APPLICATION OF RISK PERSPECTIVES:
A PROCEDURES GUIDE

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I. INTRODUCTION

A. Background

The important issues in reactor safety involve rare scenarios with potentially very serious consequences. This comes about as a result of years of dedicated and concentrated efforts in the implementation of the defense-in-depth approach to reactor safety. More specifically, the application of the fundamental concepts of redundancy, diversity and physical separation in the design and installation of engineered safety features (ESFs) and numerous other safety-related equipment and the implementation of detailed operating and emergency procedures help to render a severe reactor accident a relatively rare event. This unique characteristic of major reactor accidents being rare but with serious consequences poses a difficult situation for the delineation, discussion and resolution of important safety issues. In broadest terms, the situation can be described as one in which all safety issues are potentially significant but only a few are truly significant. This necessitates a systematic ranking of safety issues in general, and the development of a conceptual framework for judging the safety significance of specific events in particular.

A systematic and comprehensive evaluation has been conducted by the Office of Nuclear Reactor Regulation (NRR) to develop a priority ranking of over 200 generic safety issues. The results of this evaluation are documented in "A Prioritization of Generic Safety Issues, NUREG-0933" (Ref. 1). There, a priority ranking involving the value-impact score and the reduction in risk was used. This type of ranking which requires a considerable amount of effort as well as expertise in the area of probabilistic risk assessment has proven to be extremely useful in the allocation of regulatory and industry resources.

One of the main charters of the Office for Analysis and Evaluation of Operational Data (AEOD) is to review, analyze and evaluate reactor operating experience and to identify and recommend actions that would prevent serious reactor accidents. Numerous events of safety significance had been evaluated by AEOD and recommendations had been made to prevent their recurrence. It is recognized (Ref. 2) that the technical merits of AEOD recommendations can be further enhanced by a discussion of the issue under consideration within the context of engineering insights and risk perspectives gained in major risk studies (Refs. 3-17) conducted in the past decade. Such a discussion can be difficult due to the large amount of information available and the rather complex methods employed in risk studies. Therefore the development of guidelines for the application of engineering insights and risk perspectives to the discussion of safety issues is desirable.

B. Objective

The overall objective of this guide is to provide the staff of AEOD a simplified method for assessing the safety significance of events or issues related to reactor operating experiences. More specifically, such a simplified method involves the utilization of engineering insights and risk perspectives gained in comprehensive risk studies as well as appropriately defining safety significance.

A second objective of this guide is to provide the staff of AEOD a source of information regarding probabilistic risk assessment and the important insights and results from major risk studies. Sufficient information has been included in this guide to make it useful for a preliminary evaluation of the safety significance of operating events without the need for additional resources.

A third objective of this guide is to assist the user to obtain well founded estimates of the reactor accident risks associated with the event/issue under consideration, hence the risk reduction associated with major AEOD recommendations. These estimates in turn will provide the recipients of AEOD recommendations a measure of the safety significance of the issues involved.

To the extent possible, both qualitative and quantitative important results from comprehensive risk studies will be supplied in this guide. No elaborate or extensive effort in the development of logic models is envisioned in the use of this guide. This is predicated on the observation that the nature of the discussion necessary to allocate regulatory resources does not require great precision but rather involves relative assessment of risks. Perhaps equally important is the observation that some safety issues are difficult to quantify with reasonable accuracy; therefore, for such issues a deterministic evaluation may be appropriate.

II. OVERVIEW

A. 7-step procedure

This guide consists of a 7-step procedure developed for the user to follow in applying risk perspectives and engineering insights from comprehensive risk studies to the delineation, discussion and resolution of safety issues and AEOD recommendations. This 7-step procedure is intended to guide the user to expeditiously gather essential information and to formulate a conceptual framework for discussing safety issues within the context of probabilistic safety studies. It is assumed that the user has only a minimal special training in the area of probabilistic risk assessment. The seven steps which will be further amplified in Section III are:

1. Define perceived safety significance. Select a definition of safety significance from 10 CFR 50.92 perceived to be appropriate for the event or issue under consideration.

2. Identify system function. Identify the principal role of the component(s) or system(s) under consideration in the areas of accident prevention, mitigation or monitoring according to the results in Tables 2-5 in Appendix A.

3. Judge system importance. Judge whether the relative importance of the system(s) under consideration is important in the contribution to reactor accident risks based on the results from Table 6 and Figs. 1-6 in Appendix A.

4. Determine relation to dominant accident sequences. Determine if the event/issue (or component/system) under consideration is related to any dominant accident sequence from the results contained in Tables 7-15 in Appendix A.

5. Investigate new accident sequences. Evaluate whether the event/issue (or component/system) under consideration is involved in any new accident sequences.

6. Assess the increase in probability of major accidents. Assess whether the event under consideration increases the probability of major reactor accidents utilizing the results in Tables 16-19 in Appendix A. The probability may be related to either the initiation frequency or core-melt frequency.

7. Evaluate the increase in consequence of major accidents. Evaluate whether there is an increase in the consequence of major reactor accidents by using available plant-specific or generic results given in Tables 20-25 in Appendix A.

Any affirmative finding in steps 3, 4, 5, 6, and 7 would indicate that the event or issue under consideration is "significant" to reactor safety. The extent of safety significance could be quantitatively gauged if sufficient information is available for an evaluation in either step 6 or 7. If such information is not available, a judgemental conclusion that the event or issue under consideration is significant to reactor safety would be adequate.

When all the findings in steps 3, 4, 5, 6, and 7 are negative, the event or issue under consideration should be judged as "insignificant" to reactor safety. If considerable uncertainties exist in the evaluation using this 7-step procedure, the event or issue under consideration should be labeled "unresolved" and will require further evaluation. These categorizations are summarized in Table 1 in Appendix A. How the various steps in the procedure can be followed is described in detail in the next section, the Procedures Guide.

B. Usefulness and Limitations

This guide offers a simplified method to conduct a preliminary evaluation of the safety significance of operating events utilizing available results from risk studies. As such, this guide is intended to be self-contained. It has included in its tables and figures a large variety of significant results, both qualitative and quantitative, from existing comprehensive risk studies so that the user may conduct a preliminary evaluation of the safety significance of an event or issue without the need to seek additional information from other references. However, this guide doe. not contain sufficient information necessary for an in-depth evaluation of the reactor accident risks or consequences. Furthermore, reactor accident risks related to external events such as tornadoes, floods, earthquakes or fire are not discussed in this guide.

Finally, it should be pointed out that currently AEOD is also looking into the feasibility of using the computer code SIEVE (Ref. 20) to compute the change in core-melt probability resulting from equipment failures, outages or various initiating events. After appropriate modifications and verifications, this code could be a useful tool in providing quantitative estimates.

III. PROCEDURES GUIDE

Step 1 Define Perceived Safety Significance

This step involves the selection of a definition of "safety significance" perceived to be appropriate for the event or issue under consideration from the three definitions codified in Part 50.92, Title 10, Code of Federal Regulations (10 CFR 50.92). More specifically, these three definitions of "safety significance" are:

An event or issue is deemed to have "safety significance" if it

1. involves a significant increase in the probability or consequences of an accident previously evaluated; or

2. creates the possibility of a new or different kind of accident from any accident previously evaluated; or

3. involves a significant reduction in a margin of safety.

These definitions may be viewed as broad criteria from which safety significance can be clearly stated. This step helps to narrow the focus of all subsequent discussions of the event or issue under consideration to a specific area of perceived safety significance. Definitions 1 and 2 are related to accident sequences previously evaluated or new accident sequences. Definition 3 can be tied to generic issues or common-mode failures. The user may have to go to steps 4, 5, 6 and 7 before deciding which definition is more appropriate for the event or issue under consideration.

Step 2 Identify System Function

This step identifies the principal function(s) or role(s) of the component(s) or system(s) under consideration in the areas of accident prevention, mitigation or monitoring. To aid in such an identification, results from two reference studies are provided in this guide. First, Tables 2 and 3 in Appendix A provide a list of various systems (Ref. 18) associated with nine essential functions for two pressurized water reactors (Sequoyah and Zion) and two boiling water reactors (Peach Bottom and Grand Gulf). These two tables were developed by the Industry Degraded Core Rulemaking Program (IDCOR) for the four selected plants. The nine functions in Tables 2 and 3 are related to providing or maintaining:

- reactor integrity
- core makeup
- core heat removal
- containment pressure suppression
- containment heat removal
- containment integrity
- containment radioactivity removal
- monitoring reactor or containment status
- miscellaneous support

Secondly, another way of categorizing systems is to list the front-line systems challenged by generic initiating events such as loss-of-coolant accidents or transients. Such a categorization was made in NUREG-1050 (Ref. 19) and reproduced in Tables 4 and 5 in Appendix A.

With Tables 2, 3, 4 and 5 as a guide, the essential function of the system under consideration can be readily identified. This in turn leads to a specific identification of the principal role the system plays in the areas of accident prevention, mitigation or monitoring. This step serves the same purpose as that of step 1, namely helping to focus all subsequent discussions of the event or issue to a specific area of perceived safety significance.

Step 3 Judge System Importance

This step utilizes the existing results of three different ranking schemes to gauge the relative importance of various systems under consideration in terms of their contribution to reactor accident risks.

The three different ranking schemes are

- Quantitative and qualitative ranking by the Industry Degraded Core Rulemaking Program (IDCOR) Task 17 (Ref. 18), the results of which are given in Table 6, Appendix A.

- "Risk achievement and Reduction Ratio" in a Battelle Columbus study (Ref. 19) sponsored by the Nuclear Regulatory Commission (NRC), the results of which are given in Figs. 1-4, Appendix A.

- "Fussel Vesely Importance Measure" screening of 15 risk studies in a study (Ref. 19) sponsored by the NRC, the results of which are given in Figs. 5-6, Appendix A.

The user should screen all these results to assess whether or not the systems under consideration have been ranked as important to reactor accident risks. Once a system(s) has been ranked by any one of these studies as important, consider this step completed. If not, more work may be required.

The aim of this step is to obtain an approximate ranking of the system under consideration in terms of its importance to reactor accident risks. Such an approximate ranking would aid in the discussion of safety significance. First, the observation that a system has been ranked as very important by a previous study (IDCOR or NRC sponsored studies) would complement the information extracted from steps 4, 5, 6 or 7 in this procedure guide. Secondly, in the absence of appropriate information, a ranking of "very important" to reactor accident risks may provide an adequate basis for the user to assign safety significance to the system(s) under consideration. A telling example is the auxiliary feedwater system (AFW) which was ranked by all three aforementioned schemes as very important. Therefore the user could, if he decides not to conduct any further quantitative evaluations, state that major failures or degradation of the AFW system would be deemed significant to reactor safety.

The user should be aware that a precise ranking of relative importance of systems has not been established. This is due to the following reasons. First, there is no indust consensus on the definition of system importance. Second, the importance o stem is invariably tied to plant-specific dominant accident sequences as well as site characteristics. Third, the systems have interdependence the extent of which is particularly acute between the front-line systems and their associated support systems. Thus there is no industry-wide consensus as to the importance of systems in terms of their contribution to reactor accident risks. However, based on results in Table 6 and Figs. 1-6, a somewhat consistent and generic set of important systems is:

Pressurized water reactor (PWR)

- Auxiliary feedwater system (AFW)
- High pressure injection system (HPI)
- Containment spray system

Boiling water reactor (BWR)

- Residual heat removal system (RHR)
- Reactor protection system (RPS)
- High pressure coolant injection system (HPCI)
- Reactor core isolation cooling system (RCIC)

Step 4 Determine Relation to Dominant Accident Sequences

This step determines if the event/issue (or component/system) under consideration is related to any dominant accident sequences. One of the major conclusions of the results of the completed comprehensive risk studies is that reactor accident risks stem primarily from a few accident sequences. In other words, only a few accident sequences among the numerous possible sequences considered dominate the reactor accident risks. Reactor accident risks can be measured in terms of reactor core-melt frequency and its associated consequences. Thus, to achieve a significant reduction in reactor accident risks would have to involve a change in hardware or in procedures which reduce the frequency or the consequence of the dominant accident sequences. The user should be aware of the fact that events of low consequences but of high frequency are not addressed here. To what extent they impact on reactor accident risks can be a safety concern.

