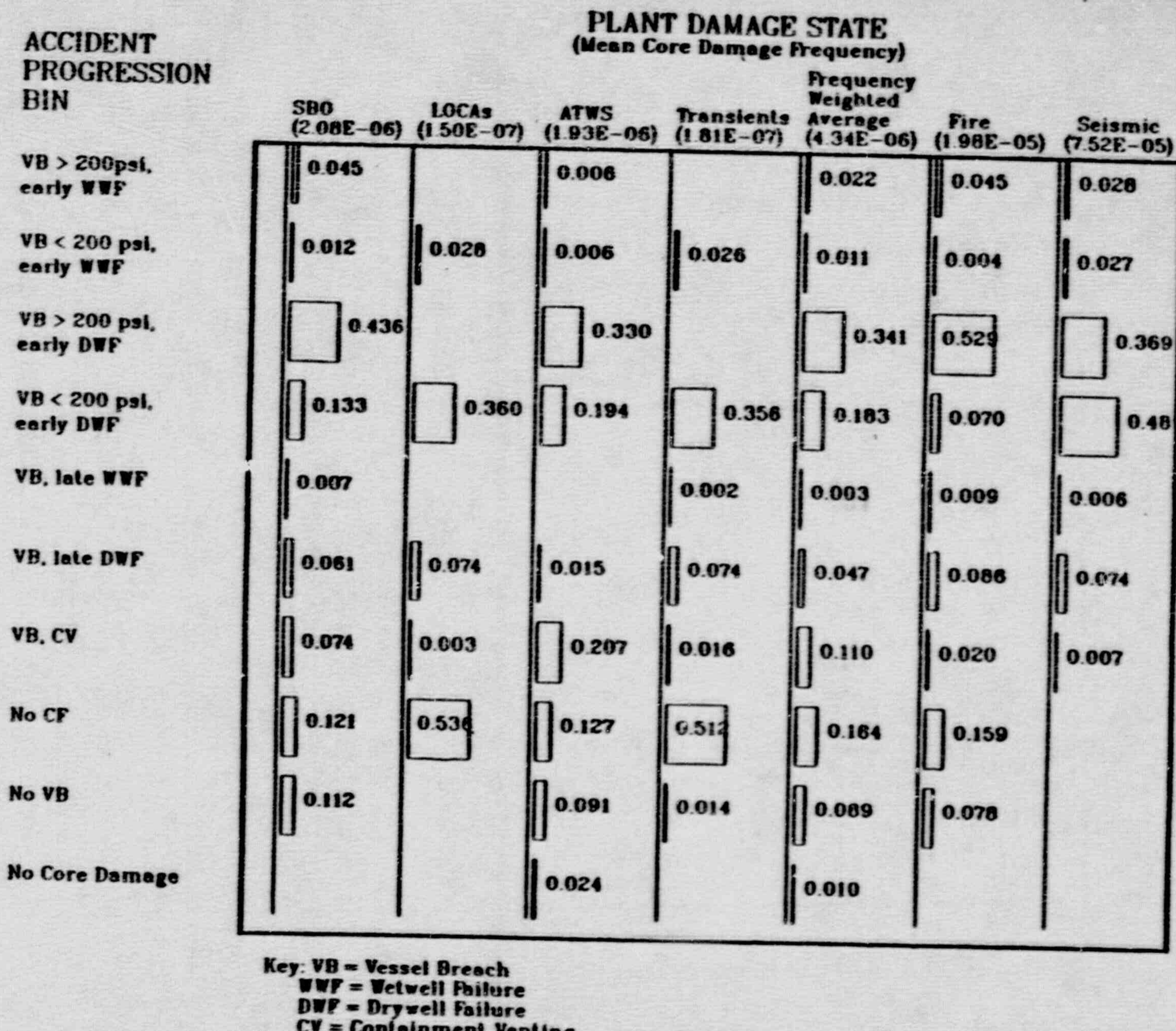


Figure 3.4. Conditional Probability of Accident Progression Bins at Peach Bottom

For Peach Bottom, NUREG-1150 predicts that ECF is principally caused by meltthrough of the drywell wall as found in the initial draft of NUREG-1150, with smaller contributions from drywell head lift due to drywell overpressure and drywell wall failure due to direct containment heating. ECF at Grand Gulf was predicted to be primarily by hydrogen combustion and to pedestal failure from ex-vessel steam explosion in combination with drywell pressure at vessel breach to a lesser degree. The chances were almost 50/50 that hydrogen combustion would fail the Grand Gulf drywell, as well as containment, leading to a substantial bypass of the containment suppression pool.

Another important result of the containment performance analysis is the prediction for PWRs that no vessel breach and no containment failure would occur for a high percentage of the time for the important damage states. For the BWRs, the conditional non-failure probability of containment failure was low.

To understand why such large differences were found in the containment performance of the PWR and BWR requires a careful examination of the evidence supporting the above observations. Although the revised draft of NUREG-1150 contains two sections that were intended to provide insights into the more important issues and results (Section 9 and Appendix C), these insights into the importance or perceived importance of some of these accident progression bins are missing. Rather Section 9 of NUREG-1150 reports the results that are most risk significant without the benefit of the expert rationale or the evidence that support the results. Since some of the conclusions drawn are very important for all plants, it is necessary that the insights be included. Evaluation similar to Appendix J of the initial draft of NUREG-1150 is needed. Appendix C attempts to provide some of the insights into selected issues placed before the expert panels. But each discussion of the expert panel results and the rationale supporting these results provided in Appendix C was brief and for some issues covered only selected parts of the issue. For example, hydrogen generation in-vessel, which is the key issue for the Grand Gulf analysis, was not included in the discussion and thus leaves no convenient way to evaluate the basis of the expert opinion. As another example, the meltthrough distribution for the Peach Bottom drywell wall results from the averaging of six expert votes, two votes against meltthrough ever occurring and two votes for meltthrough always occurring. Although this example is discussed briefly in NUREG-1150, Appendix C, the needed insights currently must be sought in the supporting technical reports when they become available, and they will have to be developed by the reader.

Other comments are offered as follows:

- As observed in the results section of this report, the early containment failure rate conditional on the plant damage states was very low for the large dry PWR containments, slightly larger for the ice condenser containment and high for the BWR containments. These large differences center around a few key issues that currently have no substantial base of experimental results and were treated by the expert polling process. Since there is a lack of data on the more controversial issues, the panels tended to develop a wide range of distributions. When probability distributions are developed that cover wide ranges of values (over several orders of magnitude), the 95th percentile bound on the distribution will carry significantly more weight as the mean will be close to it although the median value may be far below the mean (i.e. one vote at 90 percent failure probability is worth as much as nine votes at 10 percent).

- Although the BWR containment failure rates are perceived to be much higher than the PWR, the core damage frequency is much lower for the BWRs. Thus the probability of an accident involving fission product release is actually less for the BWRs. Although, containment performance is a highly important consideration, judgment must be tempered by overall performance. The focus on the BWR containment as an unacceptable severe accident barrier loses the perspective that the NUREG-1150 BWR plants have as good or better overall severe accident performance as the three PWR plants.

- The high rate of early containment failure predicted for Peach Bottom, a BWR, is caused almost entirely by the opinion that the Mark I Drywell wall will meltthrough when contacted by core debris.

Whether substantial debris/wall contact occurs is dependent on a number of factors including the amount of molten core that is ejected from the reactor vessel and the rate at which it enters the drywell. Whether a small amount of debris or an amount sufficiently large to achieve wall contact would be initially released at vessel failure is not known. There is a paucity of information available to support expert opinions.

- Another Peach Bottom early failure mode is based on the assumption that high pressure at vessel breach will fail the drywell before the Mark I containment vent system can clear and allow steam to condense in the suppression pool. Since the vent clearing time is only of the order of a second and the initial vessel breach requires some time to discharge the vessel contents, it is hard to justify this failure mode. Section 9 of the NUREG-1150 report does not provide the rationale behind this failure mode, and the supporting documentation of the expert panel is not yet available.

- The Grand Gulf early containment failures are based on hydrogen combustion for every important accident scenario. Again, NUREG-1150 does not identify why the expert panel results lead to failure of the containment and drywell about 40% of the time in which core melt has occurred (about 80% of the time for late SBO and ATWS). Or the drywell would be expected to be steam inerted in cases of transients. The results imply that the amount of hydrogen is large in every sequence, yet, in sequences such as station blackout, large amounts of hydrogen are not always necessarily produced. Presumably, the hydrogen igniter system would not function in station blackout sequences. Why does it not work in non-blackout sequences? Insights into the hydrogen production considerations made by the panels would be helpful.

- Although the predictions for early containment failure (conditional on core melt) of the Sequoyah ice condenser containment are much lower than the BWR conditional failure predictions, failures were nevertheless identified with about a 0.07 probability of failure upon core melt. One of these failures involves the seal table guide plate during a high pressure vessel breach and blowdown which subsequently allows substantial amount of core debris to contact the containment wall and melt through. Since this plate is designed for high pressure and the period for passage of entrained debris is very short, it might be expected that this failure mode is highly unlikely. What were the considerations that contribute to the failure mode, and how do they translate to other plants with similar situations? Such insights are needed in the report.

In summary, the perspectives provided in chapter 9, "Accident Progression", and Appendix C, "Issues Important to Quantification of Risk", identify the various failure modes that the analysts and the expert panels considered, but the insights needed to place the results in perspective and to utilize the results elsewhere are missing.

d. Source Term Analysis

Source terms describe the quantity, type and timing of release of radioactive material to the environment. In NUREG-1150 they are expressed as fractions of the core inventory of nine specific radionuclide groups which are released to the atmosphere. Thus, the term release fraction is sometimes used interchangeably with source term. As shown in Figure 3.1 a source term calculation was performed for each accident progression bin. Because the thousands of source terms included in the analysis of each plant represented too large a number for further analysis, they were group into source term clusters for input to the offsite consequence analysis segment of the study.

(1) Source Term Analysis Methodology

The NUREG-1150 undertaking resulted in the need to characterize thousands of source terms associated with the tens of plant damage states, hundreds of containment end states, and the variations in the source term phenomenological issues which are included in the propagation of uncertainties. Parametric source term models were collectively referred to as the XSOR Codes. One such code was utilized for each plant, e.g., SURSOR for Surry, SESOR for Sequoyah, etc.

In addressing the source term methodology employed, NUREG-1150 (page 2-17) states, in part:

> "Because of the complexity and cost of radioactive material transport calculations performed with detailed codes, the number of accidents that could be investigated with these codes was rather limited. Further, no one detailed code available for the analyses contained models of all physical processes considered important to the risk analyses. Therefore, source terms for the variety of accidents of interest were calculated using simplified algorithms. The source terms were described as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of accident progressions, and nine classes of radionuclides. The release fraction at each stage of the accident and for each pathway is determined using various information, such as predictions of detailed mechanistic codes, experimental data, etc.
>
> Release terms are divided into two time periods, an early release and a delayed release. The timing of release is particularly important for the prediction of early health effects."

The XSOR models are essentially mass balance equations, which employ factors used to describe the fraction of the mass of a given fission product group located in the reactor core, the reactor coolant system, the containment, etc., in the analysis of a specific accident scenario. In the NUREG-1150 process, numerical values of these factors are obtained both from the expert opinion polling process and from the NUREG-1150 contractor staffs. Because it is a substantial simplification of complex processes and phenomena, the XSOR; approach allows the calculation of large numbers of source terms quickly; however, these parametric models do not reflect the details present in state-of-the-art source term analyses. Thus, this parametric approach to quantifying source terms raises some fundamental questions.

(2) Source Term Analysis Results

Source term analysis results are presented in several ways in NUREG-1150. Some of the more significant results are reproduced here.

Figure 3.5, which is a reproduction of Figure 10.1 in NUREG-1150, includes curves depicting the mean frequency of exceeding specified release fractions of the core inventory of iodine, cesium, strontium and lanthanum for each of the five plants studied. These data are for internal initiated events only.

Figure 3.6 presents a comparison of the iodine, strontium and lanthanum release fractions for ECF or containment bypass. Source terms for accident sequences involving early containment failure or bypass dominate the offsite risks. The central tendencies of these source terms can be summarized as follows: median values of release fractions of iodine for the five plants studied are a few percent to slightly more than ten percent of the available core inventory. The mean values for iodine, in general, are approximately twice the medians, indicating that the central tendencies are not greatly affected by the extremes of the distributions. By contrast, the mean release fractions for strontium and lanthanum, in general, are larger than the medians by an order of magnitude or more, indicating that a few cases with relatively large release fractions, compared to the medians, greatly affect the mean values. It should be noted, however, that the data in Figure 3.6 pertain to the range of release fractions, not their frequency of occurrence per reactor year.

(3) Discussion of Source Term Analysis

The source terms reported in NUREG-1150 and the resultant offsite consequence analyses should be considered as approximations, due to the reliance

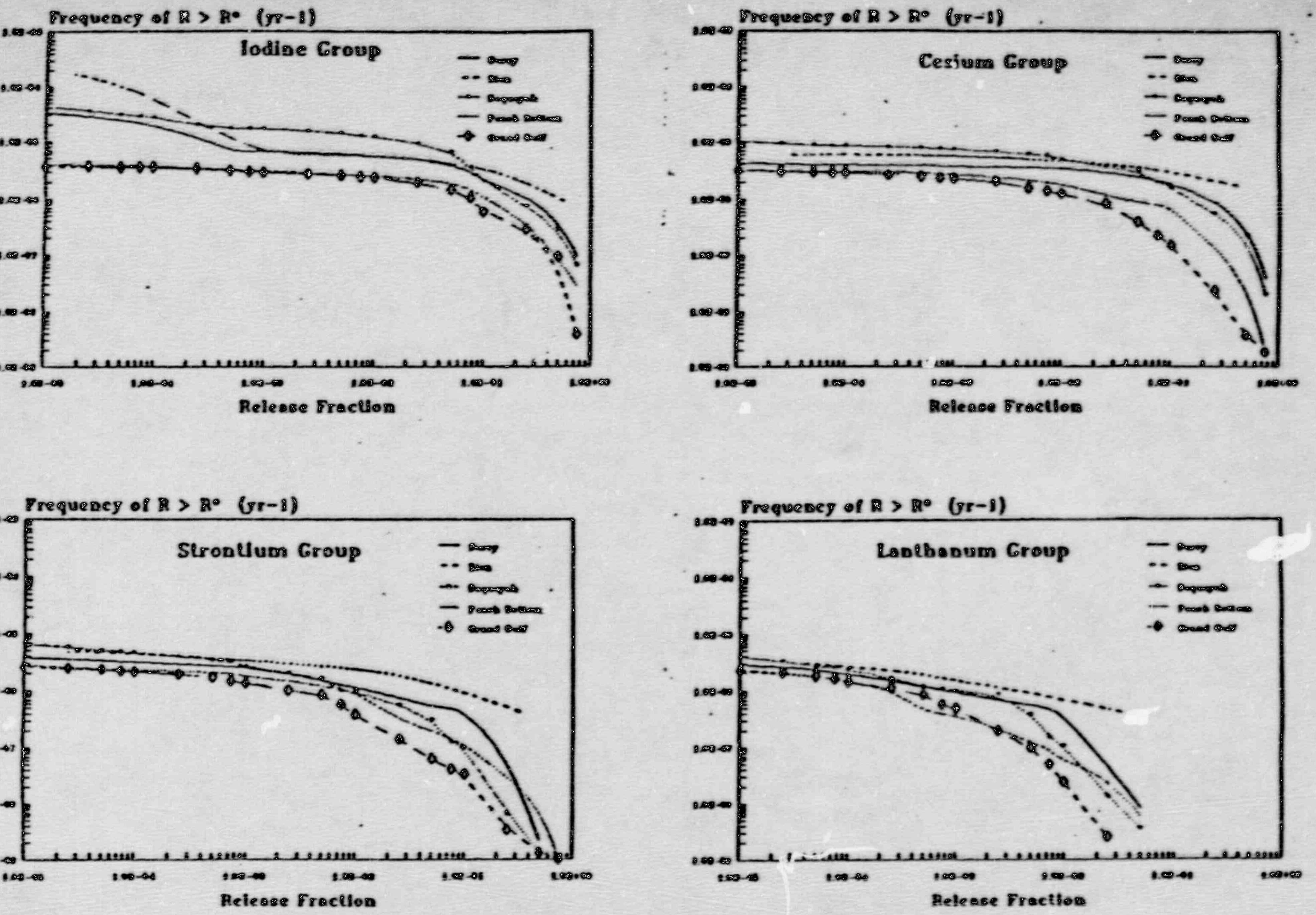
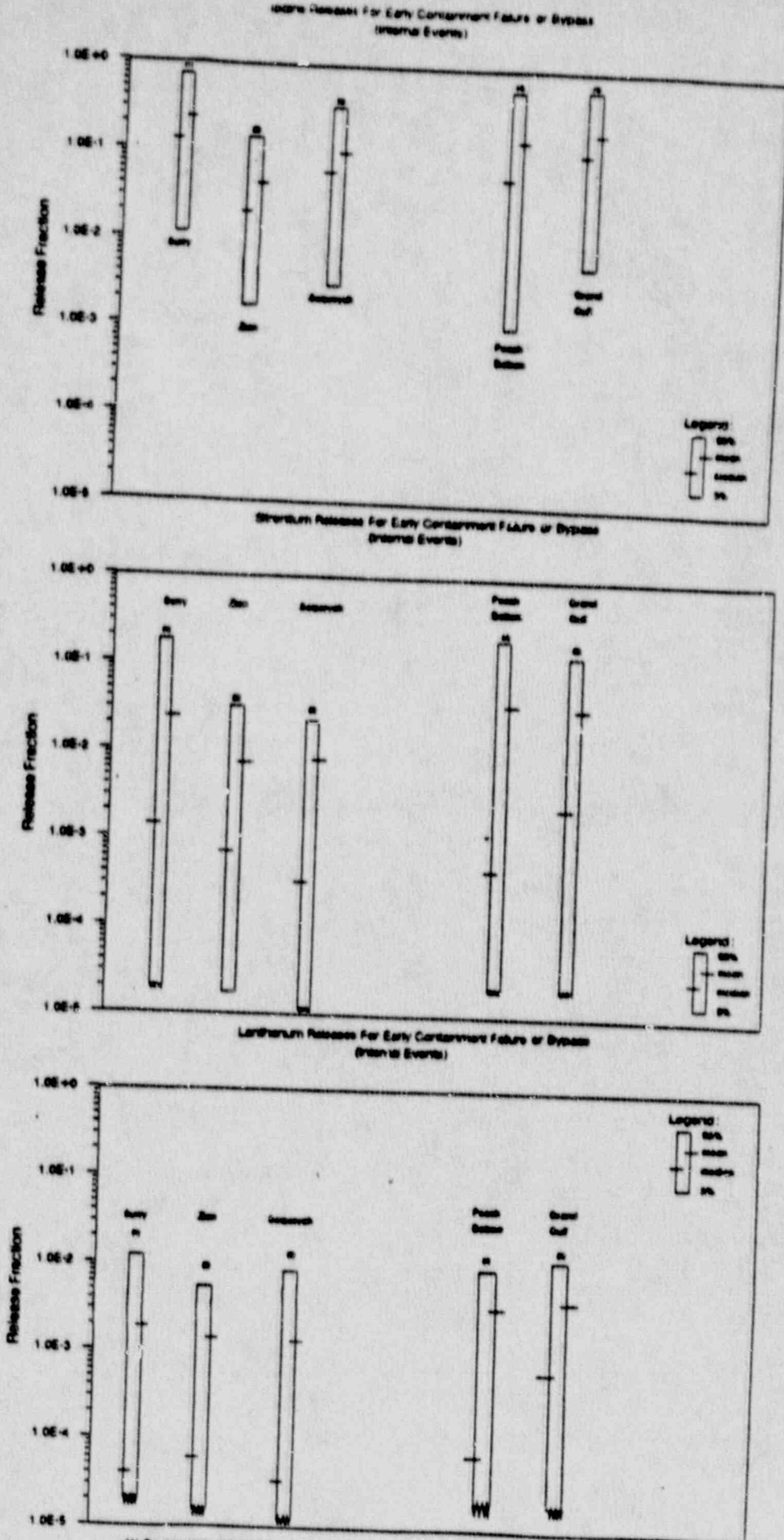


Figure 3.5 Frequency of Release for Key Radionuclide Groups



(1) Source term distributions for containment bypass at Surry.
(2) Source term distributions for early failure at Zion.
(3) Source term distributions for early failure at Sequoyah.
(4) Source term distributions for early failure in drywell at Peach Bottom.
(5) Source Term distributions for early failure with drywell failed and sprays unavailable at Grand Gulf

Figure 3.6 Iodine, Strontium and Lanthanum Releases for Early C... Failure or B...

on the simplified mass balance XSOR models used to produce large numbers of source terms. Moreover, the source terms for the risk dominant containment bypass sequences, for the PWRs studied, may be substantially overstated because the input to the XSOR models of these sequences did not reflect plant features and system modeling which have the effect of reducing releases to the environment.