To a very large extent, dominant accident sequences are related to the type of reactor (PWR or BWR), the variation in systems design and configuration within a given reactor type, and the difference in operating and emergency procedures, and in test and maintenance practices. This results in a rather large volume of information pertaining to the dominant accident sequences for various plants. Provided in this step are information at two levels of detail for the user to screen and to determine whether or not the event or issue under consideration is involved in any dominant accident sequences:

A. Generic PWR and BWR dominant accident sequences

   These sequences are the results of the Accident Sequence Evaluation Program (ASEP) (Ref. 19) sponsored by the Office of Nuclear Regulatory Research, and are given in Tables 7 and 8, Appendix A.

B. Plant-specific dominant accident sequences

   (i) A list of all the probabilistic safety studies completed to date given in Tables 9 and 10, Appendix A.

(ii) A comparison of dominant accident sequences for 11 plants, given in Table 11, Appendix A.

(iii) A brief description of some of the dominant accident sequences for Zion, Sequoyah, Peach Bottom and Grand Gulf, from the IDCOR program Tasks 17 and 23, given in Tables 12, 13, 14 and 15 in Appendix A.

For the user who can budget sufficient time and effort in this step and if the plant under consideration has been studied in a risk study, a screening and evaluation of plant-specific dominant accident sequences would be appropriate. Otherwise, the user can limit the screening to the generic PWR and BWR dominant accident sequences listed in Tables 7 and 8, Appendix A.

If in step 1 it is determined that the failure of the system under consideration involves a common-cause failure, such a common-cause failure should be studied here to assess whether or not it is related to any dominant accident sequences. This follows the logic that a common-cause failure is important when it is related to a dominant accident sequence or it creates the possibility of a new accident sequence not previously evaluated.

The user should be aware of the significant role human errors play in contributing to reactor accident risks. They can occur in practically any stage of an accident. For example, human errors can be involved in the initiation, mitigation and/or monitoring of an accident. They can be errors related to the performance of test and maintenance, design or installation. They can be either errors in commission or in omission. In general it is difficult to quantitatively estimate the various probabilities associated with human errors. However, a large amount of useful information is contained in NUREG/CR-1278, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications," which should be consulted if the user needs to obtain a quantitative estimate of a failure probability related to human errors.

Step 5 Investigate New Accident Sequences

This step investigates if the event/issue (or component/system) under consideration is involved in any new accident sequences. This step should only be taken if the answer to the effort in step 4 is negative, namely the event/issue under consideration is not related to any dominant accident sequences. Here, a substantial amount of effort may be required to identify the new accident scenario, to assess its probable causes and how it may progress. The safety significance of a new accident sequence is immediately apparent if the likelihood of its occurence or its consequence is assessed to be comparable to those of the dominant accident sequences.

An accident sequence is a new sequence if it has not been previously evaluated. In principle the user can determine whether or not an accident sequence under consideration is a new one by comparing it against all the accident sequences investigated in past studies. In practice, this is seldom feasible. Thus, initially the user should consult an individual who has considerable experience in evaluating accident sequences.

Furthermore, the role of any common-cause failure identified in step 1 should be investigated here as well. The user should be aware that a quantitative evaluation of a common-cause failure and its associated impact also requires a large amount of effort, particularly if such a common-cause failure involves more than one system. For existing common-cause failure data regarding valves, pumps, instrumentation and control assemblies, the user should consult the following references:

- "Common-Cause Fault Rates for Valves," NUREG/CR-2770, Feb. 1983.
- "Common-Cause Fault Rates for Pumps," NUREG/CR-2098, Feb. 1983.
- "Common-Cause Fault Rates for Instrumentation and Control Assemblies," NUREG/CR-2771, Feb. 1983.

Step 6 Assess the Increase in Probability of Major Accidents

This step assesses the increase in the probability of major (dominant or new) reactor accidents. The probability can be either the initiation frequency or core-melt frequency.

Here, major reactor accidents refer to the dominant accident sequences identified in step 4 or the new accident sequence identified in step 5. A very large amount of data exists as to the accident initiation frequency, the estimated probabilities of system and component failures, and the estimated containment failure modes and their associated probabilities. For plant-specific data, the user should screen the available probabilistic safety study reports listed in Tables 9 and 10 given in step 4. For generic results, the following data should be evaluated.

- Baseline frequencies for PWR transient initiators, given in Table 16, Appendix A.
- Baseline frequencies for BWR transient initiators, given in Table 17, Appendix A.
- Core-melt probabilities for PWR (Surry) dominant accident sequences predicted by the Reactor Safety Study and re-evaluated by Brookhaven National Laboratory, given in Table 18, Appendix A.
- Core-melt probabilities for BWR (Peach Bottom) dominant accident sequences predicted by the Reactor Safety Study and re-evaluated by Brookhaven National Laboratory, given in Table 19, Appendix A.

These tables should be studied to obtain a broad overview of the relative frequencies associated with various dominant accident sequences. The containment failure modes and their associated probabilities are discussed in the next step. The user should be aware that variations in both initiator frequency and core-melt frequency can be substantial from plant to plant.

The user may need to consult the results related to system failure probabilities and basic component failure data (e.g., valves, pumps, instrumentation and control assemblies) contained in the various completed risk studies (see Tables 9 and 10). The following references are also helpful.

- "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," NUREG/CR-1363, vols 1 & 2, June, 1980
- "Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants," NUREG/CR-1205, Jan. 1980
- "Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants," NUREG/CR-1331, Feb. 1980
- "Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants," NUREG/CR-1362, March 1980
- "Common-Cause Fault Rates for Valves," NUREG/CR-2770, Feb. 1983
- "Common-Cause Fault Rates for Pumps," NUREG/CR-2098, Feb. 1983
- "Common-Cause Fault Rates for Instrumentation and Control Assemblies," NUREG/CR-2771, Feb. 1983

Quantitative estimates of the probability of a system failure can be obtained using the fault tree technique. Detailed descriptions of how this may be conducted are given in the "Fault Tree Handbook," NUREG-0492. A copy of this handbook has been provided to the user and it should be used in conjunction with the procedures guide when necessary.

The user should be aware that considerable uncertainties exist regarding the estimated core-melt probabilities in various safety studies in general, and regarding the system/component failure probabilities in particular. These uncertainties are primarily associated with those involved in the collection of failure data, the modeling of systems and the issue of completeness. Therefore it is important to be consistent in the assessment of probability increases so that uncertainty in the existing results will not become an overwhelming issue. Furthermore, it is generally helpful when key assumptions are stated explicitly, and to the extent feasible, stated whether or not they are regarded as those to be used in a bounding calculation.

Step 7 Evaluate the Increase in Consequence of Major Accidents

This step evaluates the increase in the consequence of major reactor accidents by using plant-specific or generic results.

The modeling of the consequences of a severe reactor accident is exceedingly complex, and furthermore, it is frequently updated to reflect new insights gained in research activities. Some of the key issues involved are how the accident may progress during core degradation; how a reactor vessel and the containment fails; the estimated fraction of radioactive inventory being released and its associated energy; the modeling of atmospheric dispersion; and the role of emergency evacuation. These issues had been somewhat extensively discussed in the completed full-scope probabilistic safety studies and the user should consult the reports listed in Table 9 for details. Here, two summaries are offered in looking at containment failure modes and their associated releases. The first summary is associated with the containment failure modes, their estimated probabilities and the release categories defined in the Reactor Safety Study (WASH-1400). The second summary is a broad categorization of release functions given in NUREG-1050.

The user is cautioned that the Reactor Safety Study results which are useful in providing a consistent framework for comparison should not be used as absolute estimates. They are subject to revision as more current knowledge is gained through the study of various important degraded-core phenomena and from the results of source term research. Containment failure probabilities are also estimated in the Zion safety study and in the Limerick safety study.

Tables 20 and 21 in Appendix A give the probabilities for various containment failure modes for a PWR and a BWR as defined in the Reactor Safety Study. These containment failure mode probabilities are necessarily dependent on the accident sequences. They are required in assessing the release categories associated with the various accident sequences.

The PWR and BWR release categories are defined in Table 22, Appendix A. Briefly stated, they are categories of radioactive release defined by the

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6″

release fraction of various radioactive isotopes, together with the associated time of release, duration of release, and the elevation and energy involved in the release.

The PWR dominant accident sequences and their associated release categories are given in Table 23 in Appendix A and the BWR dominant accident sequences together with their associated release categories are given in Table 24, Appendix A. These tables provide a consistent source of information for comparing the severity of release (consequence) associated with the dominant accident sequences. For example, from Table 23, the accident sequence TMLB'-δ which denotes a station-blackout accident (TMLB') leading to the containment failure mode δ which is containment overpressure rupture is in the PWR release category 2. The same accident TMLB' with a more benign containment failure mode ε which is a basemat melt-through (sequence TMLB'-ε) is in the PWR release category 6. From Table 22, the PWR release category 2 involves substantially higher fractions of radioactivities released to the environment than those associated with a PWR release category 6. Thus the consequence of accident sequence TMLB'-δ is more severe (by a factor of 3 to 10,000 depending on the isotope) than that of sequence TMLB'-ε.

With Tables 22 and 23 as a source of information, the user can proceed to gauge the reduction in releases, hence the improvement of reactor safety, associated with a recommendation. This is predicated on such a recommendation being clearly related to a dominant sequence in one of these 2 tables and the containment failure modes associated with such an accident being readily identifiable. To illustrate, consider the same sequences TMLB'-δ and TMLB'-ε mentioned before. If a recommendation were to have the impact that the likelihood of occurrence of TMLB'-δ would be reduced and at the same time the likelihood of occurrence of TMLB'-ε would be increased, such a recommendation would lead to a substantial reduction in radioactive releases, hence significant improvement of reactor safety. Another way of improving reactor safety is, as discussed in Step 6, via the reduction of the likelihood of occurrence of the accident TMLB' itself.

The second summary of results pertaining to accident consequences which were stated in NUREG-1050 is given in Table 25, Appendix A. It summarizes the range of release fractions surveyed in eight probabilistic risk studies (Big Rock Point, Indian Point 2 and 3, Zion, Limerick, Shoreham, Surry-1, Peach Bottom and Yankee Rowe) for three generic types of containment failures. They are:

- Severe containment failures which include early overpressure rupture and containment bypass,

- Late containment failures with ESFs functioning,

- No containment failure.

The user can use Table 25 to gauge the relative magnitude of release for Xe, Ke, I, Cs and Te associated with the above generic types of containment failures. Note that the range of variation is fairly large for the risk studies surveyed. Despite such a large range of variation, a trend is clear: early containment failures lead to the largest radioactive releases.

IV. SELECTED ILLUSTRATIONS

Three events are chosen to illustrate the use of this procedures guide. These three events, one of which was reported as an AEOD case study, were selected because they had been extensively evaluated. The usefulness of this guide will be further tested in an in-depth examination of other recently completed AEOD case studies.

A. Example 1 Major Degradation of Primary Containment Boundary at the Palisades Nuclear Plant

This event involved an inadvertently open 4-inch bypass line around the main 48-inch containment purge valve for a period of 17 months at the Palisades Nuclear Plant during 1978-1979. Two manual valves on the 4-inch bypass line were left locked open instead of locked closed as a result of a personnel error and remained undetected for the ensuing 17-month period. This event was described in detail in NUREG-0090, vol 2., No. 3, "Report to Congress on Abnormal Occurrences, July-September 1979".

Step 1 Define perceived safety significance. Select a definition of "safety significance" perceived to be appropriate for the event or issue under consideration.

For this event, either the first or the third definition of safety significance in step 1 can be used. The first definition can be used because an open 4-inch line in the containment has significant impact on the releases associated with all dominant accident sequences in which the containment is assumed to be intact during the core-melt process. The third definition can also be selected because a 4-inch opening in the containment for a 17-month period involved a significant reduction in safety margin which more specifically in this case is related to the containment integrity.

Step 2 Identify system function. Identify the principal role of the component(s) or system(s) under consideration in the areas of accident prevention, mitigation or monitoring.

The inadvertently open 4-inch bypass line for a 17-month period caused a major degradation of containment integrity which in Table 2 was identified as an important function in accident mitigation. More specifically, the containment is relied upon to prevent small radioactive releases associated with minor accidents from reaching the environment, and to mitigate major releases from severe accidents.

Step 3 Judge system importance. Judge, according to existing ranking results, the relative importance of the system(s) under consideration in terms of its contribution to reactor accident risks.

The role of the containment was not addressed in Table 6 or in Figs. 1-6. This step can be omitted.

Step 4 Determine relation to dominant accident sequences. Determine if the event/issue (or component/system) under consideration is related to any dominant accident sequences.

The containment being open would adversely impact all generic classes of PWR dominant accident sequences (Table 7) in which the containment was assumed to be intact during the core-melt processes. These generic classes of PWR dominant accident sequences include transients or small-break LOCAs with loss of core cooling. The adverse impact is related to the failure to mitigate early radioactive releases.

Step 5 Investigate new accident sequences. Investigate if the event/issue (or component/system) under consideration is involved in any new accident sequences.

An open containment by itself is not an accident initiator. Its involvement in any new accident scenarios will be in the same role, namely the degradation in mitigating radioactive releases, as that in dominant accident sequences.

Steps 6 and 7 Assess the increase in probability or consequence of major accidents.

In the Reactor Safety Study results (Table 20), the containment failure mode β was used to denote the containment failure resulting from an inadequate isolation of containment openings and penetrations and was assessed to have a probability of 10^{-4} per reactor year. Here, with a 4-inch opening for 17 months, the probability for β on a per reactor year basis would be unity which is higher than the 10^{-4} value given in Table 20 by four orders of magnitude. Therefore, going to Table 23, the accident sequences in the PWR 4 and PWR 5 release categories which are tied to the β containment failure mode would now have higher probabilities (by four orders of magnitude) than those assessed in the reactor safety study. More specifically, these higher probabilities are in the range of 10^{-7} and 10^{-4} respectively for the accidents in the PWR 4 and PWR 5 release categories. This magnitude of increase in probability for these accident sequences represents a significant increase in hazards.

Concluding remarks for Example 1

From the criteria stated in Table 1, this event is judged significant to reactor safety because it is related to all PWR dominant accident sequences in which the containment was previously assumed to be intact during the core-melt process (Step 4), and involved a significant increase (four order of magnitude) in probability of accident sequences in the PWR 4 and PWR 5 release categories (Steps 6 and 7).

B. Example 2 Injection Check Valve Stuck Open on the Residual Heat Removal System at Hatch-2

The injection check valve on a 24-inch injection line of the residual heat removal (RHR) system at Hatch 2 was found open and could not be reclosed on Oct. 28, 1983. The root cause of this event was a maintenance error on the air operator of the check valve which caused it to hold the check valve open even though it was not in the test mode. This error went undetected for a four-month period of power operation. This event was described in detail in an AEOD engineering evaluation report (AEOD/E414, May, 1984).

Step 1 Define perceived safety significance.

The first definition in step 1 would be appropriate. The stuck open check valve could lead to an interfacing LOCA involving the reactor coolant system (RCS) and the low pressure RHR system if a single failure of the normally closed motor-operated injection valve which is situated upstream of the stuck open check valve were to occur. Such an interfacing LOCA could disable the RHR system as well as bypassing the containment.

Step 2 Identify system function.

The RHR system is relied upon to provide core heat removal as well as containment heat removal. (Tables 3 and 5).

Step 3 Judge system importance.

The RHR system is one of the most important systems in a BWR because its unavailability contributes significantly to reactor accident risks. (Table 6, and Figs. 4 and 6).

Step 4 Determine relation to dominant accident sequences.

An interfacing LOCA involving the RCS and the RHR system is perceived to be a severe accident but not a likely one because it ordinarily involves the failure of both the check valve and the motor-operated injection valve. Here, the check valve being stuck open for a four-month period substantially increased the likelihood of occurrence of such an interfacing LOCA. More discussion is offered in step 6 below.

Step 5 Investigate new accident sequences.

This step can be omitted.

Step 6 Assess the increase in probability of major accidents.

Intuitively, a single failure of the motor-operated injection valve (inadvertent opening) has a higher probability than the concurrent happening of its failure and the injection check valve failing open. Therefore qualitatively, this event presented a situation in which the likelihood of occurrence of an interfacing LOCA had been increased.

A somewhat more quantitative argument can also be derived as follows. The probability of the motor-operated injection valve failing open together with the check valve failing open is the probability of an interfacing LOCA. It is of the order of 10^{-7} per reactor year as estimated in various safety studies (Refs. 3, 6, and 8). This estimate was in part based on the observation that common-cause failures involving both valves are very unlikely.

The probability of a single failure of the motor-operated injection valve was estimated in the reference AEOD engineering evaluation to be of the order of 2×10^{-4} per reactor year. This estimate had two components: the spurious actuation of the motor-operated injection valve, and the instantaneous rupture of the injection valve disk. The spurious actuation of a normally closed motor-operated valve was assessed to have a failure rate of approximately 10^{-8} per hour as estimated in IEEE Standards-500 (Ref. 35). The rate of disk rupture was determined to be of the order of 10^{-7} per hour in a recent study (Ref. 21) and in the Reactor Safety Study. These failure rates then gave a failure probability of the motor-operated injection valve (inadvertent opening due to spurious actuation or disk rupture) of 2×10^{-4} for a four-month period assuming a 80% capacity factor $((1 \times 10^{-7}/\text{hr} + 10^{-8}/\text{hr})(120 \text{ days}) (24 \text{ hrs/day}) (0.8) = 2 \times 10^{-4})$.

Therefore an increase of probability of three orders of magnitude was observed if no credit could be taken for the held-open check valve to reclose. The reclosure of the held-open check valve is not certain for several reasons. First, there are uncertainties in the extent of actuator interference and the flow conditions associated with a sudden discharge of reactor coolant. Second, if suddenly forced to reclose, the valve disk may not survive the dynamic loadings from such a rapid closure. Finally, holding a check valve open for a prolonged period of time may increase the likelihood that the check valve will be stuck open from causes not related to the actuator interference (e.g., corrosion of the hinge pin or loose part obstruction).

Step 7 Evaluate the increase in consequence of major accidents.

This step can be omitted.

Concluding remarks for Example 2.

This event was judged significant to reactor safety according to the criteria stated in Table 1 because it involves a system (RHR/LPCI) important to reactor accident risks (step 3); it is related to an accident sequence with severe consequence (interfacing LOCA involving the RCS and RHR system, step 4); and it involves a significant increase in the probability of such an interfacing LOCA (step 6).

C. Example 3 Steam Binding Of AFW Pumps

The generic safety implications of backleakage to the auxiliary feedwater (AFW) system in PWRs were delineated in a recent AEOD case study report (AEOD/C404, Ref. 43). The report stated that operational experience had shown that on numerous occasions an AFW pump was rendered inoperable due to steam binding resulting from the leakage of hot feedwater to the AFW system. These events highlight the potential for a common cause failure of the AFW system due to steam binding because the AFW pumps are connected by common piping (discharge header and/or recirculation piping) with only a single check valve to prevent backleakage of hot water to a second or third pump. See Ref. 43 for detailed discussions.

Step 1 Define perceived safety significance.

The first definition in step 1 given on page 7 would be appropriate. The failure of the AFW system is involved in many PWR accident sequences previously evaluated. Of particular importance are two sequences: one involves the total loss of reactor core decay heat removal (sequence TML), and another involves a station blackout scenario (sequence TMLB', see symbol keys for TML and TMLB' in Table 18).

Step 2 Identify system function.

The AFW system is relied upon to remove the decay heat from the reactor core during accident conditions involving a transient or a small-break LOCA (Tables 2 and 4).

Step 3 Judge system importance.

The AFW system is one of the most important systems in a PWR because it is one of the front-line systems being challenged early in numerous accident sequences of which the loss of core heat removal and the station blackout scenarios are two dominant sequences (see Table 4). Furthermore, AFW unavailability contributes significantly to reactor core melt frequency as indicated in Figs. 1, 2, 3 and 5.

Step 4 Determine relation to dominant accident sequences.

As stated in steps 1 and 3, the operability of the AFW system is involved in two dominant accident sequences. In the loss of core heat removal scenario (sequence TML), the loss of main feedwater and the subsequent failure of the AFW system is postulated to lead to the uncovery of the reactor core (Refs. 3 and 10). In the station blackout scenario (sequence TMLB'), all ac power, off-site and on-site, is lost at the beginning of the accident. The failure of the steam turbine driven train of the AFW system will constitute the total AFW system failure because the other AFW trains are dependent on ac power which has been lost.

Step 5 Investigate new accident sequences.

This step can be omitted.

Step 6 Assess the increase in probability of major accidents.

AEOD/C404 gave the following data:

Number of events in 1983	Number of AFW Pumps involved (assumed failed) due to steam binding
10	1
3	2

Based on these data, the failure probabilities of AFW pumps due to steam binding can be estimated as follows. First, assume that there are 47 operating PWR plants each with 3 AFW pumps subject to 15 demands per reactor year (RY) based on 12 surveillance tests and 3 AFW challenges after reactor trips. Then,

P_1 = failure probability of 1 AFW pump due to steam binding

$$= \frac{\text{(10 events involving 1 AFW pump failure/RY)}}{\text{(47 plants)(3 AFW pumps/plant)(15 demands/RY)}}$$

$= 4.7 \times 10^{-3}$/demand.

P_2 = failure probability of 2 AFW pumps due to steam binding

$$= \frac{\text{(3 events involving 2 AFW pumps/RY)}}{\text{(47 plants)(3 AFW pumps/plant)(15 demands/RY)}}$$

$= 1.5 \times 10^{-3}$/demand.

P_3 = failure probability of all 3 AFW pumps due to steam binding

$= 0.1\ P_2$

$= 1.5 \times 10^{-4}$/demand,

where the value 0.1 was estimated to be the conditional probability that the third pump would fail given 2 pumps have already failed, which is based on the common-cause dependency of the hardware between trains having the same design and subject to the same environment, and the same test and maintenance procedures.

Sequence TML

For the Sequoyah plant (Ref. 11, Chapter 9.1.1), the following comparison can be made using the estimated P_3 value for event L (failure of AFW) for sequence TML which is a dominant accident sequence in the PWR-3 release category.

	Event T, number of transients/RY	Event M, loss of power conversion	Event L, loss of AFW	Sequence TML
No steam binding of any AFW pump	7	10^{-2}	4×10^{-5}/demand	2.8×10^{-6}/RY
Steam binding of all AFW pumps	7	10^{-2}	1.5×10^{-4}/demand	1.1×10^{-5}/RY

Therefore the sequence probability of TML is increased by approximately a factor of 4. This increase doubles the contribution of the TML sequence to the probability of the PWR-3 release category which is already the most probable release category at Sequoyah.

Sequence TMLB'

For the Sequoyah plant, the unavailability of the AFW system given a total loss of ac power is 1.9×10^{-2} (Ref. 11, Appendix B.17). The failure probability of 1 AFW pump due to steam binding, in this case the turbine-driven pump, is represented by P_1 which has been estimated to be 4.7×10^{-3}/demand. Since the steam binding phenemenon had not been considered until now, the P_1 value should be added to the 1.9×10^{-2} value. Then this P_1 value represents about 20% of the total unavailability of the AFW system when all ac power is lost. As such, the contribution of pump failure as a result of steam binding to total AFW system unavailability, hence sequence TMLB' probability, is marginally significant. The contribution here is not as great as that in the

TML sequence discussed earlier in which the sequence probability was increased by a factor of four. However, since TMLB' is assessed to lead to PWR-1 and PWR-2 release categories which involve higher release fractions than a PWR-3 release category, a 20% increase in TMLB' probability should be considered marginally significant.

Step 7 Evaluate the increase in consequence of major accidents.

This step can be omitted.

Concluding remarks for Example 3

From the criteria stated in Table 1, the failure of AFW pumps due to steam binding is judged significant to reactor safety because it involves a system (AFW) important to reactor accident risks (step 3); it is related to at least two dominant accident sequences (TML and TMLB', step 4); and it leads to a significant increase in the probability of sequence TML (by a factor of 4) and a marginally significant increase in the probability of sequence TMLB' (about 20%, Step 6).

V. SUMMARY

This procedures guide has been developed to provide the staff of AEOD a simplified method for assessing the safety significance of events or issues related to reactor operating experiences. It consists of a 7-step procedure for the user to follow in applying risk perspectives and engineering insights from comprehensive risk studies to such an assessment of safety significance. It also serves as a source of information regarding probabilistic risk assessment and the important engineering insights and risk perspectives gained from various comprehensive risk studies.

This guide was written for the user with only a minimal special training in the area of probabilistic risk assessment. No elaborate effort in the development of logic models or detailed numerical calculation is envisioned in the use of this guide. It is intended to guide the user, in a step by step fashion, to important categories of information distilled from various risk studies which consist of systematic and integrated evaluations of plant performance. A large variety of such information has been gathered in Appendix A. These categories of information would form the basis from which the user could obtain an estimate, either qualitative or quantitative, of the risk reduction potential associated with a major recommendation. Selected examples are given to illustrate the use of the guide.