For example, for the largest risk contributors for PWRs, the steam generator tube rupture (SGRT) sequences, which account for more than 90% of the median frequency of the bypass sequences for Surry, it was assumed that there would be no fission product retention in the secondary system, (i.e., the steam generators, steam lines, relief lines and safety relief valves (SRVs)) for the 75% of the cases in which it was assumed that the safety relief valves stuck-open. This was assumed despite the availability of auxiliary feedwater until the condensate storage system is exhausted for the very small (S_3) break size analyzed.

For the SGTR sequences with the largest source terms it was assumed that operators would fail to follow procedures and depressurize the reactor coolant system in a timely fashion. No emergency core coolant injection was assumed, despite the fact that releases to the environment did not occur for more than 10 hours in most cases. The operators are trained specifically to respond to SGTR accidents, and recent analyses indicate that much more than ten hours would elapse prior to releases for the conditions analyzed in NUREG-1150. Thus the modeling of both the systems and the fission product transport appear to be quite conservative.

The following general perspective on severe accident source terms from NUREG-1150 is found on page 10-1.

> "The variation in source term estimates associated with phenomenological uncertainties can vary by many orders of magnitude. Thus, the range of uncertainty in source terms tends to mask the variation that occurs because of actual differences in accident progression behavior

among the plants or among different accident progression event tree pathways within a plant."

The Committee's perspective is quite different than that quoted above in that we observe that the range of uncertainty in source terms reported does not tend to mask the variation that occurs in the other portions of the analysis in NUREG-1150. For example, Figure 3.7 shows the progression of the uncertainty ranges for early releases for Surry and Zion, including:

- Systems Issues Only (i.e., Core Damage Frequency)
- Systems and Containment Issues
- Systems, Containment, and Source Term Issues

Although essentially the entire core damage frequency distribution for Zion falls above the comparable distribution for Surry, as indicated by the two bar graphs to the left in Figure 3.7, when the systems and containment issues are combined, as indicated by the bar graphs in the middle of this figure, the corresponding Zion and Surry distributions are quite comparable. Moreover, the range of uncertainty from the median to 95th percentile is not appreciably different from that for the systems only issues, i.e., no additional uncertainty is added to the upper half of the distribution. When the source term uncertainties are included, as is the case in the two bar graphs to the right in Figure 3.7, the range of uncertainty is virtually the same as for the two previous sets of bar graphs. In other words, the range of Fig. 3-7 uncertainty in source term does not mask the variation that occurs because of actual differences in accident progression behavior, at least for the more important upper half of the distribution for large releases, e.g., 10 percent or more of the iodine core inventory.

Section 10 of NUREG-1150 includes a list of general perspectives related to source terms. In general, the Committee's perspectives related to source terms differ substantially from those stated in Section 10. Our comments in this regard are included above in Section 2, Findings and Recommendations, and in Section 4, Comparison with Reactor Safety Study (WASH-1400).

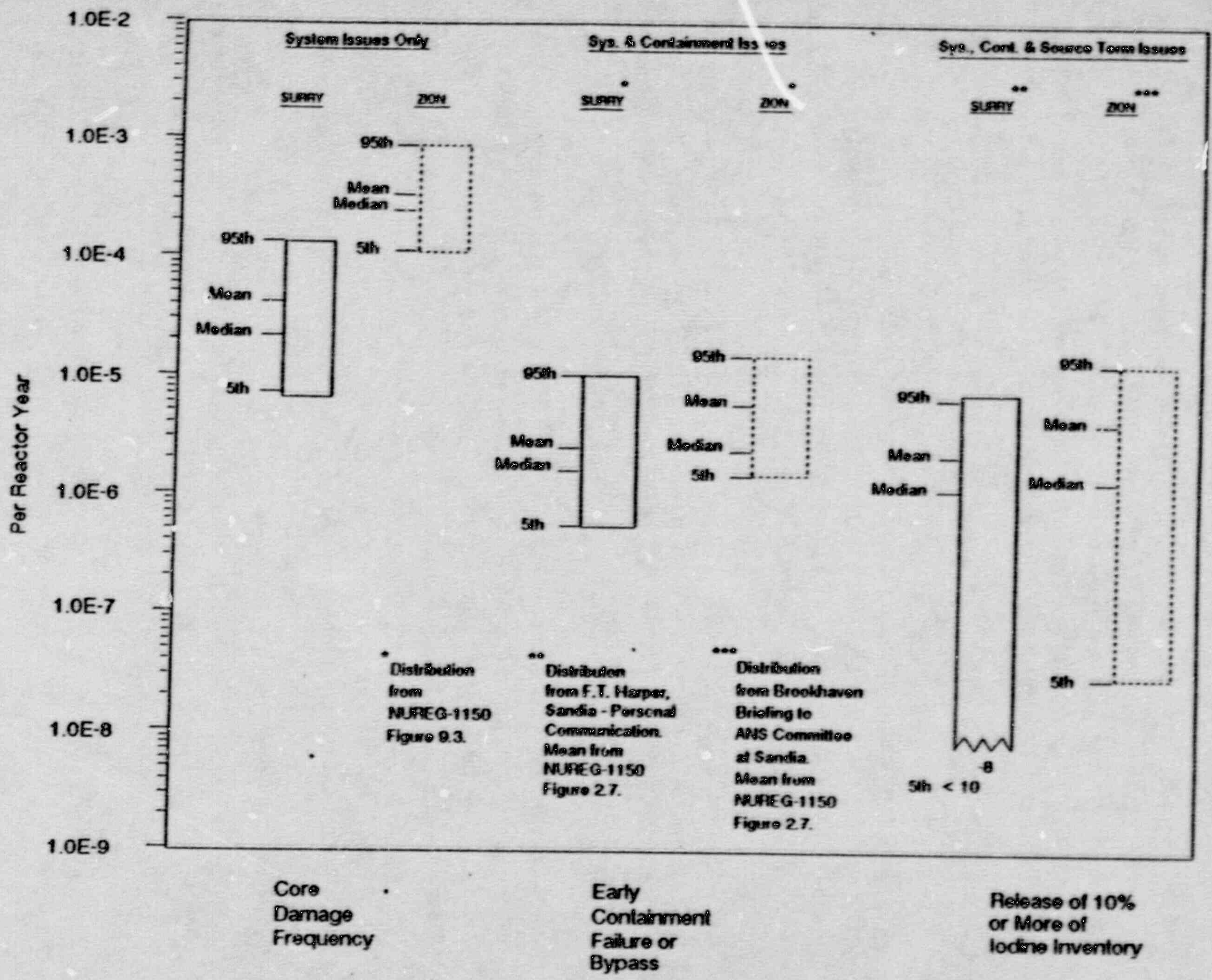


Figure 3.7 Results of Analysis Progression for Surry and Zion Early Releases (Internal Events Only)

e. Offsite Consequences and Public Risk Estimates

The term "offsite consequences" deals with the impact of radioactive releases to the environment and the population. The impact is manifested as early and delayed health effects to people, loss of access to land (because of contamination), and corresponding economic losses. A particular set of offsite consequences is the result of a corresponding set of source term values.

(1) Methodology for Offsite Consequences and Public Risk

In the analysis of offsite consequences the inventory of radioactive materials before the postulated accident, was determined using the SANDIA-ORIGEN code. The input to the code consisted of information related to the power history and the refueling cycles of each plant. For the source term, the radioisotopes were placed into nine groups according to similar chemical behavior: noble gases, I, Cs, Te, Sr, Ru, Ce, Ba, and La. The magnitude of the release for each isotope was determined by multiplying the release fractions, included in the source term clusters, and the core inventory for each radionuclide group.

The radioactive materials released as a result of the postulated accident were transported in the air for a large number of weather conditions (about 2500), using the code MACCS version 1.5 (MELCOR Accident Consequence Code System).

MACCS is the successor to CRAC-1 and CRAC-2. CRAC-1 (Calculation of Reactor Accident Consequences) was the code used in WASH-1400; CRAC-2 and MAACS were used in the initial NUREG-1150 draft. The input to MAACS, relative to releases, came from the XSOR results. The calculation included a determination of amounts of each group deposited on the ground for various distances downwind from the plant, up to a total distance of 1000 miles.

In MACCS, the radiation dose at a specific distance from the plant, and for a particular time period, is determined based on the assumption that a human is standing at the particular distance and receives a radiation dose from the contaminated air in which he/she is immersed, from material deposited on the ground, from the air that person breathes, from drinking contaminated water, and from eating contaminated food. The transformation of particle flux into dose was based on references A.61 (D. C. Kocher), A.62 (ICRP 26, 1977), and A.63 (ICRP 30, 1978).

Dose mitigation was included in the calculation by applying emergency response actions. It was assumed that 99.5 percent of the population within the 10-mile emergency planning zone (EPZ) was evacuated at a speed determined by the plant's evacuation plan. The rest of the population was assumed to be relocated within 12 to 24 hours after the plume passage. Based on the dose received, early fatalities were calculated using the risk-effect relationship given in reference A.65 (NUREG/CR-4214, SAND85-7185, 1985). Early fatalities are expected to occur within one year after the accident. The estimate of latent cancer deaths was based on data presented in reference A.66 (the BEIR-III report). Latent cancer fatalities are those expected to occur during the lifetime of the exposed individuals. The NRC staff did not carry the evaluation of uncertainties into this segment of the study but did perform sensitivity studies on evacuation models.

(2) Offsite Consequence and Public Risk Results

The detailed results of the consequence analyses appear in NUREG-1150, Figures 3.8-3.18 (Surry), 4.8-4.18(Peach Bottom), 5.7-5.12 (Sequoyah), 6.7-6.12 (Grand Gulf), and 7.6-7.11 (Zion). Tables 11.1 and 11.2 present cumulative results. NUREG-1150 Figures 12.1, 12.6, 12.10, and 12.11, which represent comparisons of cumulative risk, individual risk, tentative safety goal, and effects of emergency response options respectively are reproduced here as Figures 3.8 through 3.11. The definition of large release is given in Figure 3.10. These figures clearly show that the risks resulting from the postulated accidents are within the safety goals set by the NRC, and the results depend on the site characteristics and emergency response effectiveness of each plant.

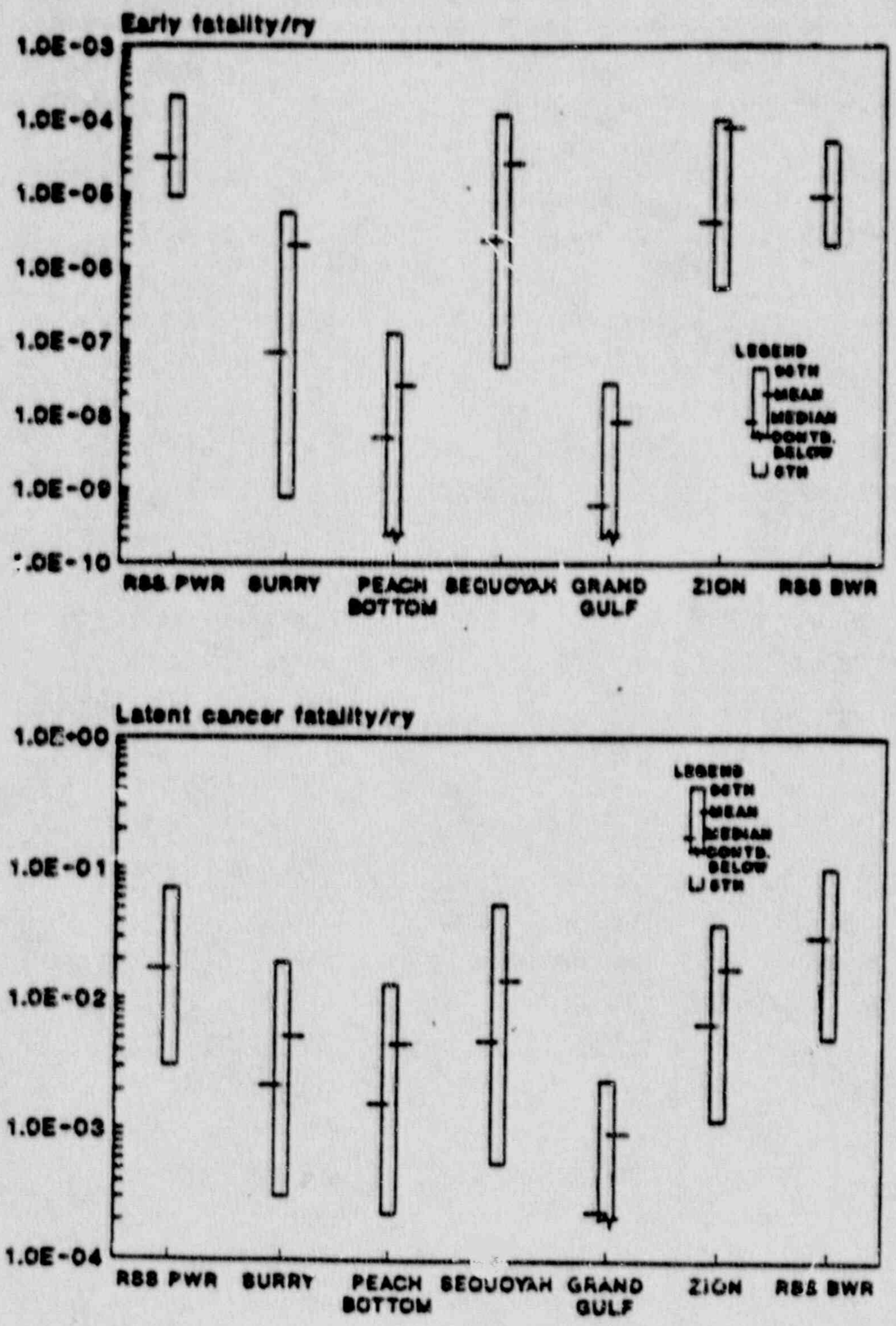


Figure 3.8 Comparison of early and latent cancer fatality risks at all plants—internal initiators.

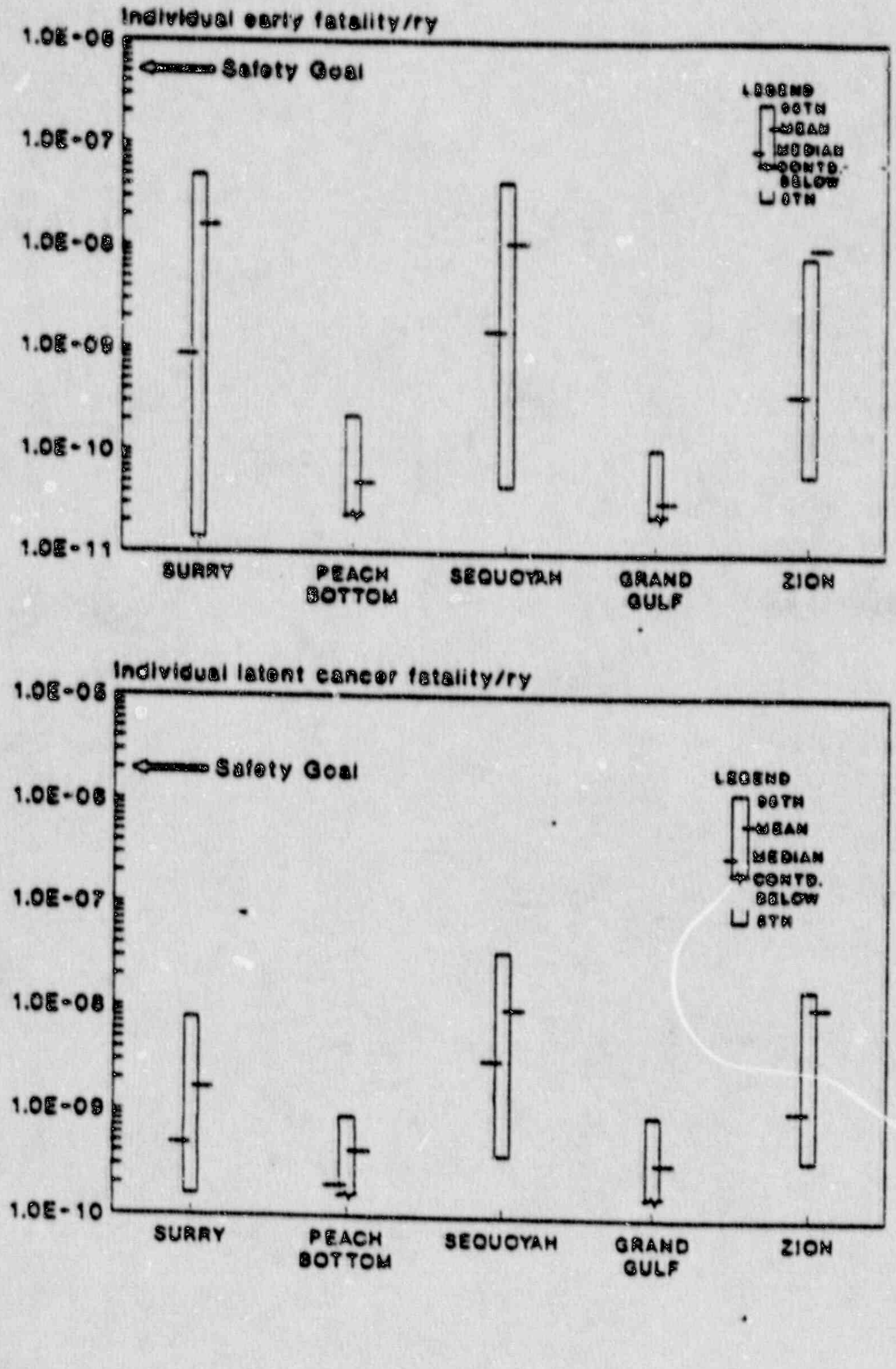


Figure 3.9 Comparison of individual early and latent cancer fatality risks at all plants–internal initiators.