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Appendix A

Tables and Figures

Table 1
EVENT CATEGORIZATIONS

1. "Significant to reactor safety" event/issue
   An event or issue is judged significant to reactor safety if there is at least one affirmative answer to the following questions:

   - Is the system(s) under consideration important in terms of its contribution to reactor accident risks? (procedure step 3)

   - Is the event/issue (or component/system) related to any dominant accident sequences? (procedure step 4)

   - Is the event/issue (or component/system) involved in any new accident sequences? (procedure step 5)

   - Is there a potential of increase in the probability of major reactor accidents? (procedure step 6)

   - Is there a potential of increase in the consequence of major reactor accidents? (procedure step 7)

2. "Insignificant to reactor safety" event/issue

   An event or issue is judged insignificant to reactor safety if all answers to the above five questions are negative.

3. "Unresolved" event/issue

   An event or issue is judged unresolved (requiring further evaluation) if substantial uncertainties exist when answering the above five questions.

Table 2
A LIST OF SYSTEMS BY FUNCTION
FOR TWO PRESSURIZED WATER REACTORS (SEQUOYAH AND ZION)

FUNCTION	SEQUOYAH	ZION
Reactor integrity	1. Reactor pressure vessel 2. Reactor coolant pressure boundary 3. Safety/relief valves	1. Reactor pressure vessel 2. Reactor coolant pressure boundary 3. Safety/relief valves
Core makeup	1. Charging system 2. Upper head injection system 3. Accumulator system 4. High pressure injection/recirculation system 5. Low pressure injection/recirculation system	1. Charging system 2. Accumulator system 3. High pressure injection/recirculation system 4. Low pressure injection/recirculation system
Core heat removal	1. Reactor protection system a. Control rods and associated systems b. Boron injection system 2. Safety/relief valves 3. Steam system a. Steam generators b. Main & auxiliary feedwater systems c. Power conversion system d. Secondary system steam relief 4. RHR system a. High pressure recirculation system b. Low pressure recirculation system c. RHR heat exchangers component d. Component cooling water system e. Essential raw cooling water system	1. Reactor protection system a. Control rods and associated systems b. Boron injection system 2. Safety/relief valves 3. Steam system a. Steam generators b. Main & auxiliary feedwater systems c. Power conversion system d. Secondary system steam relief 4. RHR system a. High pressure recirculation system b. Low pressure recirculation system c. RHR heat exchangers d. Component cooling water system e. Service water system 5. Charging/HPI & letdown system (feed and bleed)

Step 2

Table 2 (continued)
A LIST OF SYSTEMS BY FUNCTION
FOR TWO PRESSURIZED WATER REACTORS (SEQUOYAH AND ZION)

FUNCTION	SEQUOYAH	ZION
Containment pressure suppression	1. Ice condenser system/air return fan system	1. Containment spray injection/recirculation system
	2. Containment spray injection/recirculation system	2. Containment fan cooling system
	3. RHR system	3. RHR system
	a. High pressure recirculation system	a. High pressure recirculation
	b. Low pressure recirculation system	b. Low pressure recirculation
	c. RHR heat exchangers	c. RHR heat exchangers
	d. Component cooling water system	d. Component cooling water system
	e. Essential raw cooling water system	e. Service water system
	4. Hydrogen control system	4. Hydrogen control system
Containment heat removal	1. Containment spray injection/recirculation system	1. Containment spray injection/recirculation system
	2. RHR system	2. Containment fan cooling system
	a. High pressure recirculation system	
	b. Low pressure recirculation system	
	c. RHR heat exchanger	
	d. Component cooling water system	
	e. Essential raw cooling water system	
	3. Natural heat loss to environment (passive)	3. RHR system
		a. High pressure recirculation system
		b. Low pressure recirculation system
		c. RHR heat exchanger
		d. Component cooling water system
		e. Service water system
		4. Natural heat loss to environment (passive)

Step 2

Table 2 (continued)
A LIST OF SYSTEMS BY FUNCTION
FOR TWO PRESSURIZED WATER REACTORS (SEQUOYAH AND ZION)

FUNCTION	SEQUOYAH	ZION
Containment integrity	1. Containment building and isolation system 2. Systems serving the function of containment pressure suppression or containment heat removal	1. Containment building and isolation system 2. Systems serving the function of containment pressure suppression or containment heat removal
Containment radio-activity removal	1. Ice condenser system 2. Containment spray injection/recirculation system 3. Containment interior and equipment surfaces (passive)	1. Containment spray injection (sodium hydroxide)/recirculation system 2. Containment fan cooling system 3. Containment interior and equipment surfaces (passive)
Monitoring reactor or containment status	1. Monitors and associated components for reactor a. Reactor water level/pressurizer level b. Hot and cold-leg temperatures c. Reactor pressure d. PORV and SRV position 2. Monitors and associated components for containment a. Temperature b. Pressure c. Sump water level/temperature d. Radiation e. Hydrogen concentration f. Isolation valves position	1. Monitors and associated components for reactor a. Reactor water level/pressurizer level b. Hot and cold-leg temperatures c. Reactor pressure d. PORV and SRV position 2. Monitors and associated components for containment a. Temperature b. Pressure c. Sump water level/temperature d. Radiation e. Hydrogen concentration f. Isolation valves position
Miscellaneous support	1. Electrical power a. Offsite AC b. Emergency diesels c. DC power d. Interconnections and shared systems	1. Electrical power a. Offsite AC b. Emergency diesels c. DC power d. Interconnections and shared systems

Step 2

Table 2 (continued)
A LIST OF SYSTEMS BY FUNCTION
FOR TWO PRESSURIZED WATER REACTORS (SEQUOYAH AND ZION)

FUNCTION	SEQUOYAH	ZION
	2. Cooling water systems	2. Cooling water systems
	a. Component cooling water system	a. Component cooling water system
	b. Essential raw cooling water systems	b. Service water system
	c. Heat exchangers	c. Heat exchangers
	d. Ultimate heat sink	d. Ultimate heat sink
	3. Actuation, control and instrumentation systems for	3. Actuation, control and instrumentation systems for
	a. Reactor protection system	a. Reactor protection system
	b. ESFs	b. ESFs
	c. Injection-recirculation switch-over system	c. Injection-recirculation switch-over system
	4. Compressed air systems	4. Compressed air systems
	5. Fire protection systems	5. Fire protection systems
	6. Sampling systems	6. Sampling systems
	7. Systems related to control room	7. Systems related to control room
	a. Shielding	a. Shielding
	b. HVAC	b. HVAC
	c. Computer system	c. Computer system
	d. Communication system	d. Communication system
	e. Emergency control room	e. Emergency control room

Table 3
A LIST OF SYSTEMS BY FUNCTION
FOR TWO BOILING WATER REACTORS (PEACH BOTTOM AND GRAND GULF)

FUNCTION	PEACH BOTTOM	GRAND GULF
Reactor integrity	1. Reactor pressure vessel 2. Reactor coolant pressure boundary 3. Safety/relief valves 4. Automatic depressurization system	1. Reactor pressure vessel 2. Reactor coolant pressure boundary 3. Safety/relief valves 4. Automatic depressurization system
Core makeup	1. Feedwater system 2. High pressure coolant injection system 3. Low pressure coolant injection (RHR/LPCI mode) 4. Reactor core isolation cooling 5. Core spray system 6. Automatic depressurization system	1. Feedwater system 2. High pressure core spray system 3. Low pressure core spray 4. Low pressure coolant injection (RHR) 5. Reactor core isolation cooling system 6. Automatic depressurization system
Core heat removal	1. Reactor protection system a. Control rods drive system b. Standby liquid control system c. Recirculation pump trip 2. Power conversion system a. Turbine b. Main condenser & associated systems c. Bypass valves 3. Suppression pool, including a. Relief valves 4. RHR systems a. Shutdown cooling mode b. LPCI mode	1. Reactor protection system a. Control rods b. Standby liquid control c. Recirculation pump trip 2. Power conversion system a. Turbine b. Main condenser & associated systems c. Bypass valves 3. Suppression pool, including a. Relief valves b. Pool makeup systems 4. RHR systems a. Shutdown cooling mode b. LPCI mode c. Steam condensing mode (with RCIC)

Table 3 (continued)
A LIST OF SYSTEMS BY FUNCTION
FOR TWO BOILING WATER REACTORS (PEACH BOTTOM AND GRAND GULF)

FUNCTION	PEACH BOTTOM	GRAND GULF
Containment pressure suppression	1. Suppression pool (Mark I) a. Safety/relief valves b. Downcomers c. Vacuum breakers 2. RHR sprays in the drywell and wetwell	1. Suppression pool (Mark III) a. Safety/relief valves b. Horizontal vents c. Vacuum relief valves d. Pool makeup system 2. RHR sprays in primary containment
Containment heat removal	1. RHR systems, related to a. Shutdown cooling mode b. LPCI mode c. Pool cooling mode 2. High pressure service water system	1. RHR systems, related to a. Shutdown cooling mode b. LPCI mode c. Pool cooling mode d. Steam condensing mode (with RCIC) 2. High pressure service water system
Containment integrity	1. Containment building and isolation system 2. Secondary containment building systems serving the function of containment 3. Pressure suppression or containment heat removal 4. Containment inerting system	1. Containment building and isolation system 2. Systems serving the function of containment pressure suppression or containment heat removal 3. Hydrogen control system
Containment radioactivity removal	1. Suppression pool 2. Drywell and wetwell spray 3. SGTS in the reactor building	1. Suppression pool 2. Containment spray
Monitoring reactor or containment status	1. Monitors and associated components for reactor a. Reactor water level b. Steamline and feedwater temperatures	1. Monitors and associated components for reactor a. Reactor water level b. Steam line and feedwater temperatures

Step 2

Table 3 (continued)
A LIST OF SYSTEMS BY FUNCTION
FOR TWO BOILING WATER REACTORS (PEACH BOTTOM AND GRAND GULF)

FUNCTION	PEACH BOTTOM	GRAND GULF
	c. Reactor pressure	c. Reactor pressure
	d. ADS and SRV position	d. ADS and SRV position
	2. Monitors and associated components for containment	2. Monitors and associated components for containment
	a. Temperature (drywell and wetwell)	a. Temperature (drywell and wetwell)
	b. Pressure	b. Pressure
	c. Radiation	c. Radiation
	d. Isolation valves position	d. Isolation valves position
	e. Inerting	e. Hydrogen concentration
Miscellaneous support	1. Service water system	1. Service water system
	a. High pressure service water system	a. Standby service water systems
	b. Emergency service water system	b. HPCS service water system
	2. Electrical Power	2. Electrical power
	a. Offsite AC	a. Offsite AC
	b. Emergency diesels	b. Emergency diesels
	c. DC power	c. DC power
	d. Interconnections and shared systems	d. Interconnections and shared systems
	3. Actuation, control and instrumentation systems for	3. Actuation, control and instrumentation systems for
	a. Reactor protection system	a. Reactor protection system
	b. ESFs	b. ESFs
	4. Compressed air systems	4. Compressed air systems
	5. Fire protection systems	5. Fire protection systems
	6. Sampling systems	6. Sampling systems
	7. Systems related to control room	7. Systems related to control room
	a. Shielding	a. Shielding
	b. HVAC	b. HVAC
	c. Computer system	c. Computer system
	d. Communication systems	d. Communication systems
	e. Emergency control room	e. Emergency control room

Step 2

Table 4
TYPICAL FRONT-LINE SYSTEM* FOR PWRS

Initiating Event	Function	Front-Line System
LOCA	Render Reactor Subcritical	Reactor Protection System
	Remove Core Decay Heat/Provide Core Makeup	High Pressure Injection System Low Pressure Injection System High Pressure Recirculation System Low Pressure Recirculation System Core Flood Tanks Auxiliary Feedwater System Power Conversion System
	Prevent Containment Overpressure	Reactor Building Spray Injection System Reactor Building Spray Recirculation System Reactor Building Fan Coolers Ice Condensers
	Scrub Radioactive Materials	Reactor Building Spray Injection System Reactor Building Spray Recirculation System Ice Condensers
Transients	Render Reactor Subcritical	Reactor Protection System Chemical Volume and Control High Pressure Injection System
	Remove Core Decay Heat/Provide Core Makeup	Auxiliary Feedwater System Power Conversion System High Pressure Injection System Power-Operated Relief Valves
	Prevent Containment Overpressure	Containment Spray Injection System Containment Spray Recirculation System Containment Fan Cooling System Ice Condenser
	Scrub Radioactive Materials	Containment Spray Injection System Containment Spray Recirculation System Ice Condenser

*From Reference 19.

Table 5
TYPICAL FRONT-LINE SYSTEM* FOR BWRS

Initiating Event	Function	Front-Line System
LOCA	Render Reactor Subcritical	Reactor Protection System
	Remove Core Decay Heat/Provide Core Makeup	Main Feedwater System Low Pressure Coolant Injection (Residual Heat Removal System) Low Pressure Core Spray System Automatic Pressure Relief System High Pressure Coolant Injection System Reactor Core Isolation Cooling System
	Prevent Containment Overpressure	Suppression Pool Residual Heat Removal System Containment Spray System
	Scrub Radioactive Materials	Suppression Pool Containment Spray System
Transients	Render Reactor Subcritical	Reactor Protection System Standby Liquid Control System
	Remove Core Decay Heat/Provide Core Makeup	Power Conversion System High Pressure Core Spray System High Pressure Coolant Injection System Low Pressure Core Spray System Low Pressure Coolant Injection (Residual Heat Removal System) Reactor Core Isolation Cooling System Feedwater Coolant Injection Standby Coolant Supply System Isolation Condensers Control Rod Drive (cooling water) System Condensate Pumps
	Prevent Reactor Coolant System Overpressure	Safety Relief Valves Power Conversion System Isolation Condenser
	Prevent Containment Overpressure	Residual Heat Removal System Shutdown Cooling System Containment Spray System
	Scrub Radioactive Materials	Suppression Pool Containment Spray System

*From Reference 19.

Table 6
IMPORTANT SYSTEMS IDENTIFIED IN THE IDCOR PROGRAM

	SEQUOYAH	ZION	PEACH BOTTOM	GRAND GULF
Systems failure important to core damage frequency	HPR LPR	LPR HPI/HPR AFW	PCS HPSWS ESWS RPS	RHR-SPC PCS
System failure important to public risk	HPR LPR	HPR/LPR AFW	PCS HPSWS RPS	RHR-SPC PCS RPS
Additional important systems	HMS	CFCS CS**	CS* CRD ADS/SRV	HPCS LPCS RCIC ADS/SRV

*Core spray.

**Containment spray, diesel-driven pump.

A list of acronyms is given in the following page.

Step 3

LIST OF ACRONYMS FOR TABLE 6

ADS	- Automatic Depressurization System
AFW	- Auxiliary Feedwater
CFCS	- Containment Fan Cooler System
CRD	- Control Rod Drive
CS	- Containment Spray
CS	- Core Spray (BWR only)
ESW	- Emergency Service Water
HMS	- Hydrogen Mitigation System
HPCI	- High Pressure Coolant Injection System
HPCS	- High Pressure Core Spray
HPI	- High Pressure Injection
HPR	- High Pressure Recirculation
HPSWS	- High Pressure Service Water System
LPCS	- Low Pressure Core Spray
LPR	- Low Pressure Recirculation
PCS	- Power Conversion System
RCIC	- Reactor Core Isolation Cooling
RHR-SPC	- Residual Heat Removal-Suppression Pool Cooling
RPS	- Reactor Protection System
SRV	- Safety Relief Valve

Step 3

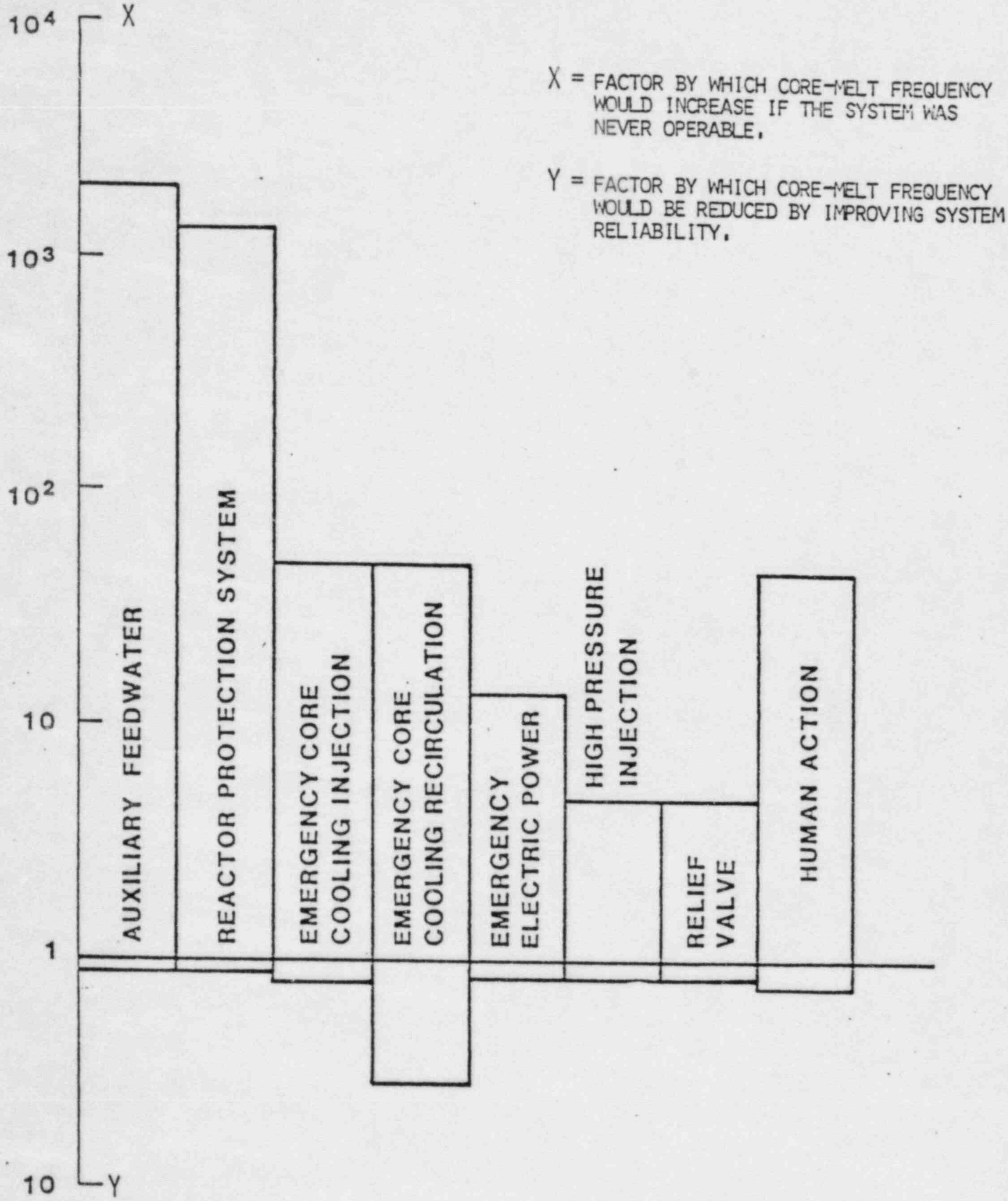

Figure 1. Risk Worth Ratios for Sequoyah Safety Systems with Regard to Core-Melt Frequency

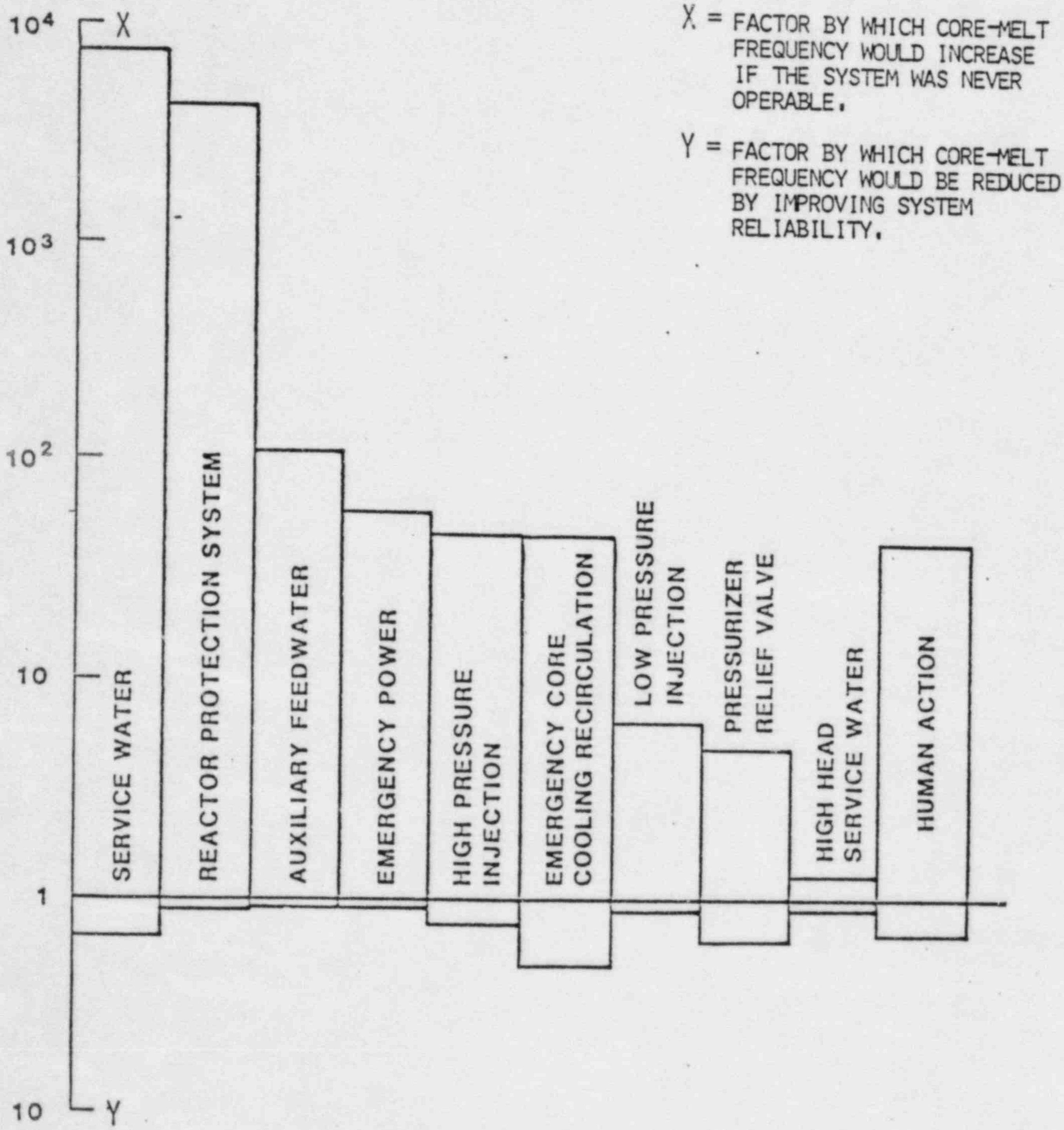

Figure 2. Risk Worth Ratios for Oconee Safety Systems with Regard to Core Melt-Frequency