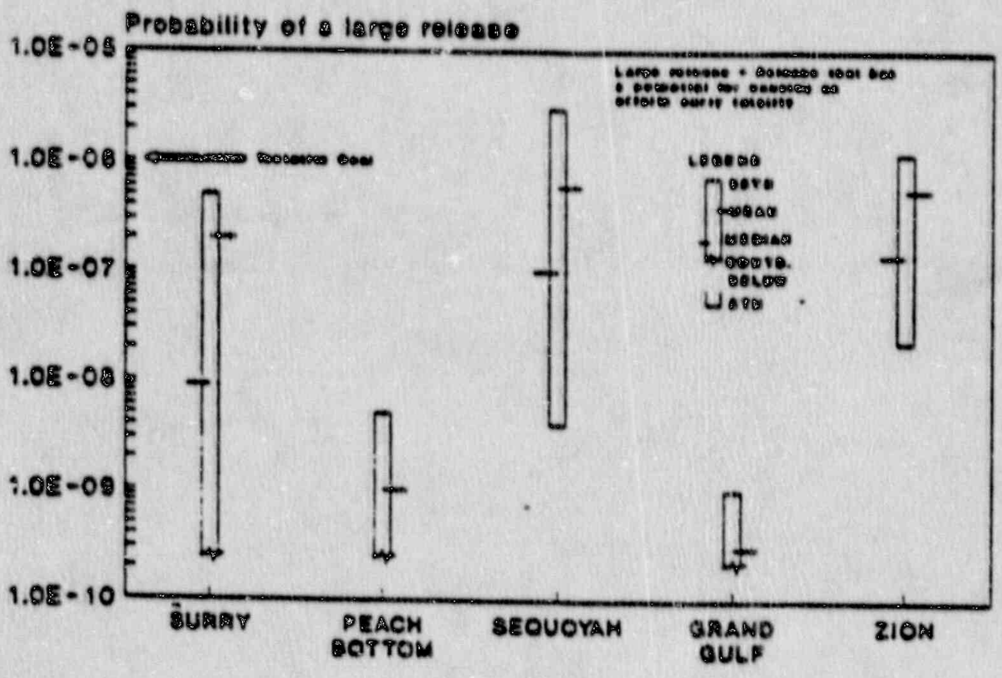
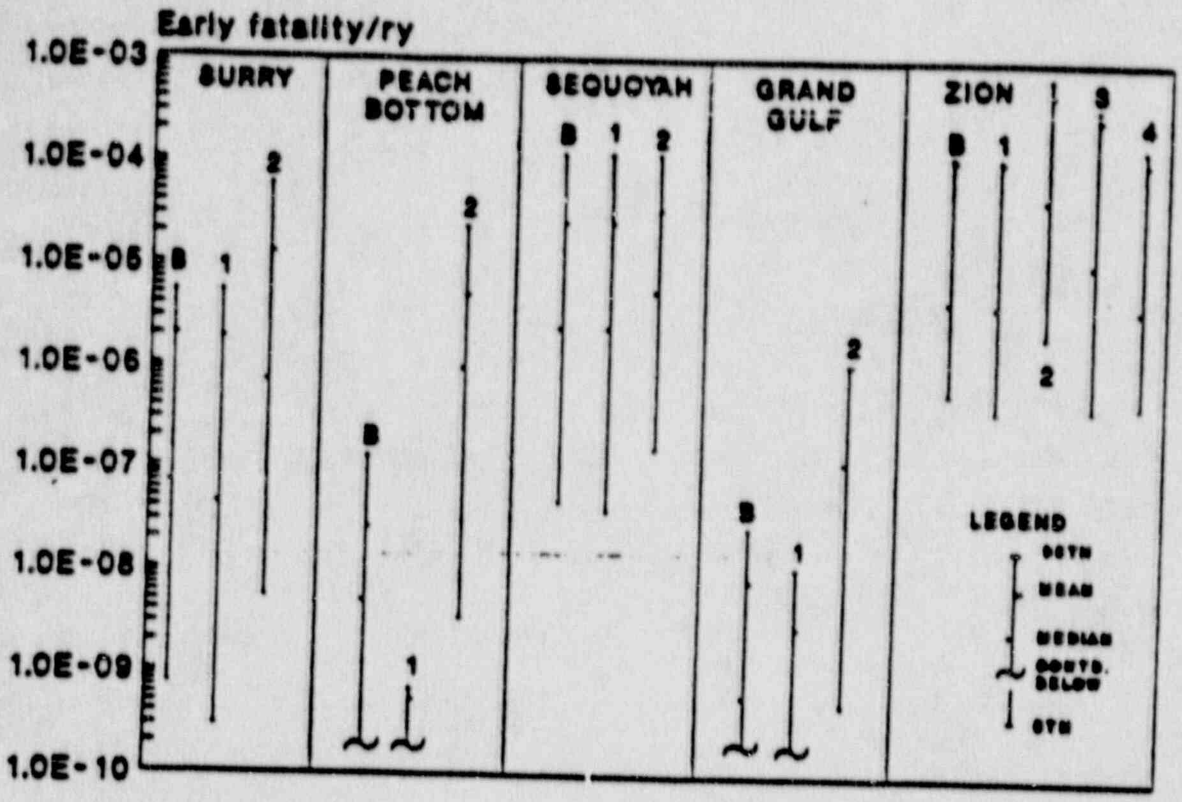


Figure 3.10 Comparison with tentative safety performance guideline—internal initiators.



BASE CASE (B)

99.5% Evacuation from 0 to 10 miles

EMERGENCY RESPONSE OPTIONS (1 TO 4)

1. **100% Evacuation from 0 to 10 miles**
2. **0% Evacuation with early relocation from high contamination areas**
3. **100% Sheltering**
4. **100% Evacuation from 0 to 5 miles, and 100% sheltering from 5 to 10 miles**

Figure 3.11 Comparison of effects of emergency response options on early fatality risks at all plants—internal initiators.

The results cannot be compared by simply reading Tables 11.1 and 11.2 or the corresponding graphs for the five plants studied. For deeper understanding of the results, one needs to read carefully the accident scenarios and the site-dependent assumptions and conditions.

(3) Discussion of Public Risk Analysis

The frequency distributions of the consequences are presented in terms of complimentary cumulative distribution functions (CCDF). The CCDF gives the probability that a certain value of the variable will be exceeded. For each case studied, 200 CCDF's were computed corresponding to different weather conditions. Because plotting 200 CCDF's would be cumbersome and probably confusing, only four statistical measures are plotted for the 5th percentile, the 95th percentile, the median, and the mean.

To make it clear how each curve was produced, consider the generation of the mean curve for Latent Cancer Fatalities (LCF) in Fig. 3.12. For 100 LCF's, there are 200 values of the frequency for which the 100 LCF's are exceeded. Since all 200 values are equally likely, the 200 frequency values are summed and divided by 200 to give the mean value for 100 LCF's. This process is repeated for 1 LCF, for 10, 15, 105 etc. and the points for the mean curve are thus obtained.

One should read these graphs as follows. Again look at Figure 3.12 and consider the occurrence of 10 LCF's. The median frequency of exceeding 10 LCF's is about 3.5×10^{-6} per reactor year. In 95% of the cases, the frequency that 10 LCF's will be exceeded is less than about 2×10^{-5} per reactor year. In 5% of the cases the frequency that 10 LCF's will be exceeded is less than about 10^{-6} per reactor year.

It is also interesting to note from Figure 3.12 (Fig. 3.8 of NUREG-1150), that above about 150 early fatalities, the mean value exceeds the 95% percentile. This means that the mean beyond that point is determined by less than 10 observations out of 200.

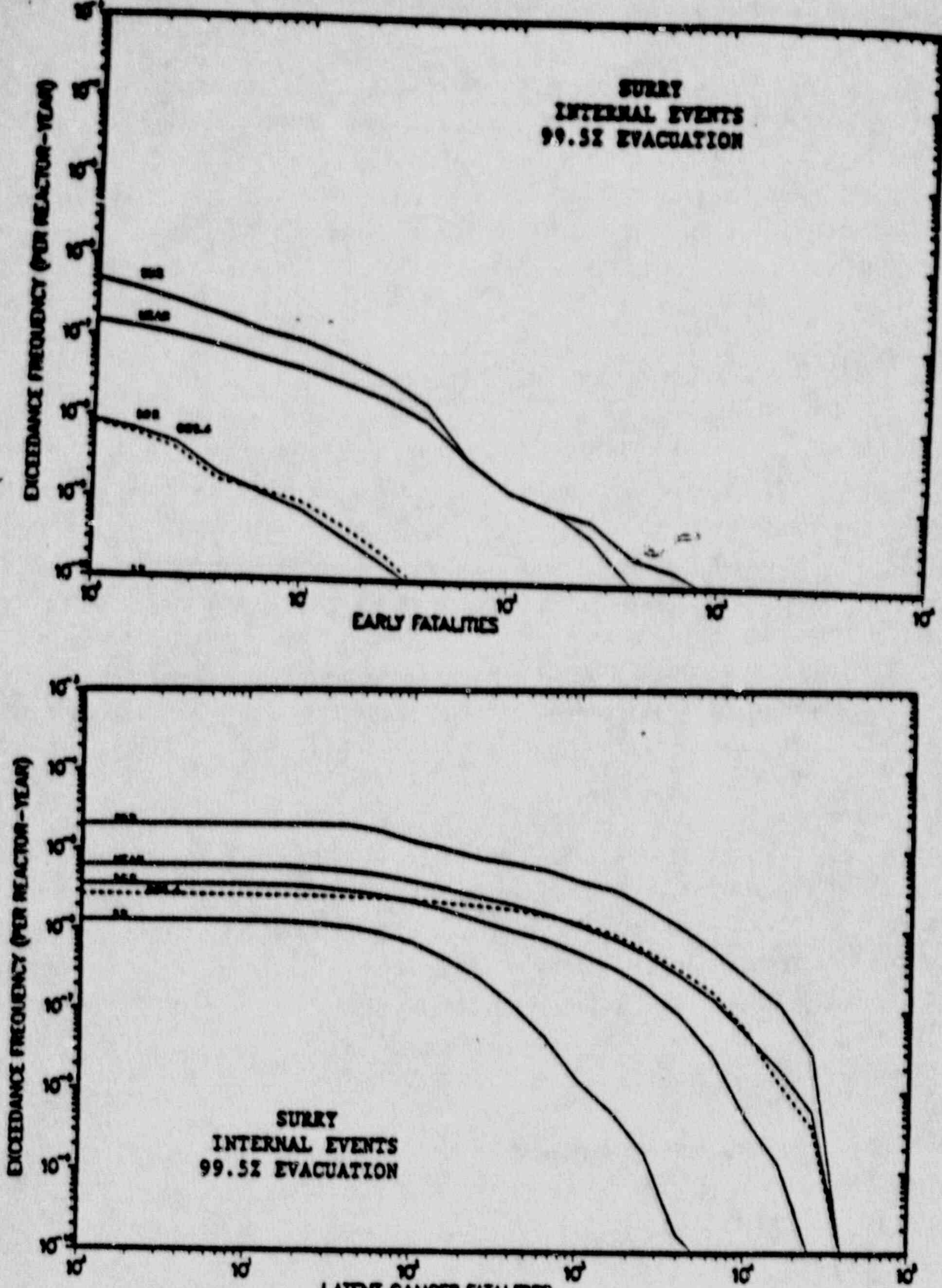


Figure 3.12 Computed curves showing four statistical measures of 200 CCDFs for Surry for early fatalities and latent cancer fatalities. (The CCDF for Observation 4 is also shown.)

The results depend strongly on the plant site through (a) local meteorological conditions, (b) population density and distribution around the plant, (c) evacuation plans (as provided by the plant), and effectiveness of emergency response. Comparison with the results of the Reactor Safety Study is not strictly valid for the reasons explained in Section 4.

f. <u>External Events Analysis</u>

(1) Methodology for External Events

The revised draft of NUREG-1150 includes an assessment of the risks due to external events for two nuclear power stations, Surry (3-loop PWR) and Peach Bottom (BWR with Mark I containment). These are the plants that were used to typify the two major reactor designs in WASH-1400. Although a complete range of external accident initiators was considered (e.g., tornadoes, flooding and aircraft crashes), only two were found to contribute in a significant way to the possibility of core damage, fires and earthquakes. A cutoff value for initiating event frequency of 10^{-6} per reactor year was apparently used to screen out unlikely contributors (page A-22).

Core damage frequencies due to fire at Surry and Peach Bottom were calculated with the same level of detail as for internal initiators. Fire probabilities were determined for various plant zones. Once started, fires led to equipment failures which were analyzed using the same event trees as for the internal initiators. Such actions as the likelihood of successful intervention of the fire brigade or fire suppressant systems were modeled. Human performance, however, was assumed to be degraded in fires.

The frequency of core damage due to earthquakes was also calculated using event trees similar to those created for internal initiators. The first step in an individual seismic analysis was to select a peak ground acceleration from a seismic hazard curve for the site, and then to subject the plant to an assumed time-history-of-motion anchored at this acceleration level. This procedure produced seismic motion values at individual component locations in the building.

Seismic fragility data were used to compute the likelihood of resultant equipment failures, leading hence into the accident event tree. Loss of offsite power was an almost sure result of every significant seismic event.

(2) Summary of the External Event Results

Core damage frequencies resulting from fires and seismic events for the two plants analyzed are presented in the revised NUREG-1150, and are summarized in Figure 3-13 (Figures 8.8 and 8.9 of NUREG-1150) along with frequencies for internal initiators. Fire and seismic initiators were also carried through the accident progression phase of the analysis and yielded relatively high containment failure probabilities (Figures 3.4, 3.5, 4.4, and 4.5 of NUREG-1150). Consequence results, such as early deaths and latent cancer death probabilities, were calculated for fire initiators, but not for seismic events. The authors of NUREG-1150 concluded that the magnitude of and large uncertainties in the seismic hazards estimates, and the extent of other forms of societal damage that would result from the very large seismic events which are required to damage nuclear plants, obscure the relevance of nuclear accident consequences (pages 1-3 and 4). Seismic risk (i.e., consequence) calculations are presented as sensitivity studies in the documents underlying NUREG-1150, references 1.20, 1.21, 1.27, and 1.28.

(3) Perspectives from the External Event Analysis

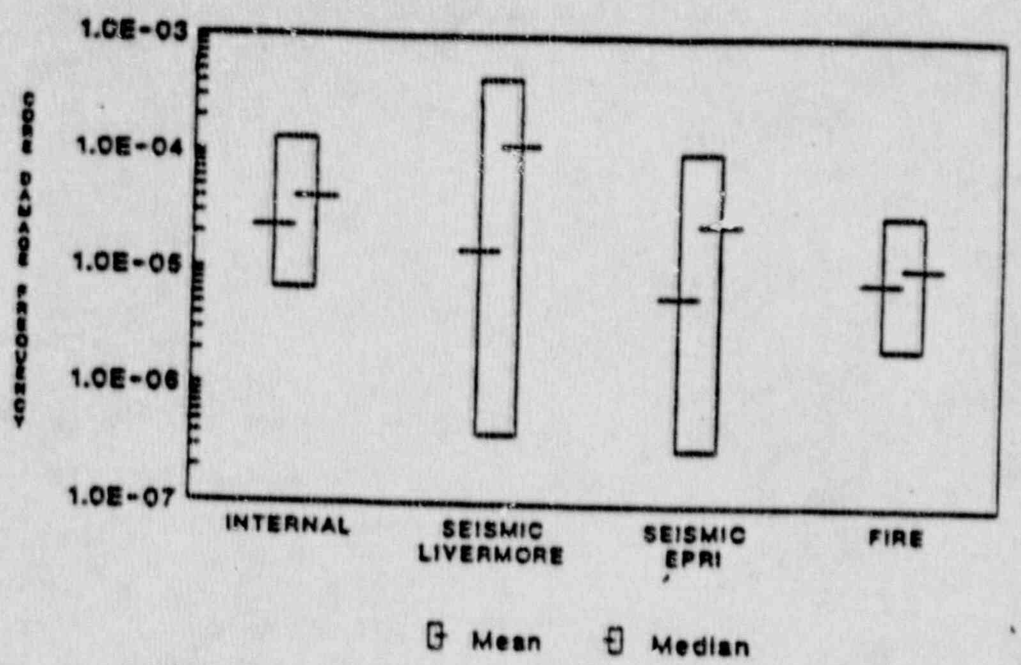
(a) Fire Initiators

The relatively large core damage frequencies for fire initiators are a result of plant data showing a high frequency of fire incidence and the fact that there are locations where damage from a single fire could affect the performance of a number of vital systems, notably switchgear rooms. The fire analysts appear to have made a strong effort to determine actual conditions for fire likelihood at the two plants studied. Although plant experience (gathered from many plants, not just those being analyzed) was used to establish the fire initiation frequency, judgmental factors were used to determine whether a fire, once started, would persist and cause damage in spite of fire mitigation systems and actions. It would seem that the same data base that was used for fire initiation could, and should, have been used to give a more realistic value of fire persistence.