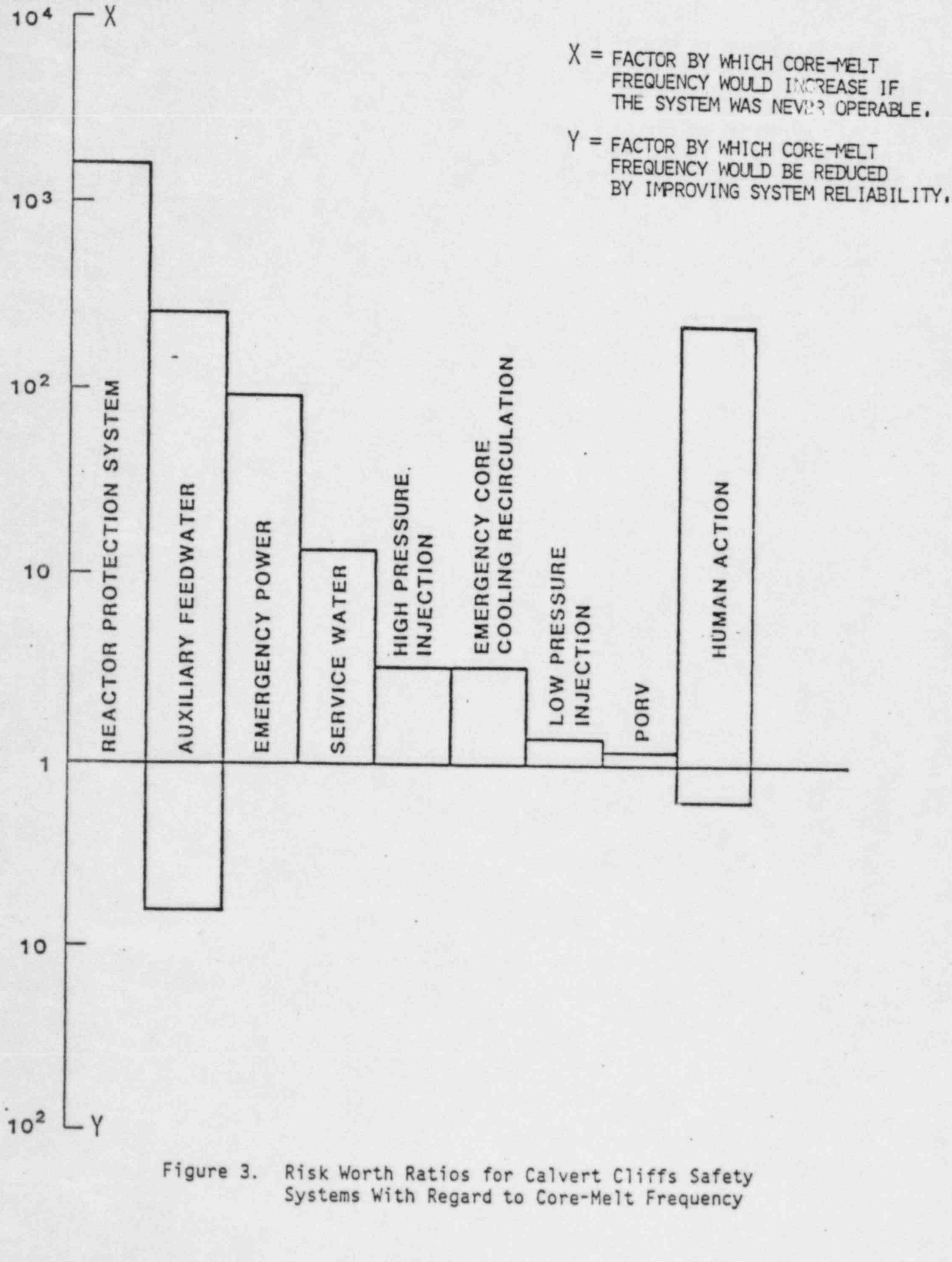

Figure 3. Risk Worth Ratios for Calvert Cliffs Safety Systems With Regard to Core-Melt Frequency

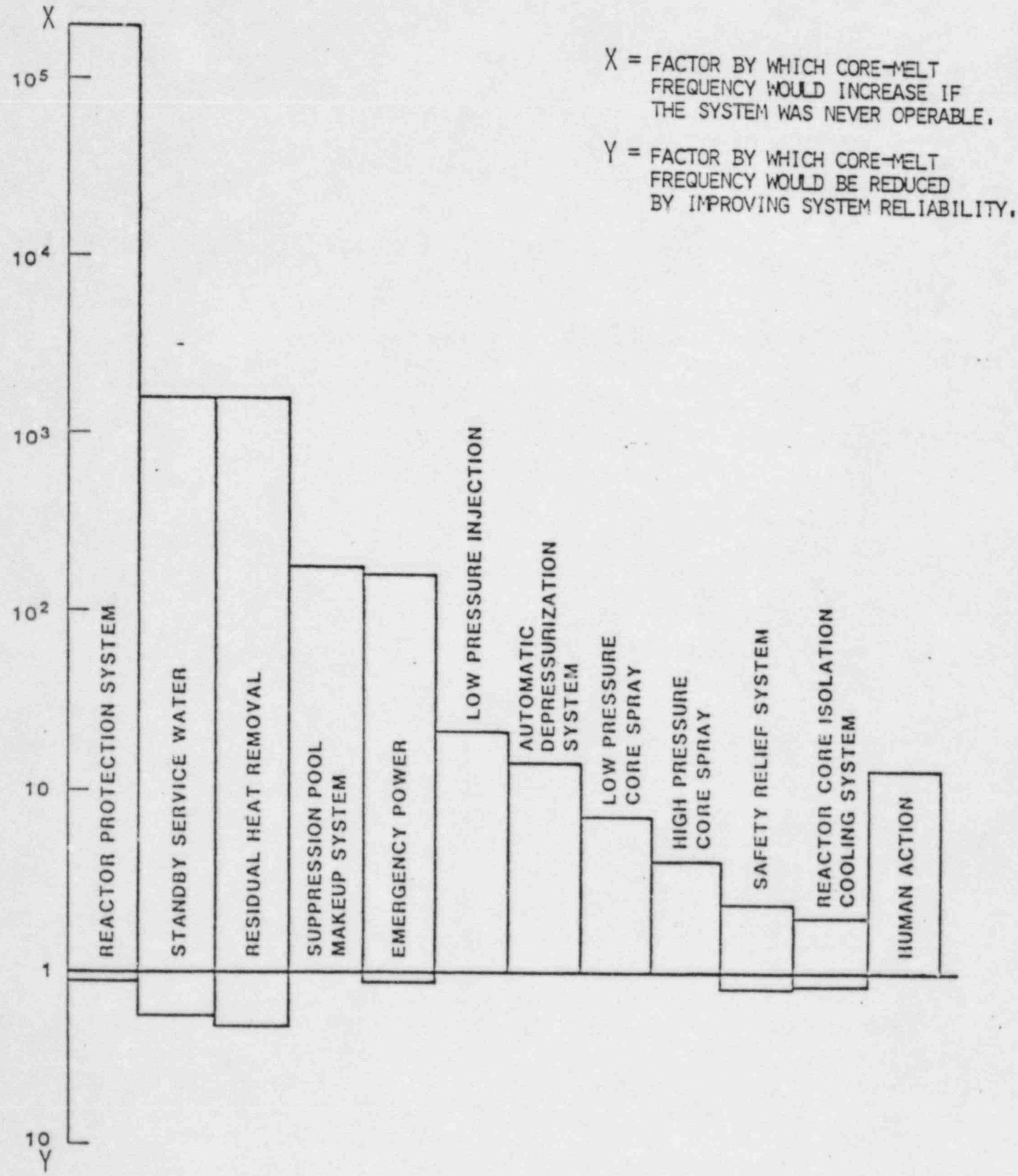

Figure 4. Risk Worth Ratios for Grand Gulf Safety Systems with Regard to Core-Melt Frequency

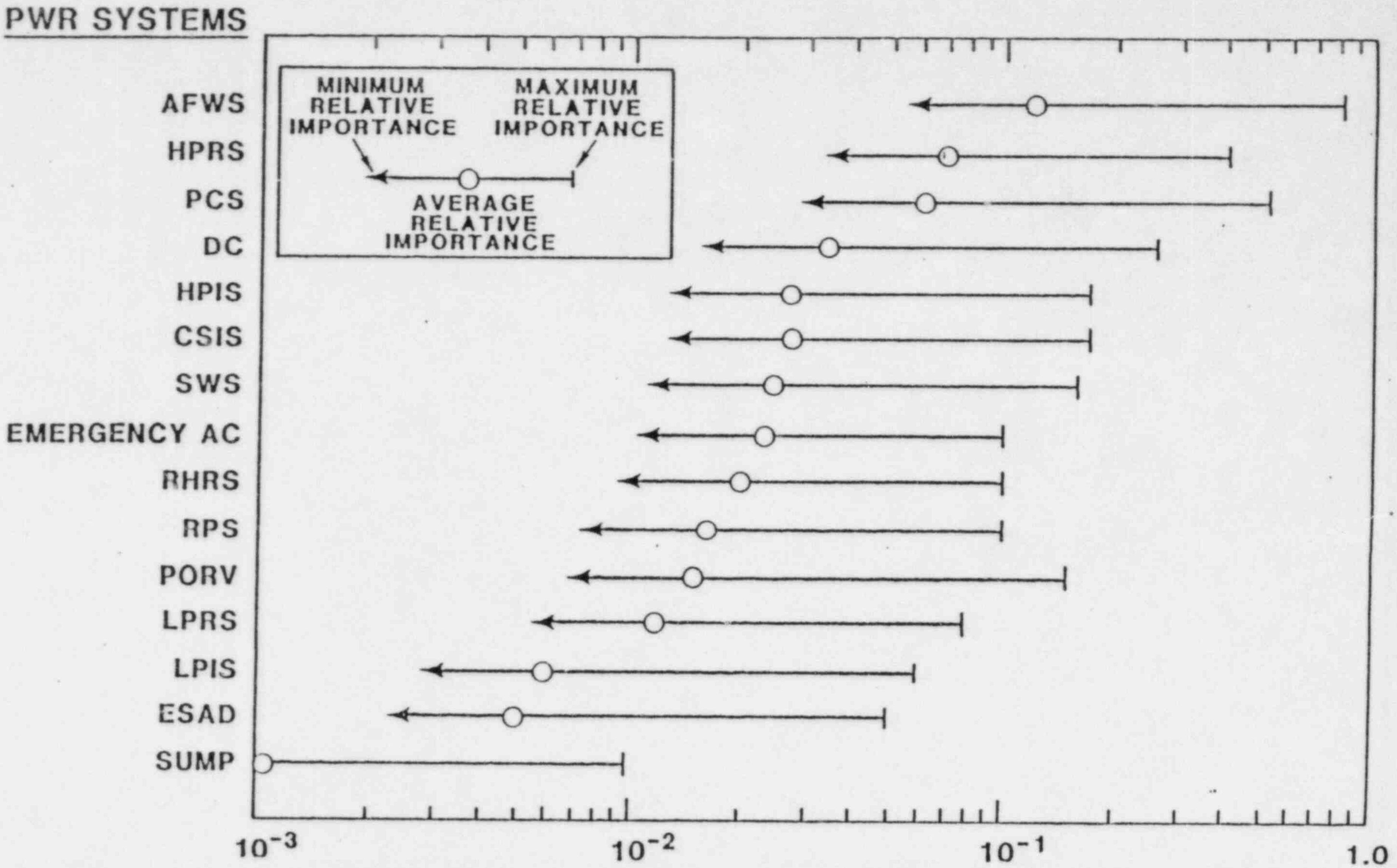

Figure 5. Relative Importance of PWR Systems Considering Dominant Accident Sequences from 15 PRAs