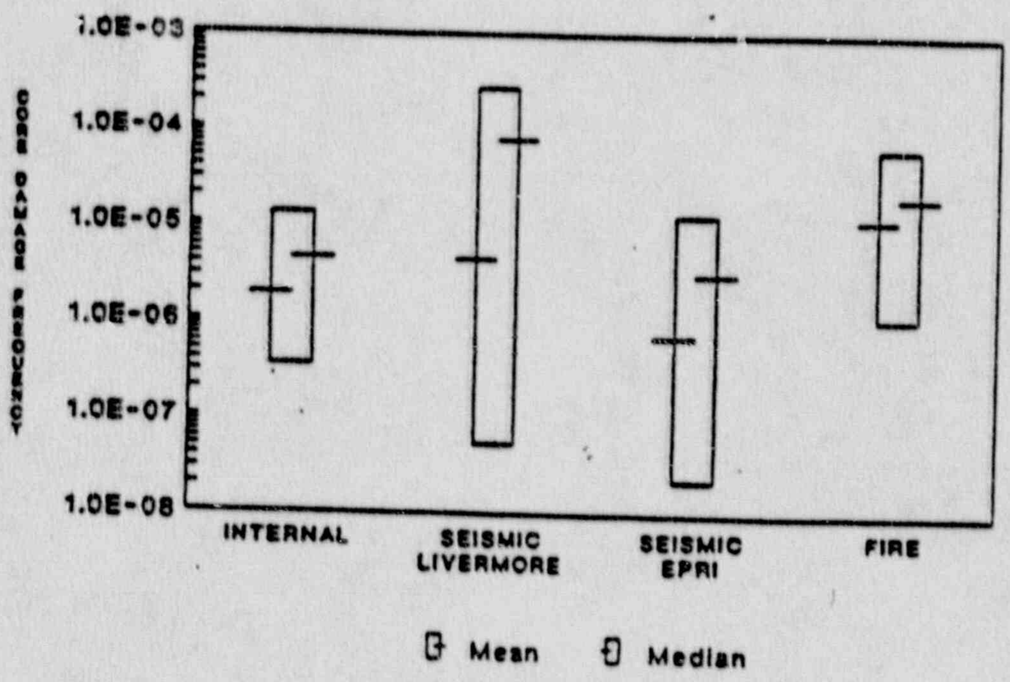
(b) Seismic Hazard Curves

The probability of seismic events of increasing intensity (i.e., ground acceleration values) can be displayed as seismic hazards curves. Two different sets of hazards curves, developed by an expert elicitation process, were used in NUREG-1150, one generated under the auspices of the Lawrence Livermore National Laboratory (LLNL) and one by the Electric Power Research Institute (EPRI). The probabilities for strong ground motion depicted by the two curves are quite different, as evidenced by the differences in the predicted seismic core damage frequencies of Figure 3.13. The probability of an event producing ground accelerations at the Peach Bottom site exceeding about 0.6g given by the LLNL curve, at the 85th percentile (close to the mean), exceeds the EPRI prediction by more than a factor of 10. The LLNL curve is apparently dominated by the judgement of one of the experts who developed it (page C-115). For both seismic hazards estimations approximately two thirds of the core damage frequency results from ground motion in excess of 0.5g. At this level of ground acceleration the uncertainty in the seismic hazards estimation (taken as the difference between the 15th and 85th percentile values), is on the order of a factor of 100.

In addition to the uncertainty in the prediction of the frequency of strong ground motion, there is very little data on the time-history-of-motion or spectral composition of strong eastern earthquakes. There is some evidence that eastern earthquakes would be richer in higher frequency motion than the well-documented typical western event. Western earthquake data provided the basis for spectrum and time-history assumptions used in the NUREG-1150 analysis.



Surry external events, core damage frequency ranges (5th and 95th percentiles).



Peach Bottom external events, core damage frequency ranges (5th and 95th percentiles).

Figure 3.13 (from NUREG-1150, Figures 8.8 and 8.9)

4. COMPARISON WITH REACTOR SAFETY STUDY (WASH-1400)

The U.S. NRC and its predecessor, the U.S. AEC, sponsored the pioneering Reactor Safety Study, which was published in 1975 under the report number WASH-1400. It established the new field of probabilistic risk assessment (PRA) for nuclear power plants, and it has been widely used and referenced throughout the world. NUREG-1150 presents a more recent analysis of the two plants studied in WASH-1400, Surry and Peach Bottom, and includes three other plants as well, Sequoyah, Zion, and Grand Gulf. A great deal of progress in PRA technology and a number of changes in the plants systems have occurred in the intervening fifteen years since completion of WASH-1400. As noted earlier in our report, the Committee believes that NUREG-1150 should supplant the now outdated WASH-1400 study.

In this section, we present the results of a brief comparison of some of the major results of the two studies. This comparison utilizes the results of WASH-1400 and NUREG-1150 as reported, assuming they are accurate, regardless of reservations concerning some of the results expressed elsewhere in our report. We include all five plants studied in NUREG-1150 in the discussion.

a. Core Damage Frequency

Table 4.1 lists the central estimates and ranges of uncertainty in calculated core damage frequency reported in NUREG-1150 and WASH-1400. The core damage frequencies for the Surry and Peach Bottom plants as calculated in the two studies are compared in Figure 8.14 and 8.15 of NUREG-1150, reproduced here in Figure 4.1. The Surry data were misplotted in Figure 8.14, and a correction is included in Figure 4.1. The median value for Surry in WASH-1400 was $6x10^{-5}$yr-1. The comparable NUREG-1150 value is $2.3x10^{-5}$ per reactor year, i.e., a factor of 2.6 lower than WASH-1400. By contrast, the median value for Peach Bottom in NUREG-1150 is a factor of 15 lower than WASH-1400, i.e., $1.9x10^{-6}$ vs. $2.9x10^{-5}$ yr-1. Interestingly, the range of uncertainty from the median to 95th percentile is not greatly different in the two studies for either plant, i.e., a factor of 5.8 and 5.6, respectively, for the Surry analyses in WASH-1400 and NUREG-1150 and a factor of 4.5 and 6.8, respectively, for the Peach Bottom analyses.

TABLE 4.1

COMPARISON OF CORE DAMAGE FREQUENCIES (PER RY) IN NUREG-1150 AND WASH-1400
(INTERNAL INITIATED EVENTS)

	PWRs				BWRs		
	Surry	Sequoyah	Zion	Reactor Safety Study	Peach Bottom	Grand Gulf	Reactor Safety Study
95th	1.3e-04	1.8e-04	8.4e-04	3.6e-04	1.3e-05	1.2e-05	1.3e-04
Mean	4.1e-05	5.7e-05	3.4e-04	-----*	4.5e-06	4.0e-06	-----*
Median	2.3e-05	3.7e-05	2.4e-04	6.0e-05	1.9e-06	1.2e-06	2.9e-05
5th	6.8e-06	1.2e-05	1.1e-04	1.4e-05	3.5e-07	1.7e-07	6.6e-06

* As the frequency distributions in the Reactor Safety Study were essentially log normal, the mean frequencies were approximately three times the median values. (re. N. Rasmussen personal communication).

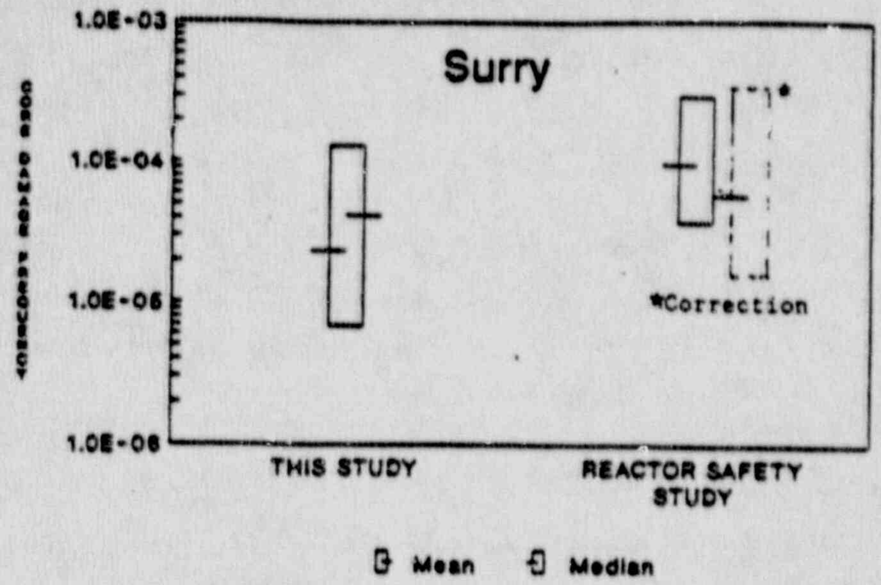


Figure 8.14 Surry core damage frequency range compared to Reactor Safety Study.

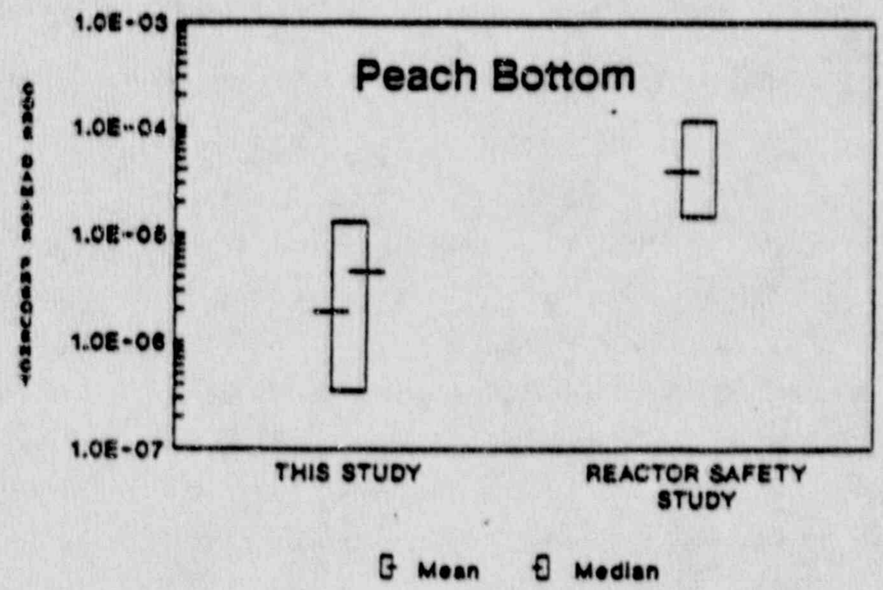


Figure 8.15 Peach Bottom core damage frequency range compared to Reactor Safety Study.

Figure 4.1 Comparison of NUREG-1150 and WASH-1400 Core Damage Frequencies

Section 8.3 of NUREG-1150 cites two important reasons for differences in core damage frequencies. (1) both plants have implemented hardware modifications and procedural improvements, which reduced core damage frequencies, and (2) the state-of-the-art in applying probabilistic risk assessments has advanced significantly in the intervening 15 years between the two studies. In some cases, the new methods have reduced or eliminated previous analytical conservatisms. However, new types of failures also have been recognized as a result of improved investigative techniques.

In the case of Surry, plant modifications since the Reactor Safety Study allowing cross-connection of the high-pressure safety injection systems, auxiliary feedwater systems, and the refueling water storage tanks for the two units have substantially reduced core damage frequency due to loss of coolant accidents, despite a tenfold increase in estimates of small LOCA initiating event frequency. Thus, although new PRA information and techniques resulted in increases in calculated core damage frequency, plant modifications offset these effects, resulting in an overall decrease for Surry.

In the case of Peach Bottom, 96% of the median core damage frequency in WASH-1400 was due to ATWS sequences and transients with failure of long-term decay heat removal. Core damage resulting from failure of long-term decay heat removal in the present study is substantially reduced by the inclusion of the possibility of containment venting (via the wetwell airspace), and realistically allowing for some core cooling after postulated containment failure. Core damage frequencies due to ATWS sequences have been reduced because the plant has implemented ATWS fixes, and modern neutronic and thermal hydraulic analyses of the ATWS sequences predict lower core power levels during the events, allowing greater opportunity for mitigation. In the NUREG-1150 analysis, the frequency of core damage due to station blackout sequences is now slightly larger than that estimated for the ATWS sequences. Thus, for Peach Bottom, advances in PRA methodology and plant modifications have both contributed to a reduction in the estimated core damage frequency, and station blackout has emerged as the largest contributor to core damage.

Note. The above explanation of changes in the core damage frequency for Peach bottom is based on the discussion in NUREG-1150. As noted earlier in Section 3b(3) of our report, the Committee has some reservations about the calculated core damage frequency for ATWS events because generic rather than plant-specific data were used in the NUREG-1150 analysis of these events for Peach Bottom and Grand Gulf.

b. Accident Progression and Containment Performance

One major difference between WASH-1400 and NUREG-1150 is the fact that the Surry containment is shown to be considerably stronger than assumed in the earlier study, as evidenced by the comparisons of cumulative containment failure probability shown in the upper portion of Figure 4.2. (This figure is a reproduction of Figures 9.6, 9.7, and 9.9 from NUREG-1150). The curves in the upper portion of this figure indicate that the median cumulative containment failure pressure has shifted from approximately 80 to 130 psig.

The failure pressure curves for Peach Bottom in the center portion of Figure 4.2, indicate no substantive difference between the two studies with regard to the BWR Mark I containment response to static pressure. As shown in the lower portion of Figure 4.2, the Sequoyah and Grand Gulf cumulative containment failure probabilities increase much more rapidly with increasing pressure than for Surry, Zion and Peach Bottom, owing to the much lower design pressures for these two plants. Whether there is containment failure due to overpressure is a function of the loads imposed and the containment response. Much more is now known concerning all aspects of severe accident progression, including pressure loads and response, than was available at the time of the Reactor Safety Study.

In NUREG-1150, p. 9-10, the following observation is made.

> "At the time the Reactor Safety Study (RSS) analyses were undertaken there had been no relevant experimentation and analysis of either loads produced in a severe accident or

Cumulative containment failure probability for static pressurization.

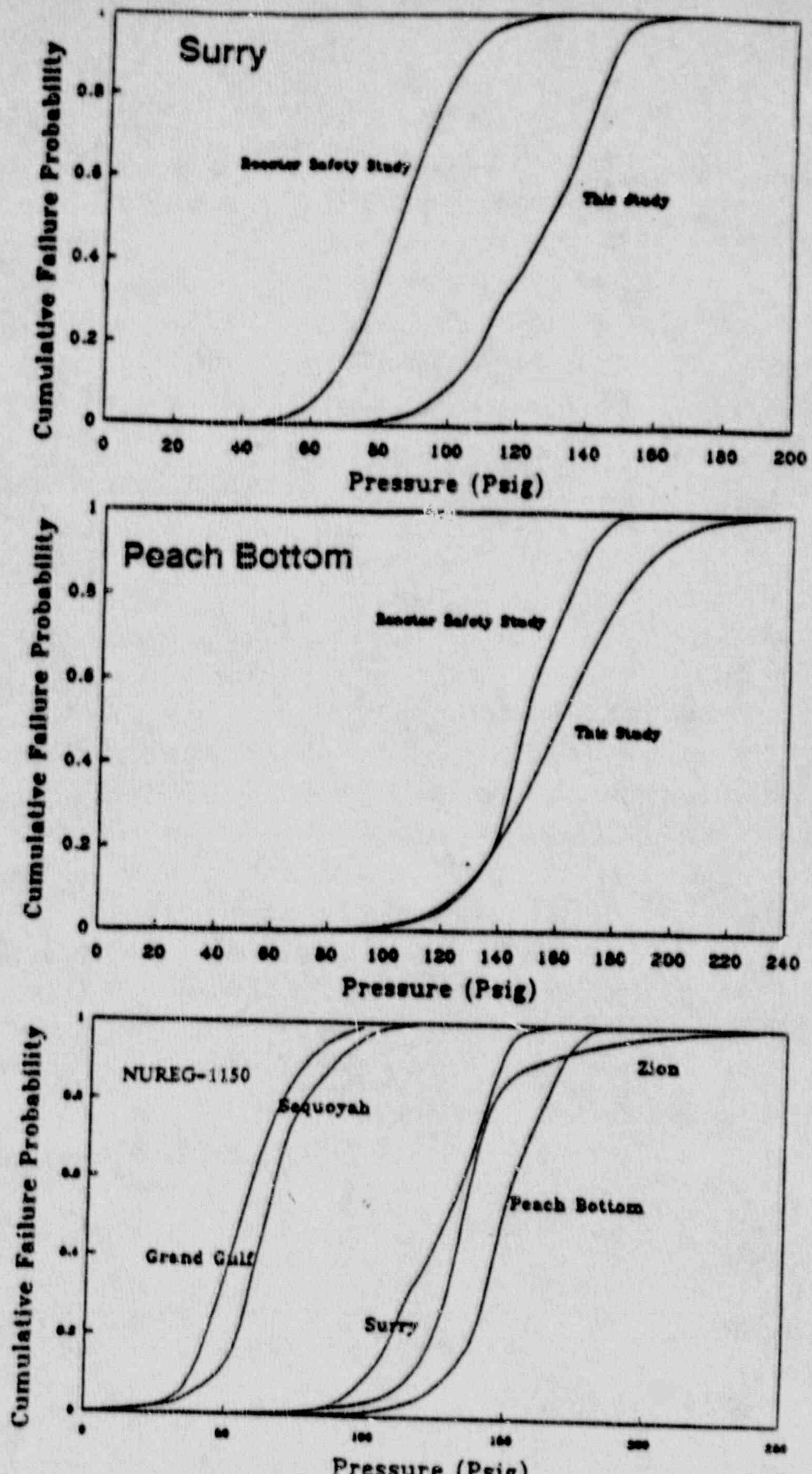


Figure 4.2 Comparison of NUREG-1150 and WASH-1400 Cumulative Containment Failure Probabilities (Figures 9.6, 9.7 & 9.9 in NUREG-1150)

the response of a containment to loads exceeding the design basis. As a result, the characterization of containment performance in the RSS is quite simplistic in comparison with the present study."

Some of the dramatic changes in containment performance since WASH-1400 are illustrated in the tabulations of containment failure probability, conditional on core damage occurring, listed in Finding #5 in Section 2 of our report. In WASH-1400, all severe core damage sequences were assumed to lead to failure or bypass of the containment. By contrast, the containment remains intact for most of the PWR severe core damage accidents in NUREG-1150 (e.g., 81% for Surry) and for a substantial fraction in the case of the BWRs (e.g., 26% no containment failure and 13% containment venting via the wetwell airspace with substantial radionuclide retention in the suppression pool for Peach Bottom).

A summary comparison of some of the principal results of the Level 1 and Level 2 analyses from WASH-1400 and NUREG-1150 are presented in Figures A.1 through A.7 of Appendix A, including apportionment of the total estimated core damage frequency by.

- Initiating Events
- Plant Damage States
- Accident Progression Bins, and
- Approximate Source Terms Magnitudes (using iodine as an example)

For the sake of completeness, summaries are included in Appendix A for Sequoyah, Zion and Grand Gulf, as well as the two plants studied in WASH-1400.

The numerical values in Figures A.1 and A.2 are based on median core damage frequencies reported in WASH-1400, whereas the data in Figures A.3 through A.7 are based on mean values from NUREG-1150. It was not possible to construct entirely comparable diagrams because only median data were reported in WASH-1400. Although median and mean data were reported in NUREG-1150 for many results, only mean conditional probabilities of containment failure were reported. An estimate of the mean values for WASH-1400 can be arrived at by multiplying the median data by approximately a factor of three, owing to the log-normal nature of the probability distributions in the Reactor Safety Study.

Some observations related to the two analyses of Surry and Peach Bottom, as depicted in Figures A.1 through A.4, follow.

Surry

- In the Reactor Safety Study analysis for Surry, 72% of the core damage frequency was associated with LOCAs and bypass events, and 28% with transients. In NUREG-1150 the reverse is observed, wherein 23% of the core damage frequency is associated with LOCAs and 77% with transients and station blackouts. This difference results from the changes in the plant (such as crossties of systems) and advancements in analysis techniques as noted in the previous discussion of core damage frequency.

- Although containment bypass events account for only approximately 8% of the core damage frequency in both studies, these sequences dominate offsite risk due to the large releases (i.e., source terms) calculated for bypass sequences. In NUREG-1150, 12% of the severe accident progression bins are associated with containment bypass, substantially more than the 0.7% associated with early containment failure, resulting in bypass sequences dominating the risk analysis.

- In the Reactor Safety Study all core melt accidents were assumed to result in containment failure; 24% of the severe core damage sequences resulted in early containment failure or bypass and 76% resulted in basemat meltthrough. By contrast, in NUREG-1150 the containment remains intact for 81% of the Surry core damage sequences, and early containment failure is calculated to occur for accident sequences constituting only 0.7% of the core damage frequency.

Peach Bottom

- Essentially all core damage is associated with transients in both studies of Peach Bottom and only a few percent is associated with LOCAs. Unlike Surry, there was no change in the relative contributions to core damage frequency due to transients and LOCAs between the two studies.

- Due to the substantial reductions in ATWS and long-term decay heat removal failure probabilities, station blackout sequences are now more significant than in WASH-1400.

- Although considered in both studies, containment bypass (i.e., interfacing system LOCAs) is not an appreciable factor in either study for Peach Bottom.

- In the Reactor Safety Study of Peach Bottom 100% of core melt sequences were assumed to result in early containment failure. By contrast, the NUREG-1150 analysis indicates 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 are seen to be substantially different in the WASH-1400 and NUREG-1150 analyses for these two plants.

The dominant contributors to risk, indicated in the diagrams of Appendix A, are as follows.

WASH-1400

Surry - Containment Bypass - Interfacing System LOCAS
- Early Containment Failure - Station Blackout

Peach Bottom - Early Containment Failure - Transients (Station Blackout, ATWS and Loss of Long-Term Decay Heat Removal)

NUREG-1150

Surry - Containment Bypass (Steam Generator Tube Rupture and Interfacing System LOCAs)

Peach Bottom - Early Containment Failure - Transients (Station Blackout and ATWS)

Sequoyah - Containment Bypass (Steam Generator Tube Rupture and Interfacing System LOCAs)

Zion - Early Cont. Failure - Small Break LOCA
- Containment Bypass (Steam Generator Tube Rupture and Interfacing System LOCAs)

Grand Gulf - Early Containment Failure (Station Blackout).

c. Severe Accident Source Terms

In NUREG-1150 at p. 10-1 the following observation is made.

> "It is widely believed that the approximate treatment of deposition mechanisms in the Reactor Safety Study analysis led to a substantial overestimation of severe accident consequences and risk. The current risk analyses provide a basis for understanding the differences that exist in source terms calculated using the new methods relative to the older methods and of their impact on estimated risk."

The substantial reductions calculated in source terms between WASH-1400 and NUREG-1150 analyses for Surry and Peach Bottom are illustrated in Figures 4.3 through 4.6 which depict the frequency of exceeding releases to the atmosphere for specific fractions of the core inventory of iodine, cesium, strontium and lanthanum. The median, mean, 5th and 95th percentile data from the NUREG-1150 analysis are included. However, only median data are available from WASH-1 400. The shaded areas in these figures illustrate the reductions in median source terms between the two studies.

Three specific observations from the Surry source term data are.

- For iodine and cesium (Figure 4.3) the median frequency of releases of 10% or more of the core inventory in NUREG-1150 are more than an order of magnitude lower than reported in WASH-1400 and the median frequency of releases of higher magnitude, similar to the PWR-2 release category (70% iodine), are off the low end of the scale of this figure (less than 10-8 per reactor year).

- For strontium (Figure 4.4) the median frequency of release of 1% or more of the core inventory of strontium in NUREG-1150 is three orders of magnitude below the comparable WASH-1400 value. However, mean and 95th percentile probabilities of releases of greater than 6% of the core inventory of strontium (the largest PWR release reported in WASH-1400) are observed in NUREG-1150, but at very low frequencies.

- For lanthanum (Figure 4.4) there is more than a three order of magnitude reduction in the median frequency of release of 0.04% of the lanthanum core inventory as compared with WASH-1400. However, mean and 95th percentile releases greater than 0.4% (the largest lanthanum release reported in WASH-1400) are observed in NUREG-1150, but at very low frequencies.

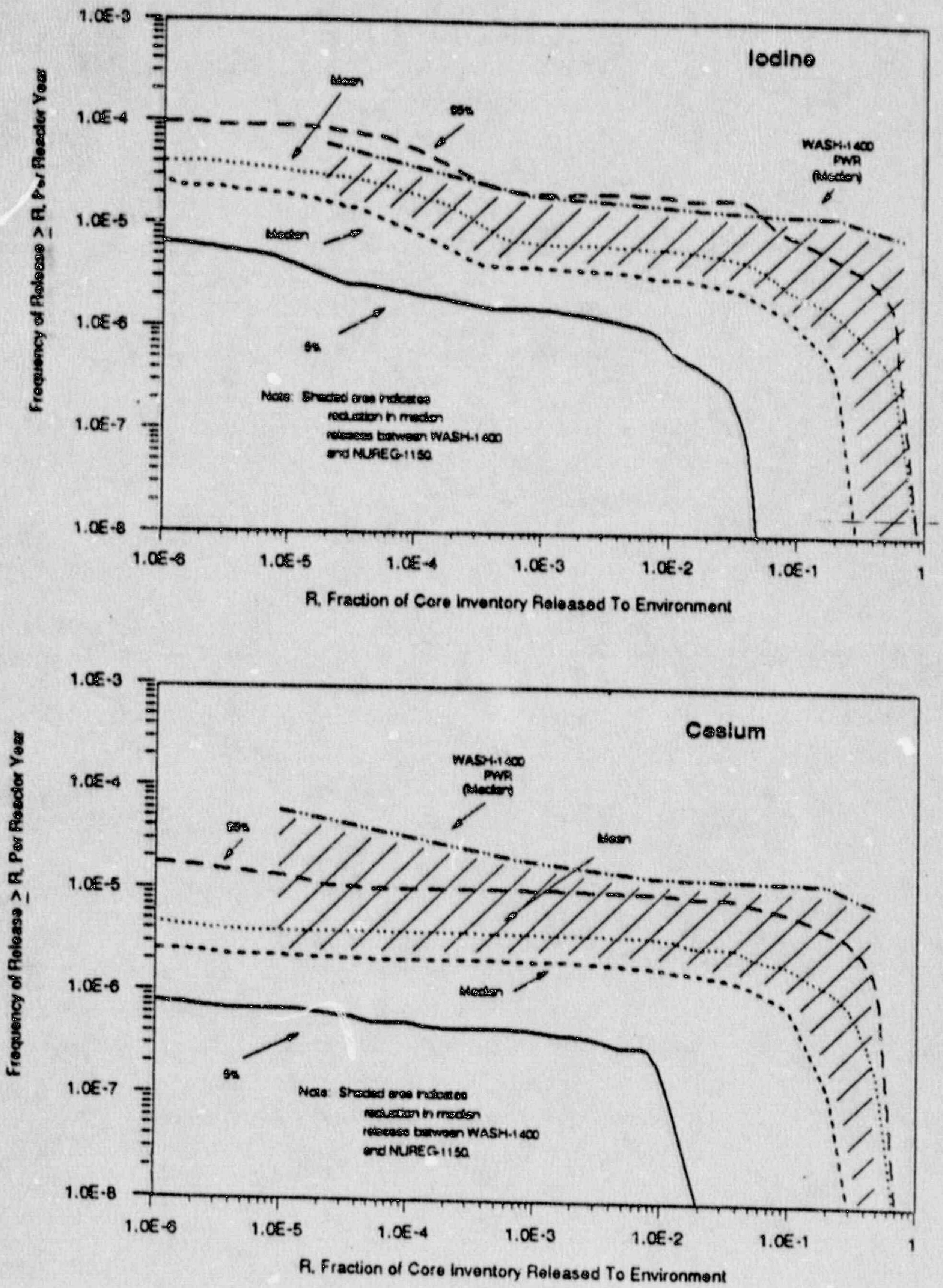
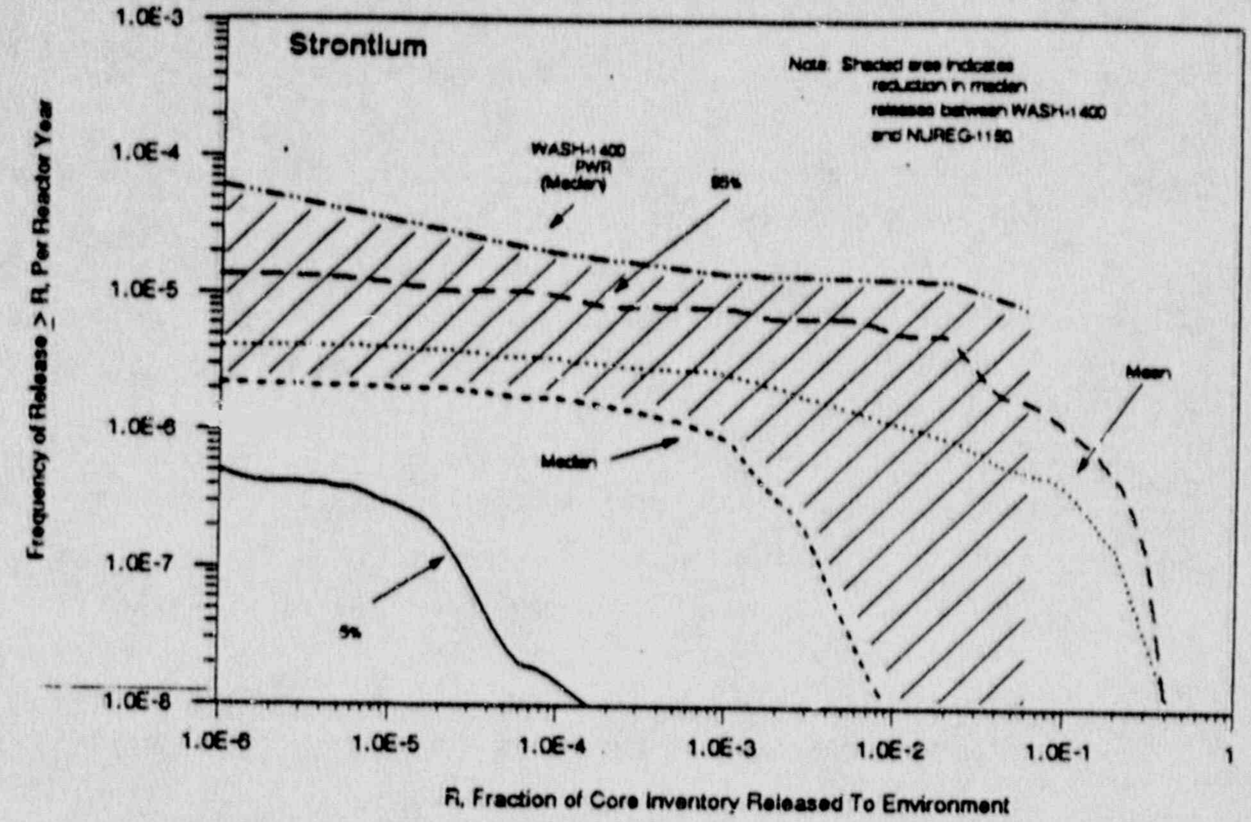


Figure 4.3 Comparison of Surry Iodine & Cesium Source Terms in NUREG-1150 and WASH-1400 (Internal Events)



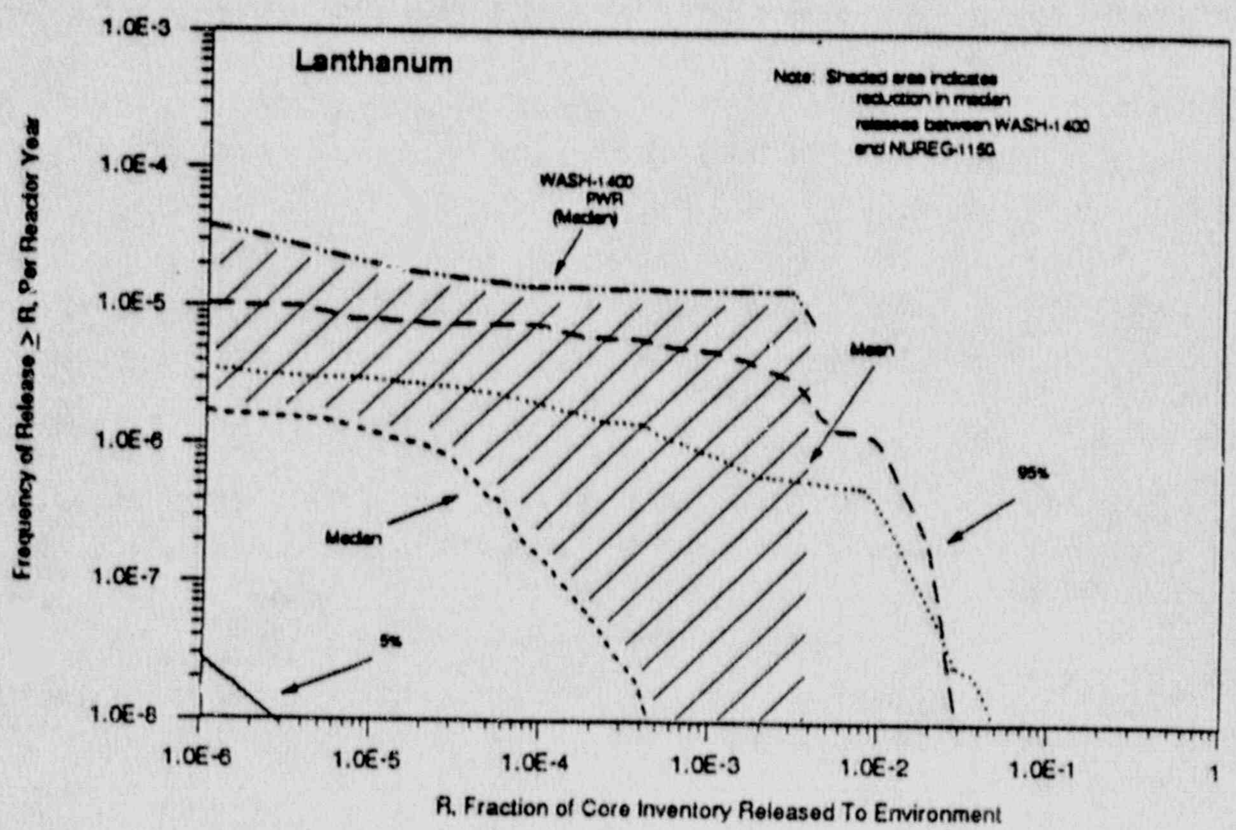


Figure 4.4 Comparison of Surry Strontium & Lanthanum Source Terms in NUREG-1150 and WASH-1400 (Internal Events)

Three specific observations from the Peach Bottom source term data are.

- For iodine and cesium (Figure 4.5) the median frequency of releases of 10% or more of the core inventory in NUREG-1150 are approximately two orders of magnitude lower than reported in WASH-1400, and the median frequency of higher magnitude, similar to the BWR-2 release category (90% iodine), are off the low end of the scale of this figure (less than 10-8 per reactor year).

- For strontium (Figure 4.6) the median frequency of release of 1% or more of the core inventory in NUREG-1150 is two orders of magnitude below the comparable WASH-1400 value. However, the mean and 95th percentile probabilities of releases greater than 10% of the core inventory of strontium (the largest BWR release reported in WASH-1400) are observed in NUREG-1150, but at very low frequencies.

- For lanthanum (Figure 4.6) there is more than a four order of magnitude reduction in the median frequency of release of 0.05% of the core inventory (the largest BWR lanthanum release reported in WASH-1400). However, mean and 95th percentile releases greater than 0.05% are observed in NUREG-1150, but at very low frequencies.

These observations illustrate the results of some of the major advancements in source term technology since the Reactor Safety Study. However, they also reflect decreases in core damage frequency and containment failure rates.

Much more is now known about the release and transport of volatile fission products such as iodine and cesium, including deposition in and subsequent revaporization from surfaces in the reactor coolant system. Similarly, new data on core concrete interactions has resulted in differences such as those observed with respect to strontium and lanthanum in Figures 4.4 and 4.6 (mean release fractions larger than the highest releases in WASH-1400)

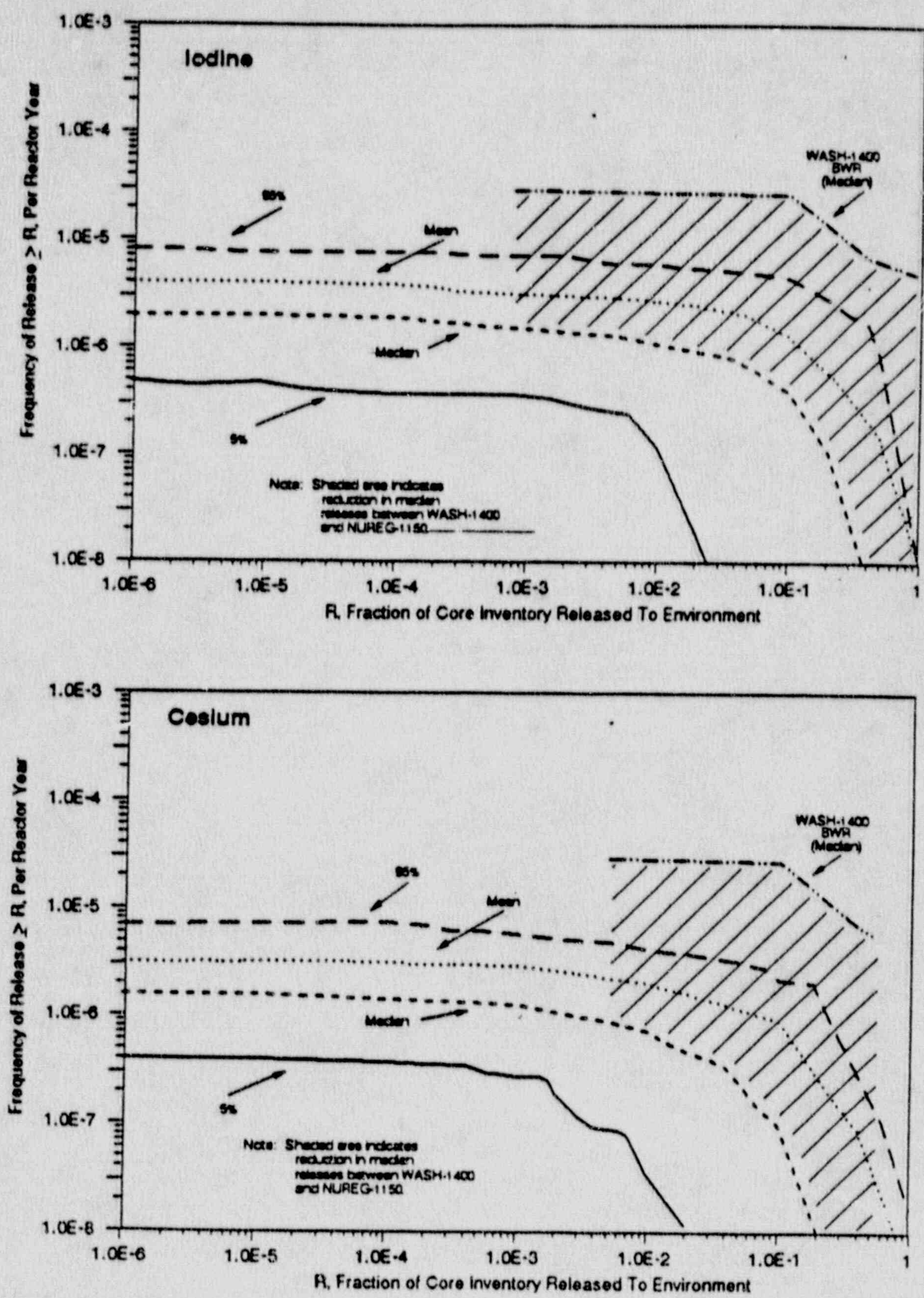


Figure 4.5 Comparison of Peach Bottom Iodine & Cesium Source Terms in NUREG-1150 and WASH-1400 (Internal Events)