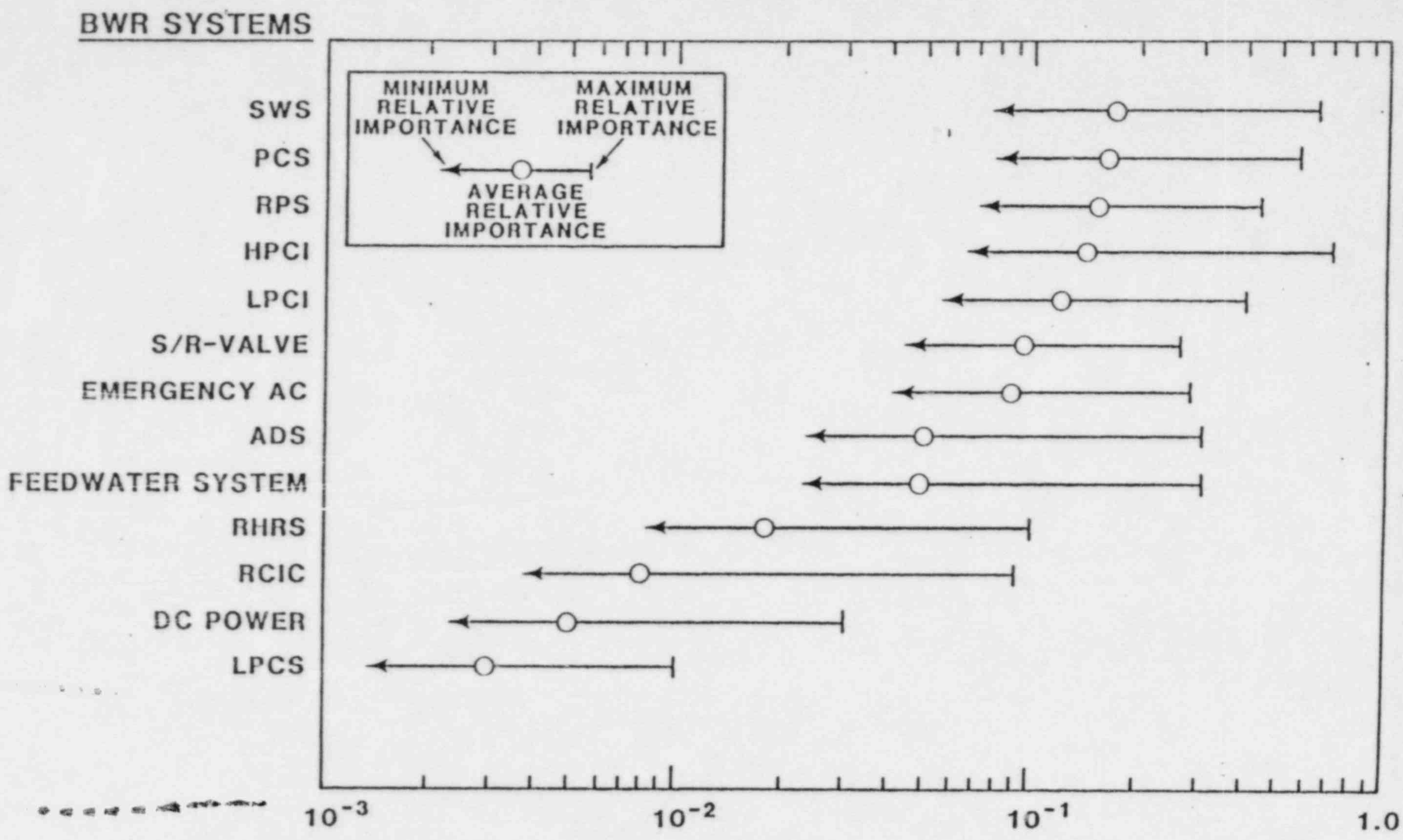

Figure 6. Relative Importance of BWR Systems Considering Dominant Accident Sequences from 15 PRAs

Table 7
DOMINANT* FUNCTIONAL ACCIDENT SEQUENCES FOR PWRS
FROM THE ASEP** STUDY

Sequence Description	Freq. Range	Major Uncertainties	Comment
1) Transient Loss of reactor subcriticality	6E-5+ 1E-6	RPS reliability RCS ability to withstand pressure spike	ATWS rule pending
2) Transient Loss of integrity Loss of core cooling	3E-5 <1E-6	PORV demand rate HPIS availability Necessity to switch-over to recirc.	TMI fixes (raising PORV Set point and anticipatory AFWS start signal) should reduce sequence freq.
3) Transient Loss of core cooling	1E-3 1E-7	Feed and bleed capability AFWS availability	TMI fixes have called for many improvements in AFWS availability
4) Transient Loss of core cooling Loss of containment heat removal	<2E-4 2E-7	Redundancy of AC Power sources Battery, CST depletion times, possibility of induced RCS pump seal leak long term ventilation loss effects AFWS availability	NRC position statement-forthcoming

*Dominating the core-melt frequency.
**ASEP: Accident Sequence Evaluation Program sponsored by the office of Nuclear Regulatory Research.
+6E-5: 6×10^{-5}

Table 7 (continued)
DOMINANT* FUNCTIONAL ACCIDENT SEQUENCES FOR PWRS
FROM THE ASEP** STUDY

Sequence Description	Freq. Range	Major Uncertainties	Comment
5) LOCA Loss of core cooling	2E-4 <4E-7	LOCA frequency ECCS success criteria ECCS redundancy	Small LOCA may be higher than thought due to RCP seal leaks TMI fixes stressed better procedures for small LOCA
6) LOCA Loss of core cooling Loss of containment heat removal	6E-6 <1E-6	LOCA frequency ECCS success criteria	Small LOCA may be higher thought due to RCP ECCS redundancy seal leaks TMI fixes stressed better procedures for small LOCA
7) Event V, interfacing LOCA	10E-7 10E-9	Valve rupture and spurious actuation Test and surveillance	

*Dominating the core-melt frequency.
**ASEP: Accident Sequence Evaluation Program sponsored by the office of Nuclear Regulatory Research.

Table 8
DOMINANT* FUNCTIONAL ACCIDENT SEQUENCES FOR
BWRS FROM THE ASEP** STUDY

	Sequence Description	Freq. Range	Major Uncertainties	Comment
1)	Transient Loss of reactor subcriticality	5E-5+ 1E-7	RPS reliability Adequacy of ECCS Unknown phenomenology in RCS, ability of operator to control water level	ATWS rule pending
2)	Transient Loss of RCS integrity Loss of core cooling	7E-5 <2E-7	ECCS availability Emergency procedures for ADS SRV demand rate	
3)	Transient Loss of RCS integrity Loss of containment cooling	1E-3 1E-7	RHR availability SRV demand rate	Estimated time to core melt appear longer than previously expected, thus longer times for recovery

*Dominating the core-melt frequency.
**ASEP: Accident Sequence Evaluation Program sponsored by the office of Nuclear Regulatory Research.
+5E-5: 5×10^{-5}

Table 8 (continued)
DOMINANT* FUNCTIONAL ACCIDENT SEQUENCES FOR
BWRS FROM THE ASEP** STUDY

	Sequence Description	Freq. Range	Major Uncertainties	Comment
4)	Transient Loss of core cooling	7E-4 2E-7	ECCS availability Emergency procedure for ADS	Station blackout rules pending
5)	Transients Loss of containment cooling	1E-4 <4E-7	RHR availability ECCS success criteria ECCS redundancy	Estimated time to core melt appear longer than previously expected, thus longer times for recovery
6)	LOCA Loss of containment cooling	5E-6 <1E-7	RHR availability Time available for recovery	Estimated time to core melt appear longer than previously expected, thus longer times for recovery

*Dominating the core-melt frequency.
**ASEP: Accident Sequence Evaluation Program sponsored by the office of Nuclear Regulatory Research.

Table 9

COMPLETED FULL-SCOPE PROBABILISTIC RISK ASSESSMENTS

Plant	Issuance	Operating License	Rating (MWe)	NSSS/AE[1]	Containment	Sponsor	Report
Surry 1	1975	1972	788	W/S&W	Dry-Cylinder	NRC	NUREG-75/014 (WASH-1400)
Peach Bottom 2	1975	1973	1065	GE/Bechtel	Mark I	NRC	NUREG-75/014 (WASH-1400)
Big Rock Point	1981	1962	71	GE/Bechtel	Dry-Sphere	Utility	USNRC Docket 55-155
Zion 1 & 2	1981	1973	1040	W/S&L	Dry-Cylinder	Utility	USNRC Docket 50-295
Indian Pt. 2 & 3	1982	1973	873	W/UE&C	Dry-Cylinder	Utility	USNRC Dockets 50-247 and 50-286
Yankee Rowe	1982	1960	175	W/S&W	Dry-Sphere	Utility	USNRC Docket 50-29
Limerick 1 & 2	1983	(1985)	1055	GE/Bechtel	Mark II	Utility	USNRC Docket 50-352
Shoreham	1983	(1984)	819	GE/S&W	Mark II	Utility	USNRC Dockets 50-322 and 50-353
Millstone 3*	1983	(1986)	1150	W/S&W	Dry-Cylinder	Utility	Controlled document
Susquehanna 1*	1983	1983	1050	GE/Bechtel	Mark II	Utility	Draft
Oconee 3*	1983	1973	860	B&W/Duke	Dry-Cylinder	EPRI/NSAC	Draft

*Completed but not yet publicly available.
[1]NSSS--Nuclear Steam System Supplier; AE--Architect-Engineer.

Table 10

COMPLETED LIMITED-SCOPE PROBABILISTIC RISK ASSESSMENTS

Plant	Issuance	Operating License	Rating (MWe)	NSSS/AE[1]	Containment	Sponsor (program)	Report
Oconee 3	1981	1973	860	B&W/Duke	Dry-Cylinder	NRC (RSSMAP)	NUREG/CR-1659
Sequoyah 1	1981	1981	1148	W/TVA	Ice Condenser	NRC (RSSMAP)	NUREG/CR-1659
Grand Gulf 1	1981	1982	1250	GE/Bechtel	Mark III	NRC (RSSMAP)	NUREG/CR-1659
Calvert Cliffs 1	1981	1974	845	CE/Bechtel	Dry-Cylinder	NRC (RSSMAP)	NUREG/CR-1659
Crystal River 3	1982	1976	797	B&W/Gilbert	Dry-Cylinder	NRC (IREP)	NUREG/CR-2515
Browns Ferry 1	1982	1973	1065	GE/TVA	Mark I	NRC (IREP)	NUREG/CR-2802
Arkansas 1	1982	1974	836	B&W/Bechtel	Dry-Cylinder	NRC (IREP)	NUREG/CR-2787
Millstone 1	1983	1970	652	GE/EBASCO	Mark I	NRC (IREP)	NUREG/CR-3085
Calvert Cliffs 2	1983	1974	845	CE/Bechtel	Dry-Cylinder	NRC (IREP)	Draft
Others*							

[1]NSSS--Nuclear Steam System Supplier; AE--Architect-Engineer.

*The German risk study, the Sizewell study and the Ringhals-2 study.

Table 11

A COMPARISON OF DOMINANT (CORE-MELT FREQUENCY) ACCIDENT SEQUENCES†

Sequence Category	% Total Core-Melt Frequency										
	BRP	Zion	Limerick	Grand Gulf	ANO	Surry	Peach Bottom	Sequoyah	Oconee	IP-2	IP-3
Small LOCAs-Injection Failure	10	0	0	0	28	27	0	18	14	37	33
Small LOCAs-LTDHR* Failure	4	41	0	14	5	20	1	67	21	3	43
Large LOCAs-Injection Failure	0	3	0	0	0	4	1	0	0	0	1
Large LOCAs-LTDHR Failure	0	18	0	0	0	2	0	1		4	11
Transients-PCS* Not Available											
a. Loss of Off-site Power	14	18	48	27	20	7	0	0	12	26	3
b. Injection failure	36	0	34	0	23	14	2	5	15	28	2
c. LTDHR failure	5	0	0	38	0	0	0	0	21	0	0
Transients-PCS Available											
a. Injection failure	0	0	5	0	0	0	0	0	1	0	0
b. LTDHR failure	0	4	3	0	0	0	47	0	0	0	0
ATWS	0	15	2	14	4	9	47	0	11	0	1
Interfacing LOCA	9	0	0	0	0	9	0	9	5	0	0
TOTALS	78%	99%	92%	93%	80%	92%	98%	100%	100%	98%	94%

*LTDHR is long term decay heat removal which includes recirculation and RHR. PCS is power conversion system.

†From Reference 19

BRP - Big Rock Point

ANO - Arkansas Nuclear One Unit One

IP-2 - Indian Point Unit No. 2

IP-3 - Indian Point Unit No. 3

Step 4

Table 12
A BRIEF DESCRIPTION OF FIVE SELECTED DOMINANT ACCIDENT SEQUENCES
FOR ZION IN THE IDCOR PROGRAM

Sequence No. 1*. A small cold leg LOCA (2") with failure of ECCS recirculation. One containment spray train is operational through the injection phase. Fan cooler operation (3 units) continues throughout the accident. Reactor scram occurs on low pressurizer pressure and the reactor coolant pumps are tripped by operator action 10 minutes after the event initiation. Activation of main feedwater trip, main steam isolation valve closure, and auxiliary feedwater actuation are automatic with ECCS actuation. ECCS operation consists of 2/2 SI pumps and 1/2 charging/SI pumps during the injection phase.

Sequence No. 2. A seismic induced loss of all AC and main DC power with concurrent failure of the turbine driven auxiliary feedwater train. Recovery potential is deemed negligible since the diesel generators, off-site power insulators, battery racks, control room, auxiliary electric equipment room, and turbine driven auxiliary feedwater pump are all involved in either direct structural failure or failure of the auxiliary building structure. Plant operators are judged to be in an extremely high stress situation with essentially no availability. A reactor coolant pump seal LOCA (200 gpm) is conservatively assumed 45 minutes into the event. No electrically powered ignition sources are available in the containment for hydrogen ignition.

Sequence No. 3. A large LOCA (28 1/2" double ended cold leg break) with failure of ECCS recirculation. One containment spray train is operational through the injection phase. Fan cooler operation (3 units) continues throughout the accident. Reactor scram ocurs on low pressurizer pressure and the reactor coolant pumps are tripped immediately after scram. Actuation of main feedwater trip, main steam isolation valve closure, and auxiliary feedwater actuation are automatic with ECCS actuation. ECCS operation during the injection phase consists of 3 accumulators, 2 RHR pumps, 2 SI pumps and one charging pump with all pumps but the RHR pumps being manually shut off essentially at the start of the event.