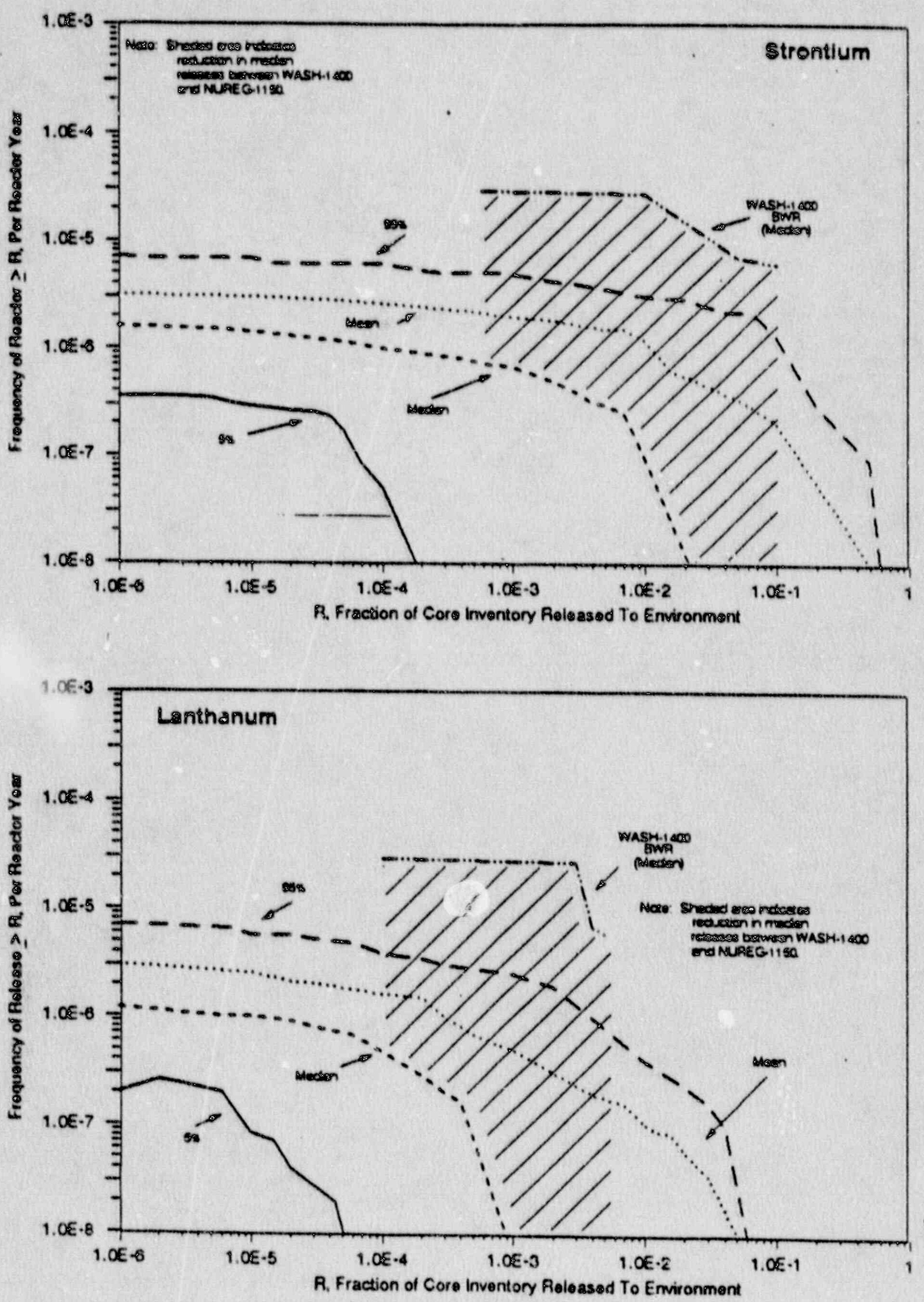


Figure 4.6 Comparison of Peach Bottom Strontium & Lanthanum Source Terms in NUREG-1150 and WASH-1400 (Internal Events)

d. Comparison of Elements of Risk Analysis Process

Sufficient data were available to the Committee, both in NUREG-1150 and as a result of briefings by the staff of Sandia National Laboratories, to allow the development of a summary comparison of progression of the results of the analysis in the two studies as presented in Figure 4.7. Data are presented for accident sequences resulting in early releases for Surry for the following three portions of the analysis.

- Systems Issues Only (i.e., Core Damage Frequency)
- Systems and Containment Issues (limited to early failure or bypass)
- Systems, Containment and Source Term Issues (using 10% iodine release as an example)

The two vertical bars to the left of Figure 4.7 compare the NUREG-1150 core damage frequencies with the core melt frequencies from WASH-1400, as embodied in release categories PWR 1 through 7. (WASH-1400 release categories PWR 8 and 9 did not include severe core damage.)

The two vertical bars in the center of this figure compare early containment failure or bypass data from Figure 9.3 of NUREG-1150 with the comparable data for PWR 1 through 5 release categories from WASH-1400, i.e., core damage sequences which progress to include early containment failure or bypass.

The two bars to the right of this figure compare the frequency of release of 10% or more of the iodine inventory. The NUREG-1150 data were obtained from Sandia and the WASH-1400 data are for release categories PWR 1 through 3.

Several observations from the data presented in Figure 4.7 follow.

- The systems issues portion of the comparison shows a reduction of a factor of 2.6 in the median core damage frequency and a range of uncertainty slightly smaller in NUREG-1150 than in WASH-1400.

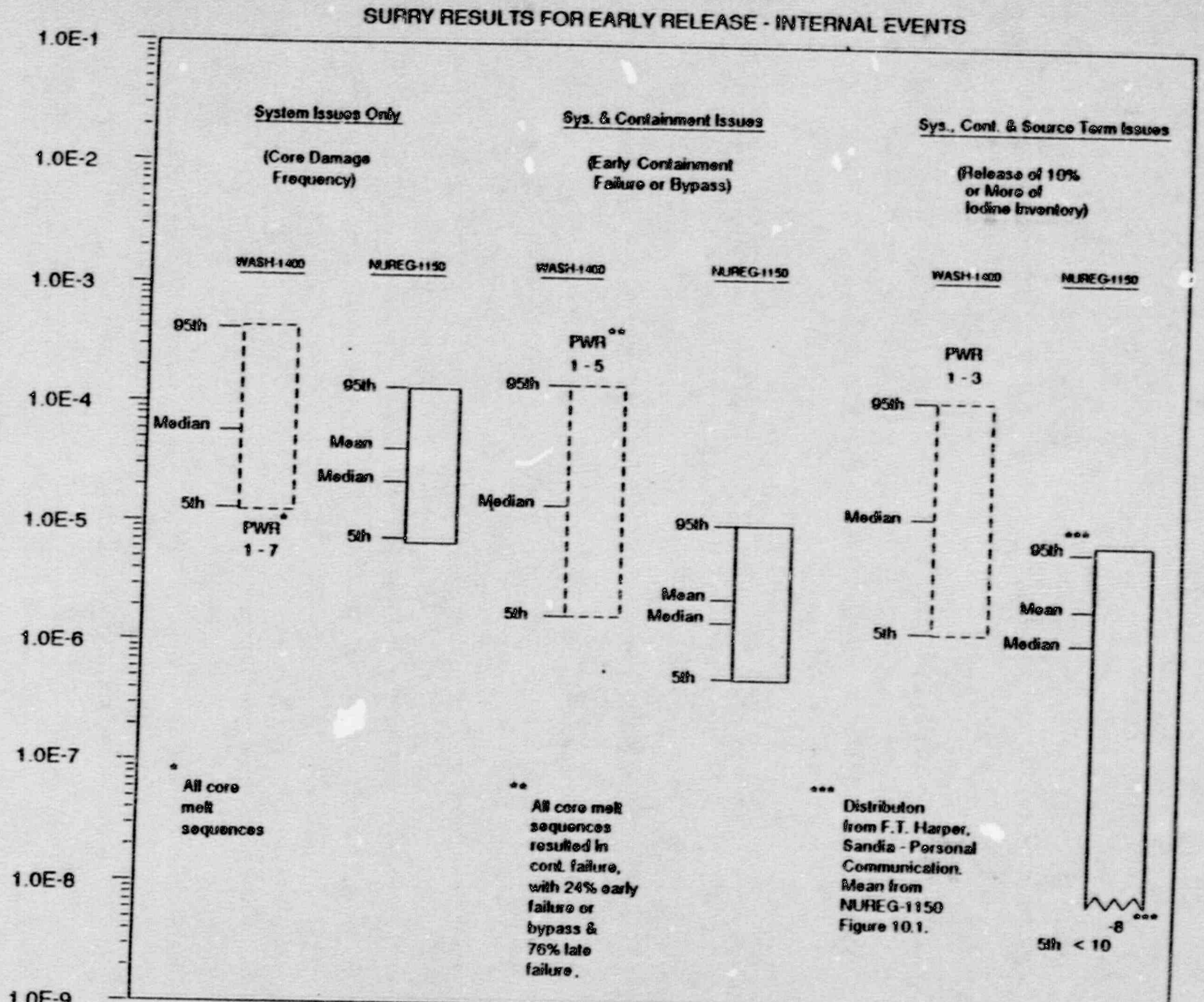


Figure 4.7 Comparison of Results of Analysis Progression for Early Releases in NUREG-1150 and WASH-1400

- When systems and containment issues are combined, as in the case for the bars in the center of the figure, the median frequency for NUREG-1150 is observed to be approximately an order of magnitude lower than WASH-1400 and the range of uncertainty is substantially smaller than in the earlier study.

- With respect to the frequency of releases of 10% or more of the core inventory of iodine, as indicated by the two bars at the right of the figure, it is not surprising that the median in each of the two studies is close to the median for early containment failure or bypass (i.e., the center bars), since large releases were postulated for such events in both studies. However, it is noteworthy that the 95th percentile frequency for release of 10% or more iodine inventory in NUREG-1150 is lower than the median value from WASH-1400.

e. Offsite Consequences

Direct comparison of offsite consequences reported in WASH-1400 and NUREG-1150 is virtually impossible because there are many differences between the two studies. For example, the CRAC computer model was used in WASH-1400 and the MACCS model was used in NUREG-1150; site specific meteorological data was used in NUREG-1150 whereas composite averaging from many sites was used in WASH-1400; different assumptions relating to emergency evacuations were used in the two studies; and the numerical results were sometimes treated differently in the two reports. Nonetheless, it may be noted that the offsite consequences in NUREG-1150, in general, are substantially lower than those reported in WASH-1400.

5. QUALITY OF THE REPORT

a. General

At the outset of our discussion of quality it is important to distinguish between the NUREG-1150 report itself (2 volumes), and the process which created the results presented and discussed there. The process is described in detail in a multitude of backup documents which are referenced in NUREG-1150, though many of them have yet to be published. As a matter of practicality (the magnitude of backup material) and timing (the present unavailability of much of it) this discussion of quality focuses primarily on the NUREG-1150 document itself. Comments relevant to the process speak to that process mainly as it is revealed in the two main documents.

In its overall configuration the revised version of NUREG-1150 is a much improved document over the original. Following a concise introductory chapter in which the objectives of the project and the report are stated, there is a chapter in which the methodology is presented, including explanatory examples of how the results will be displayed. Thus the first few pages of the report give the reader an understandable view of what is to come, and why. Volume 2 of the main report contains three appendices that amplify the discussion of methodology (Appendix A), present a sample calculation (Appendix B), and provide a description of how a number of difficult decisional issues involving uncertainty were evaluated (Appendix C). These Appendices combined with the first two chapters of Volume 1 provide a clear explanation of the process.

The five chapters in which the results for each of the plants are presented are also quite clear and readable. Because the same format is used in each of these chapters, with the same explanatory remarks, they seem repetitious, but this is inevitable since each deals with a different plant, with its own characteristics and results. The initial draft of NUREG-1150 was better than the present version in one respect: a schematic drawing of the containment and primary system of each plant was included in the section devoted to that plant, a ready reference for those not totally familiar with the different designs. (Such figures can only be found in the Appendices of the revised version).

The third part of the main report, Chapters 8 through 13, which presents "perspectives" on various aspects of the calculations and their use, is the least effective and hard to follow. Whereas certain of the material here is very worthwhile (e.g., comparisons with the RSS) much of the discussion seems forced. There are observations for which there is no apparent basis in the results, some important aspects of the results receive little or no attention, and there are conflicting statements. (Specific examples of these shortcomings are found in Subsection e below).

b. <u>Presentation of Results</u>

A major criticism of the original NUREG-1150 was the manner in which the results at the different stages of the calculations were portrayed. In short, the figures there gave the range of uncertainty (5% to 95%) of various results with no indication of central tendencies (means or medians) and no idea of the shape of the probability distributions. The current version does a much better job of presenting calculational results which may cover a wide span of uncertainty. Probability distributions, including medians and means, are shown for core damage frequency, containment failure probability and individual isotope release. A particularly helpful form of the results is the matrix-like figures in which mean values of accident progression probabilities (i.e., containment performance) are shown in conjunction with mean plant damage states (core damage frequencies) (see Figures 3.4, 4.4, et seq.). Pie charts are used effectively to display qualitatively the contributions of various initiating events and accident progression scenarios.

It is unfortunate that, apparently for conservation of pages (an objective not noticeable elsewhere in the report), some presentations of results are so reduced in size as to be virtually unreadable (e.g., Figures 2.7, 10.1, 9.1, and 9.2). Also some pie-charts have so little contrast between adjacent segments that the quantity of certain sequences and scenarios is not distinguishable (e.g., Figures 9.4 and 9.5). Figure 8.4 is a bar chart but most of the bars are so short that it is impossible to distinguish the shading intended to identify different scenarios. And Figure 13.6 provides almost no information.

c. Expert Opinion Elicitation

The original report was subjected to criticism by virtually all who reviewed it for the way in which expert opinion was elicited. Expert opinion was used to determine probability distributions for parameters not resolvable by calculation or experimental results (the so called "issues"). This procedure was substantially upgraded for the revised analyses underlying the second version of the report. The new elicitation procedure is outlined in Section A.7 of Appendix A. Briefly, experts (seven panels in different areas of expertise) were selected from a wide range of institutions (national laboratories, schools, industry, and the NRC), they underwent training in the opinion elicitation process, they were provided with formal presentations on the issues including relevant calculations, they were given a substantial amount of time (1- to 4 months) to prepare their analyses (and encouraged to do their own calculations and seek the latest data), their opinions were obtained in private sessions using established, formal procedures (and were documented), and the aggregated results were then reviewed by the individual panels.

One can always argue that expert opinion is no substitute for calculated or experimental results. However, for the very complex issues in these analyses which the expert panels quantified there is little chance that reliable analytical or experimental results will soon be at hand. The upgraded elicitation process used for NUREG-1150 provided more than "expert opinion". Rather as carried out, it is better characterized as "informed judgment" elicitation. While there were instances in which decided differences in the experts' judgment led to strongly bimodal distributions, (e.g., Figure C.7.4), generally the judgmental results of the various individuals in a panel were well bunched (e.g., Figures B.8, B.11 and B.12, C.5.7, C.10.4).

Appendix C is an interesting discussion of a number of the important uncertain issues, many of which were quantified by expert elicitation. An enlightening feature of this appendix is the presentation of diagnostic or sensitivity studies which illustrate the effect that selecting extreme values of parameters would have on the resulting risks (e.g., no common-cause failures, Table C.2.5, and no human errors, Table C.3.2). Such studies indicate that errors in quantifying these issues would not have a dramatic effect on the results.

d. The Use of the Latest Data

The original report was criticized for not incorporating the latest experimental data or calculational techniques. In its present version this particular problem would seem to have largely been solved by the expert opinion elicitation process. As noted, experts, seemingly well-versed in their fields, were selected in specific "issue" areas, and given the opportunity to work for some time to form their judgments. It is almost axiomatic that these individuals, faced with very difficult questions and the open and stylized elicitation process, would have availed themselves of all, including the latest, information at hand (unfortunately one cannot have the same degree of confidence that issues evaluated by the Project staff incorporated the most recent data). A review of the references cited for backup in Appendix C indicates, as well, that recently published information makes up the major portion of the material used. Certainly no effort such as this, spanning several years, will ever be able to avail itself of every new calculational model, experiment, or plant system modification. For this reason the author's characterization of their work, in this second version of the report, as a snapshot at a particular point in time, is wise and appropriate.

e. Specific Quality Shortcomings

Although our opinion of the revised NUREG-1150 is generally favorable, particularly when it is compared to the initial version, there are still areas in which the report falls short of expectations. In as much as there may be changes in the document before it is published in final form, we are encouraged to point out some of the shortcomings we have found.

(1) The purpose of Appendix C was to provide some insight to the resolution of key issues, in particular those addressed by expert panels. These discussions are sketchy, however, and the information and reasoning which led to the expert judgements are generally not provided. Also there seems to have been

no concerted effort to include in Appendix C a discussion of those issues which were responsible for the major elements of risk for each plant (e.g., containment failure due to hydrogen combustion at Grand Gulf). Appendix J of the initial version of NUREG-1150 was a much more informative presentation of issue resolution.

(2) Appendix B is a valuable example of an accident scenario carried through from initiation to consequence calculation. Unfortunately the example chosen for portrayal does not include early containment failure, hence many of the more interesting issues which are important to risk are not included in the presentation (such as containment failure related factors used in the SURSOR calculations).

(3) Direct containment heating (DCH) was identified in the initial version of NUREG-1150 as a major contribution to early containment failure and risk. In the revised version of the document this phenomenon goes virtually unmentioned, yet its disappearance must reflect major changes in the potential for primary system depressurization prior to vessel meltthrough and improved understanding of containment loading due to high pressure melt ejection, subjects which are worthy of fuller explanation. Chapter 9, which purports to provide perspectives on containment performance, mentions the DCH only in passing, with no reference to its reduced importance.

(4) It is not clear how credit is taken for fission product retention in the auxiliary building for PWR containment bypass sequences and the reactor building for BWR containment failures. These phenomena can result in significant source term reductions.

(5) In a number of places in the perspectives, Chapters 8 through 13 of NUREG-1150, there are statements which appear to have no basis in the analysis, or to be contradicted by other statements.

- At page 8-12 Grand Gulf is said to be better equipped than other plants, including Peach Bottom, to deal with losses of AC power because it has a diesel-driven high pressure core spray. Yet a comparison of core damage frequencies shows the likelihood for core damage due to blackout for Grand Gulf to be twice that for Peach Bottom (compare Figures 6.3 and 4.4).

- At page 9-12 a reduction in core damage frequency due to Anticipated Transients Without Scram (ATWS) events is said to be due to ATWS modifications made at Grand Gulf and Peach Bottom. Yet the specific modifications made at these plants for ATWS and their vulnerabilities were not modeled explicitly in the analyses (only generic modifications were considered) so this conclusion is hard to justify.

- At the top of page 9-12 it is stated that the Peach Bottom containment is shown to be less strong in NUREG-1150 than it was in WASH-1400, referring to Figure 9.7. The figure shows just the opposite.

- In Section 13.2.6 it is concluded that at 3 miles and beyond, for an early release, evacuation always reduces radiation doses beyond what could be achieved by sheltering people in a large building. Figure 13.5, however, appears to show that such sheltering results in doses that are equivalent to, or less than those for all evacuation cases depicted.

(6) In a number of places NUREG-1150 makes claims for itself which do not appear to be substantiated.

- At page 13-1 the methodology is described as state-of-the-art of PRA. Yet with regard to the core damage, or front end analyses, at pages A-10 and 11 the level of detail in the systems analysis, human reliability, and data base analyses are characterized as less than state-of-the-art, and are recognized as being less detailed than those often used in today's PRA's.

- At the top of page 13-3, NUREG-1150 is cited as a source of phenomenological and operational data. In fact, most of such data can only be found in the back-up documents.

- At page 13-2 the methods of uncertainty analysis and expert solicitation are stated to be a[illegible]e to other plant analyses. Given the tremendous cost associated with the application of these techniques it is unlikely that an individual utility would, or could, be expected to utilize them.

(7) In a number of instances the report suffers from poor, or inconsistent writing.

- Section 8.4.1 and the first part of Section 8.4.2 are a discussion and interpretation of the core damage frequency distributions. This material is simplistic, directed at a level of reader understanding well below the rest of the report (e.g., the terms median, majority and dominant are explained or defined here).

- The term "acute red bone marrow" dose is introduced for the first time in Section 13.2.6 without explanation or distinguishing it from the simple rem doses discussed in earlier chapters. In Section 13.2.1 acronyms like BWR and LOCA are redefined.

- Section 13.2.5 is entitled "Alternative Safety Goal Implementation Strategies". Only in the last few lines of a page-long discussion does the subject of alternative safety goals even arise, and then just one such goal is mentioned.

- The material in Section 13.3, "Major Factors Contributing to Risk" has no apparent bearing on the section title. There are numerous redundancies (three times "broad categories of risk" is defined). This section has little or no relevance to NUREG-1150, particularly a portion entitled "Reactor Research", a simplistic, generalized discussion of research prioritization.

6. ADEQUACY OF THE REPORT

In this section we discuss the adequacy of NUREG-1150 with respect to its intended uses, which, as stated in the report are:

- To develop guidance for the conduct and review of the search for vulnerabilities in the individual plant examination (IPE) program.
- To assist in the consideration of the need for improvements to containment performance under severe accident conditions.
- To add to the compendium of PRA information and data base to assist in the identification of plant operational features or practices that have an adverse impact on plant safety.
- To assist in the evaluation of alternative safety goal implementation strategies.
- To assist in the evaluation of various accident management options.
- To provide additional information for use in the evaluation of research priorities and the prioritization and resolution of generic issues.

In addition to the primary uses listed above the NRC notes that:

- NUREG-1150 is a snapshot in time of severe accident risks in the five plants studied. It provides a snapshot of the state of the art of probabilistic risk analysis (PRA) technology, incorporating many improvements since the Reactor Safety Study, but retains the limitations of such studies regarding certain human interactions, equipment failure rates and common cause effects, ageing and incomplete understanding of physical processes.

- NUREG-1150 is not the sole basis for making either plant specific or generic regulatory decisions, nor is it an estimate of the risks of all commercial power plants in the United States or abroad.

In discussing the intended and possible uses of the report, the NRC has properly recognized the station-specific nature of the quantitative results and of the principal contributors to core damage frequency and risk. On the other hand, we concur with its view that there is a high degree of generic applicability of the advances in methods resulting from the NUREG-1150 program.

In as much as the report points out that in any of its intended applications NUREG-1150 will be used only as one of a number of information sources, it is difficult to criticize such uses. Hence, we will concentrate in offering cautions or encouragements with respect to the intended uses.

"To Develop Guidance for the Conduct and Review of the Search for Vulnerabilities in the Individual Plant Examination (IPE) Program"

We agree that it is appropriate that individual plants could be reviewed with respect to the plant and plant-class vulnerabilities that were identified in NUREG-1150. We caution, however, that just because a particular vulnerability was not found to be important on one of the five plants reviewed, it automatically does not need to be considered in future plant-specific studies. New insights and information will become available in the future, and plant-specific features not reviewed in NUREG-1150 can affect results.

It is suggested that the NUREG-1150 operational data base has applicability in other IPEs. While this data base can be used in IPEs, it is well known that plant-specific operating and maintenance practices can influence accident likelihood. Therefore, it is desirable to use plant-specific data when available.

Some of the approaches used (e.g., expert opinion elicitation) are likely to be too expensive to have widespread application. When, in such cases, information is extracted from NUREG-1150 for use in an IPE, it should be reviewed and justified for use in the context of the specific plant.

"To Assist in the Consideration of the Need for Improvements to Containment Performance Under Severe Accident Conditions"

There is extensive information on containment-related matters in NUREG-1150 that can be used to assist the determination of containment adequacy for other plants. The analysis of severe accident loads and containment response involves substantial uncertainty because of the complexity of the core meltdown processes. There have been substantial advances in methods of analysis, and models are available to describe nearly all aspects of containment loads. Nevertheless, because of modeling uncertainties, sensitivity studies and expert judgment are still a necessary input to assessment of containment performance.

Because the final determination of containment adequacy is plant specific, those responsible for performing these analyses must make the final judgment regarding the applicability of information from NUREG-1150. NUREG-1150 and its supporting documents provide a compendium of topics and expert opinions to assist them in this matter.

"To Add to the Compendium of PRA Information and Data Base to Assist in the Identification of Plant Operational Features or Practices that have an Adverse Impact on Plant Safety"

As long as the plant-specific nature of the models are recognized, we concur that the NUREG-1150 results can be used in this way.

"To Assist in the Evaluation of Alternative Safety Goal Implementation Strategies"

There is no basic reason why the NUREG-1150 results, as well as other risk assessments, cannot be used for this purpose. In all cases the question will remain as to the accuracy. Also the scope of the risk assessments (e.g., inclusion or exclusion of external events) needs to be consistent with the intended applicability of the safety goal. The results quoted in the report show that the plants examined all comfortably meet the NRC safety goals on early fatalities and latent cancer risks. A caution with respect to these NUREG-1150 quantitative results, however, is that the plants considered were all previously subjected to various risk assessment/design improvement programs, and the risk results may not be representative of the entire population of nuclear power plants. The NUREG-1150 presentation of results, in comparison with the NRC staff proposed large release goal of less than 10^{-6} per reactor year, is not thorough and would not be expected to be particularly useful in the evaluation of implementation strategies.

"To Assist in the Evaluation of Various Accident Management Options"

The NRC intends to have accident management programs developed and implemented by licensees. NRC evaluation of these programs will provide an independent assessment of the licensee-proposed accident management capabilities and strategies using NUREG-1150 as a resource document in the evaluation of particular strategies. The report shows examples of benefits which can be achieved by considering alternate strategies at the plants considered. The NRC correctly, in our view, points out that the integrated nature of the methods is particularly important since actions taken early in a sequence can alter the downstream consequences significantly.

"To Provide Additional Information for Use in the Evaluation of Research Priorities and the Prioritization and Resolution of Generic Issues"

We believe that NUREG-1150 can be used for assigning priorities. Risk assessment results, especially the results of many risk assessments taken together, can and should be used to assist the decision-making process regarding generic research and issues resolution, as long as the final results are tempered with an engineering evaluation of the reasonableness of the assignment.

Taken together with the results of the IPEs, which are primarily directed at the front end level 1 analyses, NUREG-1150 information and data in the level 2 analyses should help guide evaluation of accident management from a risk reduction perspective. However, such uses of NUREG-1150 would seem to be limited due to the parametric nature of the study.

The NRC appears to believe that the information developed in NUREG-1150 will not substantially change previously developed priority rankings. This is somewhat at odds with the information and insights presented in the areas of accident progression and source terms which show that several of the issues, earlier thought to have resulted in high risk, have been reduced in importance or eliminated entirely for some plants (e.g., Direct Containment Heating). The information presented in NUREG-1150 must be carefully examined in the context of the plant being studied to determine the priority ranking of safety issues, and we caution against broad generalities.

The report also discusses the effect of emergency preparedness on consequence estimates. It demonstrates how some of the factors important to emergency planning issues affect the dose received by members of the public. The results of various mitigating actions combined with the improved understanding of source term issues reported in the report, show that serious reconsideration of the basis for emergency planning zones is warranted. While there is no fundamental reason why NUREG-1150 results can not be used to do this, we caution that assumptions (e.g., percentage of people who do not evacuate) can dominate the results of such investigations, and, therefore, must be justified in the context of the plant being studied.

In summary, we believe that the NUREG-1150 results are generally adequate for their intended uses. This conclusion is primarily based on the fact that the intended uses are limited and general in nature, and that the NRC has recognized the station-specific applicability of the results. Progress in methods development should assist in future risk assessment efforts. The NUREG-1150 report will certainly prove to be of use to both the NRC staff and those responsible for PRA preparation in that it provides, in one place, a compendium of issues important to overall plant risk.

APPENDIX A

NOTES-
ALL FREQUENCIES ARE MEDIAN CORE MELT FREQUENCIES
ALL PERCENTAGES ARE % OF MEDIAN CORE MELT FREQUENCIES
"APPROXIMATE" MEDIAN IODINE RELEASE FRACTIONS USED TO CHARACTERIZE SOURCE TERMS-

VERY LARGE > 0.4
LARGE 0.03 - 0.4
SMALL 0.001 - 0.03
VERY SMALL < 0.001

==RISK DOMINANT ACCIDENT SEQUENCES

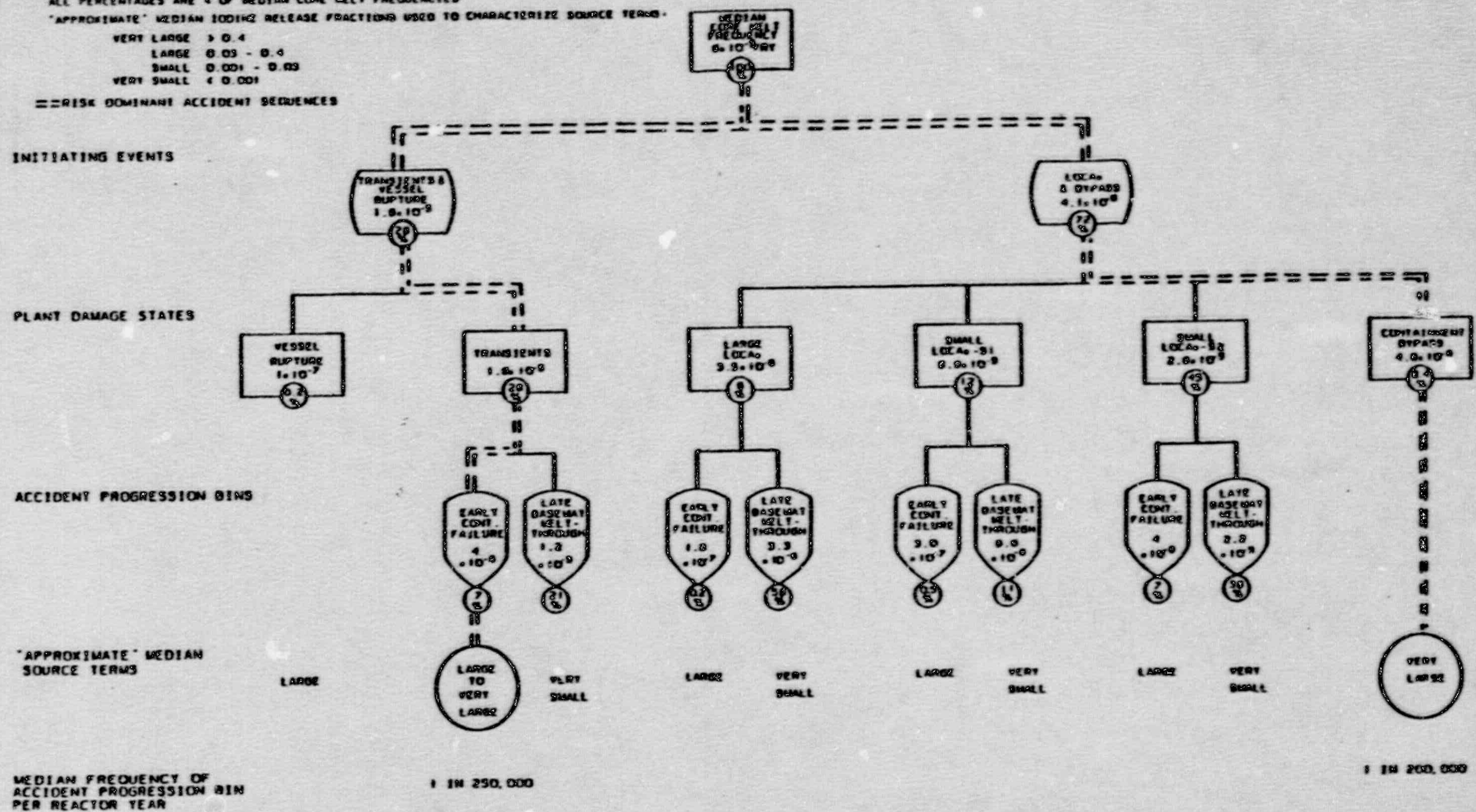


FIGURE A.1
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
WASH-1400 ANALYSIS OF SURRY - PWR

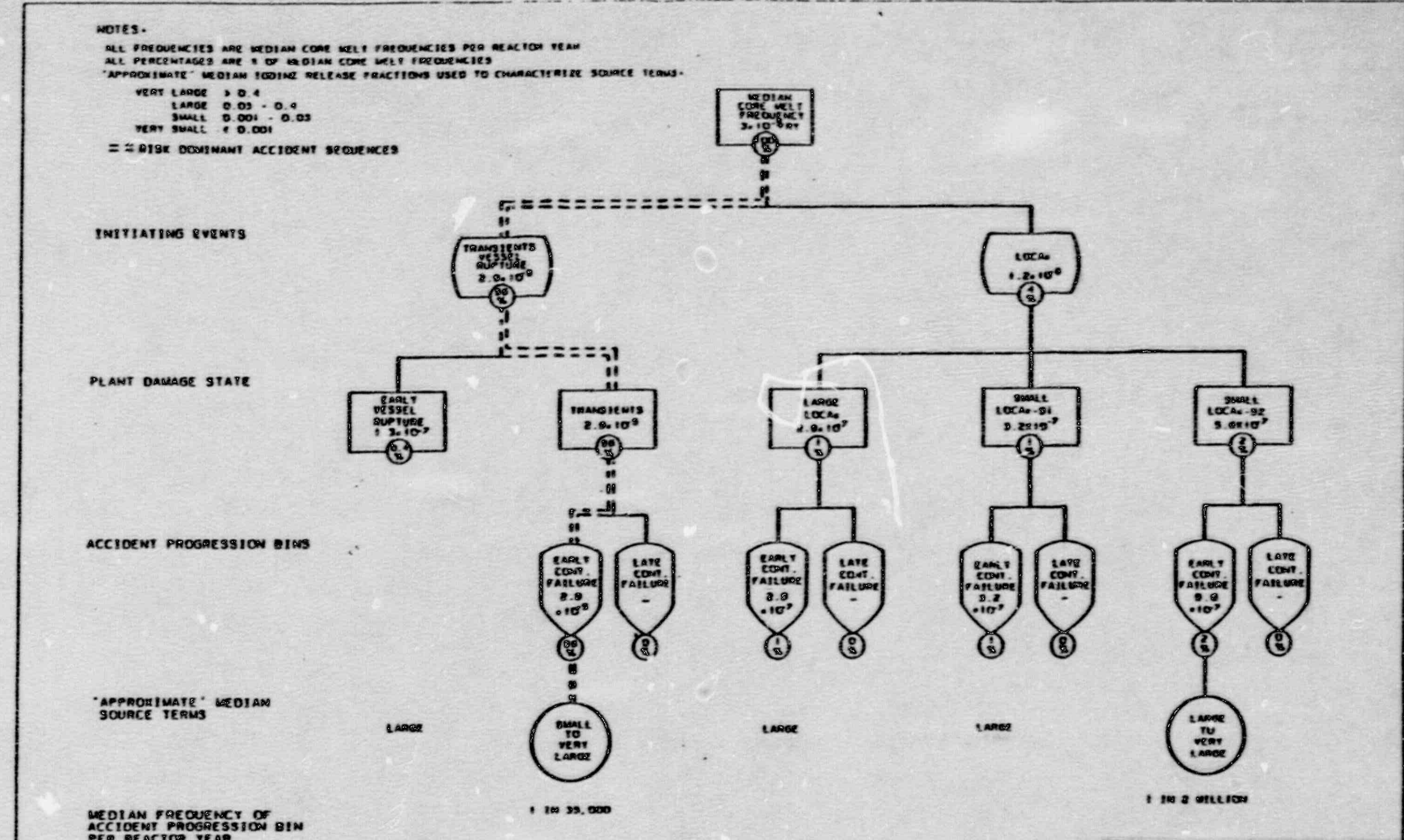


FIGURE A.2
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
WASH-1400 ANALYSIS OF PEACH-BOTTOM - BWR

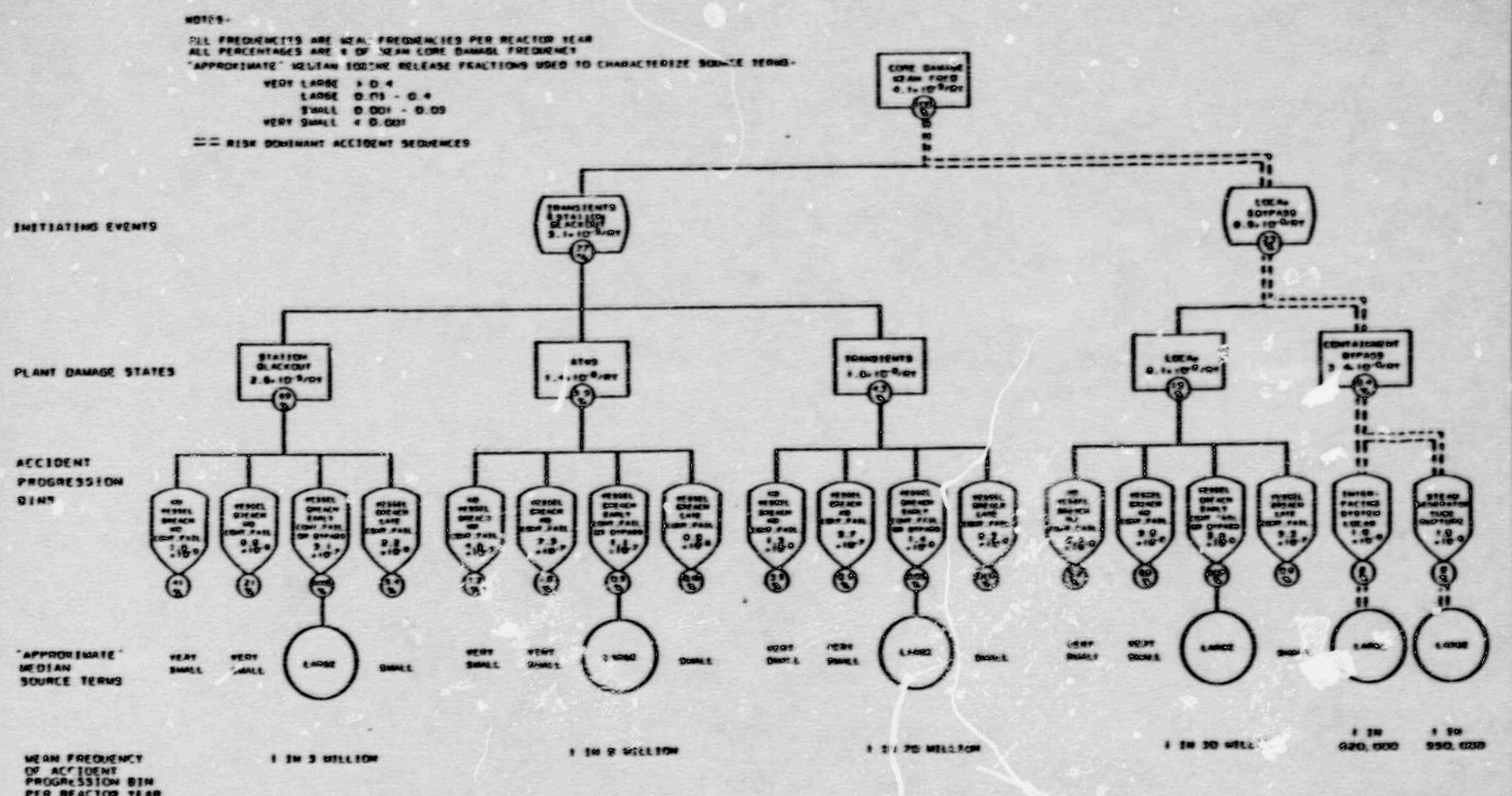


FIGURE A.9
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
NUREG-1150 ANALYSIS OF SURRY
(INTERNAL INITIATING EVENTS)

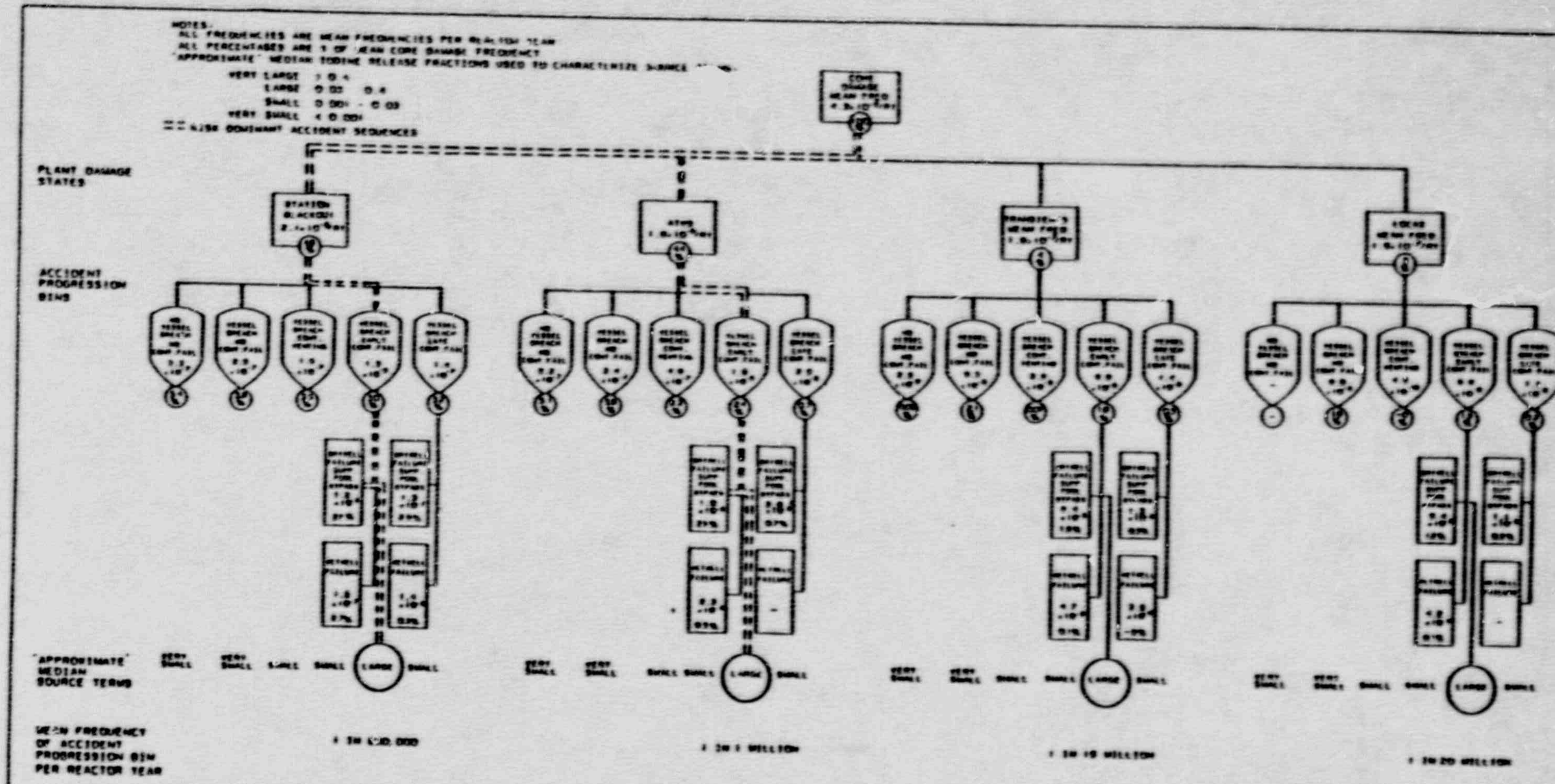


FIGURE A.4
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
NUREG-1150 ANALYSIS OF PEACH BOTTOM (INTERNAL INITIATING EVENTS)

NOTES-
ALL FREQUENCIES ARE MEAN CORE MELT FREQUENCIES PER REACTOR YEAR
ALL PERCENTAGES ARE % OF MEAN CORE DAMAGE FREQUENCY
"APPROXIMATE" MEDIAN IODINE RELEASE FRACTIONS USED TO CHARACTERIZE SOURCE TERMS-
VERY LARGE > 0.4
LARGE 0.03 - 0.4
SMALL 0.001 - 0.03
VERY SMALL < 0.001
== RISK DOMINANT ACCIDENT SEQUENCES

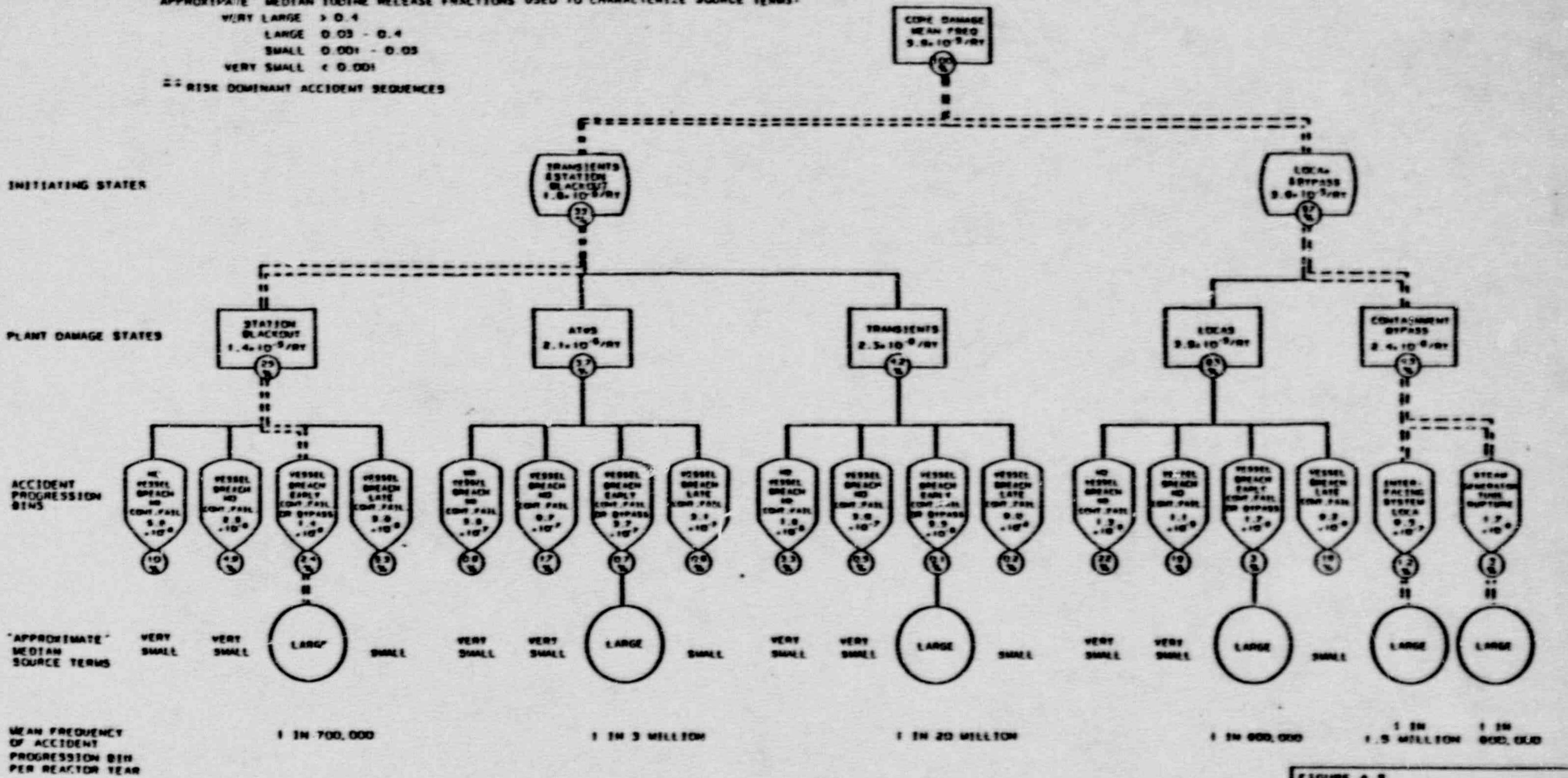


FIGURE A.5
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
NUREG-1150 ANALYSIS OF SEQUOYAH
(INTERNAL INITIATING EVENTS)

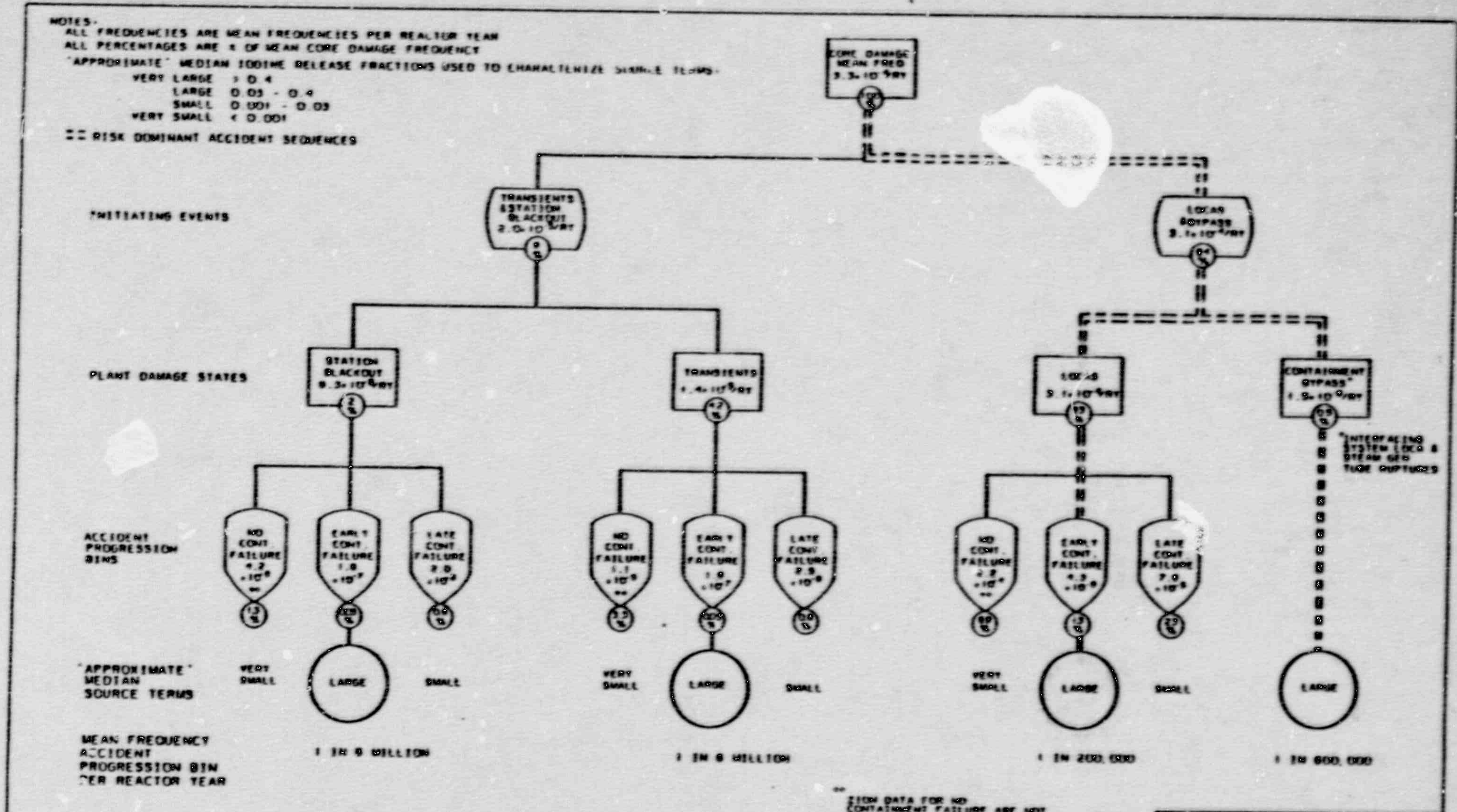


FIGURE A.9
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
NUREG-1150 ANALYSIS OF ZION
(INTERNAL INITIATING EVENTS)

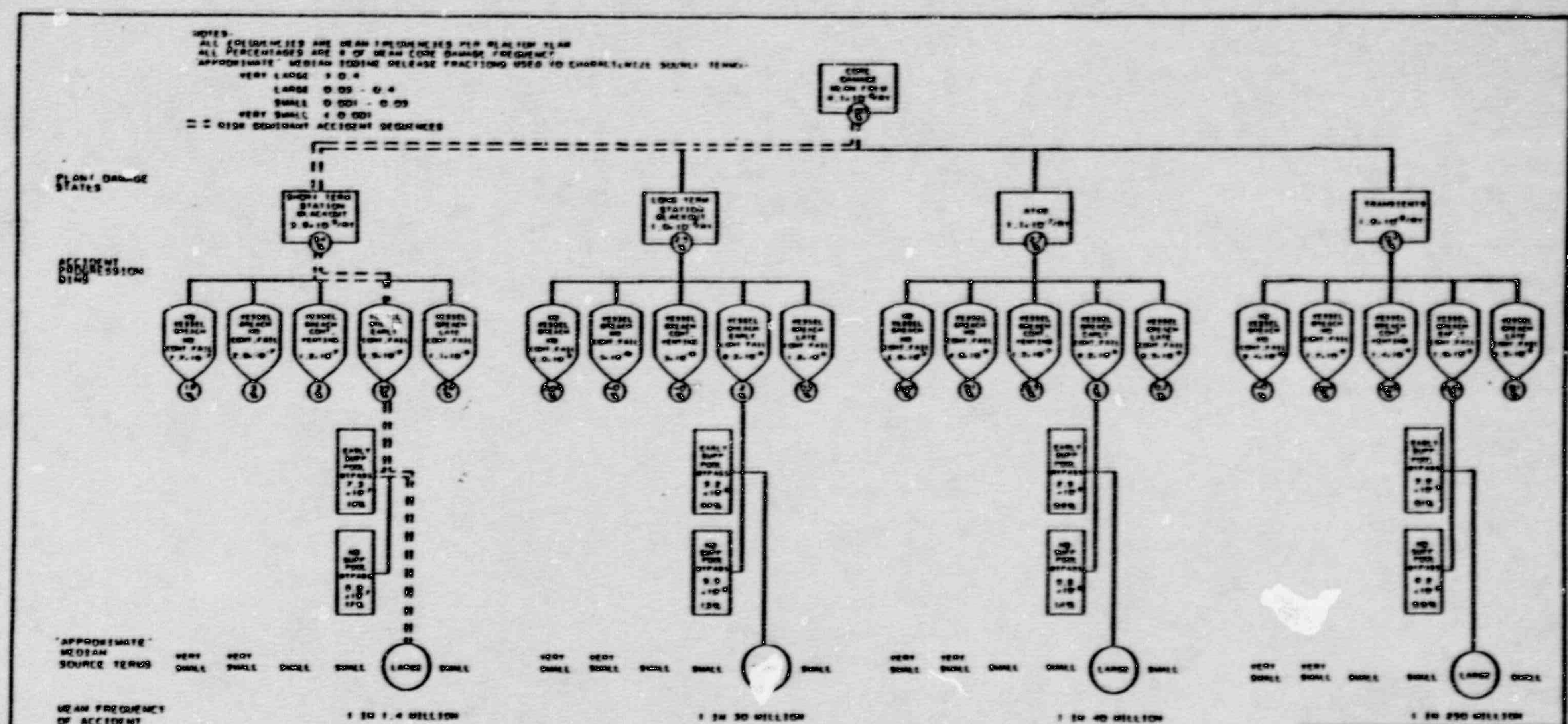


FIGURE A.7
SUMMARY OF PLANT DAMAGE STATES, ACCIDENT PROGRESSION BINS, AND SOURCE TERMS
NUREG-1150 ANALYSIS OF GRAND GULF (INTERNAL INITIATING EVENTS)

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