Sequence No. 4. A spurious safety injection, with SI terminated when the pressurizer goes solid. At this point, a PORV sticks open with subsequent failure of recirculation cooling (2 RHR pumps). Auxiliary feedwater, one containment spray train, and 3 RCFC units operate. The containment spray train is operational through the injection phase. (NOTE: The best information currently available indicates that the charging pumps will not lift a PORV. This sequence would then be realizable only given extraordinary human error and is far less likely than portrayed in the Zion Probabilistic Safety Study.)

Sequence No. 5. A large LOCA with failure of RHR injection. The LOCA is a 28 1/2 inch double ended cold leg break and RHR failure constitutes a failure of 2 RHR pumps. Two SI pumps and one charging pump are operational during the the injection phase as are 3 accumulators. One containment spray train is operational through the injection phase. Fan cooler operation (3 units) continues throughout the accident. Scram, reactor coolant pump trip and other important actuations are as described for Sequence 3 above.

*Sequence number does not correspond to the order of dominance.

Table 13
A BRIEF DESCRIPTION OF FIVE SELECTED DOMINANT ACCIDENT SEQUENCES
FOR SEQUOYAH IN THE IDCOR PROGRAM

Sequence 1. This sequence is initiated by a small loss-of-coolant accident (an equivalent diameter of about 1/2 to 2 inches). The upper level injection, charging pumps, safety injection pumps, RHR pumps, cold leg accumulators, auxiliary feedwater, air return fans, containment sprays and ignitors are all operational. However, the safety-injection pumps become inoperable because of failure to switch to the recirculation mode when the refueling water storage tank empties.

Sequence 2. This sequence is initiated by a small loss-of-coolant accident followed by failure of the emergency core coolant injection system to operate. This system continues to be unavailable in the recirculation mode. The cold leg accumulators, auxiliary feedwater, air return fans, containment sprays and ignitors are operational throughout the accident.

Sequence 3. This sequence is initiated by a small LOCA followed by failure of both the safety-injection system and the containment spray system to switch to the recirculation mode. The failure of these systems is dominated by a common mode failure because of either closed or plugged drains between the upper and lower containment compartments. All other safety engineered systems are operational.

Sequence 4. This sequence is initiated by a loss of off-site AC power with subsequent loss of onsite AC power. Due to lack of cooling, the reactor coolant pump seals fail at 2700 seconds resulting in a small LOCA. Both main and auxiliary feedwater are assumed lost at the time of the initiating event. Emergency core cooling, containment sprays, air return fans, and hydrogen ignitors are not available due to the loss of all AC power.

Sequence 5. This sequence has a transient initiator other than loss of offsite power with automatic reactor trip. It assumes that both normal and emergency means of adding water to the steam generators fail and the steam generators boil dry. The pressure relief valves open resulting in a loss of coolant which leads to core melt because the injection system cannot keep up with the loss of coolant. AC power is available and therefore emergency core cooling and containment safeguards are available; however, no operator action is assumed in this case. Therefore, emergency core cooling is not effective.

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*Sequence number does not correspond to the order of dominance.

Table 14
A BRIEF DESCRIPTION OF FIVE SELECTED DOMINANT ACCIDENT SEQUENCES FOR PEACH BOTTOM IN THE IDCOR PROGRAM

Sequence No. 1*. The MSIV closure is followed by a scram of the reactor. The turbine driven high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps add water to the reactor core. Steam produced in the reactor core is relieved through the relief valves into the suppression pool. The pool heats up because of steam quenching with the absence of RHR pool cooling. When the pool reaches 120°F it is assumed that the operator depressurizes the reactor by opening three safety relief valves (SRV). Low pressure core spray (LPCS) and low pressure core injection (LPCI) are then available to supply makeup water to the core until insufficient net positive suction head (NPSH) causes the pumps to cavitate. Following the loss of all make up water to the core, the core boils dry. Core degradation and core melt soon follow.

Sequence No. 2. The reactor protection system fails to scram concurrent with the MSIV closure, and stand-by liquid control (SLC) is not activated. The recirculation pump trip is successful and the decreased flow through the core causes an increase in the average core void fraction thus decreasing the power. Given the successful activation of high pressure injection, the power stabilizes at an average of 20% of rated power.

It is assumed for this sequence that the primary system remains pressurized and that the high pressure coolant injection systems (HPCI and RCIC) continue to operate until the suppression pool reaches 170°F. At this point the high pressure pumps fail due to high bearing temperature, and the core water level drops. If automatic depressurization occurs the low pressure cooling systems (LPCI and LPCS) are available after depressurization until the reduction of NPSH causes the pumps to cavitate. Suppression pool cooling is activated 10 minutes into the event and is maintained until the pumps cavitate. After the core uncovers, core degradation and core melt begin.

Sequence No. 3. As in Sequence No. 1 the MSIV closure is followed by a scram. However, for this sequence both high pressure and low pressure ECCS fail. CRD flow is the only available core make-up system; however, at the normal flow rate it is not expected to be sufficient to keep the core covered. The core boils dry and the steam is vented to the suppression pool. The pool cooling mode of the RHR is unavailable, and therefore, pool heat-up occurs due to steam quenching. After the core boils dry, degradation and core melt occur.

Sequence No. 4. For this sequence a small break accident occurs with MSIV closure. The small break accident is represented by a stuck open relief valve. The reactor is able to scram, but all core injection systems are assumed to fail. CRD flow is the only core make-up available. The steam produced in the core is quenched in the suppression pool. Suppression pool cooling is available, and no significant pressurization of the containment prior to reactor vessel failure is therefore expected. It is assumed that the operator leaves the reactor pressurized. No automatic depressurization occurs because the high drywell pressure permissive signal is absent.

*Sequence number does not correspond to the order of dominance.

Sequence No. 5. A loss of all off-site and on-site AC power (station blackout) occurs. For this sequence only the HPCI and RCIC turbine driven pumps are available to supply make-up water to the reactor. Since the RHR system is unavailable for suppression pool cooling due to loss of AC power, the suppression pool will heat up with the venting of steam through the steam relief lines. When the pool reaches 170°F the HPCI and RCIC pumps will fail due to high bearing temperature, and then the core proceeds to boil dry. With the loss of off-site AC power, the CRD flow is also lost. The automatic depressurization system is not activated because of the lack of a permissive signal for the LPCI. As a result the primary system will not be depressurized at the time of core overheating and migration of core material into the lower plenum.

Table 15
A BRIEF DESCRIPTION OF FIVE SELECTED DOMINANT ACCIDENT SEQUENCES FOR GRAND GULF IN THE IDCOR PROGRAM

Accident Sequence No. 1*. The accident is assumed to be initiated by the inadvertent closure of all MSIVs. A reactor scram is assumed to occur immediately after the initiating event. The MSIV closures cause a Reactor Vessel pressure excursion, which opens the safety/relief valves, routing steam to the suppression pool. One of the SRVs is assumed to stick open. During the accident, both normal feedwater and main condenser are assumed to be unavailable. All other plant systems are assumed to be available during the accident. No credit is taken for any operator action. Containment failure is caused by steam overpressure.

Accident Sequence No. 2. The accident is assumed to be initiated by the inadvertent closure of all MSIVs during full-power operation. Both normal feedwater and main condenser are assumed lost at the initiation of the event and not restored during the event. It is further assumed that neither the reactor protection system nor the operator successfully insert either the control rods or inject soluble poison into the core during the accident. All other plant systems are assumed to be available during the accident. The only operator action credited during the event is the manual alignment and actuation of the RHR system in its SP cooling mode. Containment failure is caused by steam overpressure.

Accident Sequence No. 3. The accident is assumed to be initiated by the inadvertent closure of all MSIVs during full-power operation. During the accident, both normal feedwater and main condenser are assumed to be unavailable. The high-pressure makeup systems (HPSC and RCIC) are also assumed unavailable and thus unable to supply high-pressure makeup to the reactor vessel. In addition, the RHR heat exchangers are assumed to be unavailable for cooling either the primary system or the containment. No credit is taken for any operator actions. Containment failure is caused by steam overpressure.

Accident Sequence No. 4. The accident sequence is assumed to be initiated by the inadvertent closure of all MSIVs during full-power operation. A reactor scram is assumed to occur immediately after the initiating event. During the accident, both normal feedwater and main condenser are assumed to be unavailable. The RHR heat exchangers are also assumed to be unavailable for cooling either the primary system or the containment. All other plant systems are assumed to be available during the accident. No credit is taken for any operator action. Containment failure is caused by steam overpressure.

Accident Sequence No. 5. The accident is assumed to be initiated by a loss of offsite power event. A reactor scram is assumed to occur immediately after the initiating event. During the accident, all systems powered from the plant normal AC power busses are assumed to be unavailable. Among the most significant of these systems are the main feedwater and the main condenser systems.

*Sequence number does not correspond to the order of dominance.

Also unavailable, due to the initiating event, is the CRD cooling flow into the vessel. The accident sequence also specifies that neither the high-pressure (HPCS and RCIC) nor the low-pressure (LPCS and LPCI) makeup systems are available at any time during the accident. The faults in these makeup systems are taken to be such that the systems are unavailable in any mode of operation. Thus, for this event, no water makeup or cooling to the RV or containment occurs, including RHR cooling and containment spraying. All other plant systems are assumed to be available. No credit is taken for any operator action. Containment failure occurs as a result of a hydrogen burn.

Table 16

BASELINE FREQUENCIES FOR PWR TRANSIENT INITIATORS (NUREG-1050)

Init.	PWR Transient Categories	Mean	Variance	Median
1	Loss of RCS Flow (1 Loop)	4.4 E-1	1.3 E-1	3.2 E-1
2	Uncontrolled Rod Withdrawal	2.0 E-2	3.2 E-4	1.3 E-2
3	CRDM Problems and/or Rod Drop	6.1 E-1	3.1 E-1	4.2 E-1
4	Leakage from Control Rods	2.3 E-2	5.0 E-4	1.6 E-2
5	Leakage in Primary System	1.1 E-1	1.1 E-2	7.3 E-2
6	Low Pressurizer Pressure	3.1 E-2	6.5 E-4	2.3 E-2
7	Pressurizer Leakage	9.6 E-3	1.5 E-4	6.0 E-3
8	High Pressurizer Pressure	2.8 E-2	5.5 E-4	2.0 E-3
9	Inadvertent Safety Injection Signal	5.4 E-2	2.3 E-3	4.0 E-2
10	Containment Pressure Problems	1.0 E-2	1.8 E-4	5.9 E-3
11	CVCS Malfunction-Boron Dilution	3.6 E-2	8.3 E-4	2.7 E-2
12	Pressure/Temperature/Power Imbalance-Rod Position Error	1.5 E-1	2.2 E-2	1.0 E-1
13	Startup of Inactive Coolant Pump	4.8 E-3	5.7 E-4	2.3 E-3
14	Total Loss of RCS Flow	2.8 E-2	5.4 E-4	2.0 E-2
15	Loss or Reduction in Feedwater Flow (1 Loop)	1.8 E+0	9.2 E-1	1.5 E+0
16	Total Loss of Feedwater (All Loops)	1.8 E-1	3.0 E-2	1.1 E-1
17	Full or Partial Closure of MSIV (1 Loop)	2.3 E-1	4.8 E-2	1.5 E-1
18	Closure of All MSIV	3.0 E-2	6.6 E-4	2.1 E-2
19	Increase in Feedwater Flow (1 Loop)	6.4 E-1	3.3 E-1	4.4 E-1
20	Increase in Feedwater Flow (All Loops)	1.6 E-2	3.0 E-4	1.0 E-2
21	Feedwater Flow Instability - Operator Error	1.8 E-1	3.2 E-2	1.1 E-1
22	Feedwater Flow Instability - Mechanical Cause	2.0 E-1	4.0 E-2	1.3 E-1
23	Loss of Condensate Pumps (1 Loop)	1.0 E-1	9.8 E-3	6.8 E-2
24	Loss of Condensate Pumps (All Loops)	4.8 E-3	5.7 E-4	2.3 E-3
25	Loss of Condenser Vacuum	2.3 E-1	4.2 E-2	1.7 E-1
26	Steam Generator Leakage	3.7 E-2	8.0 E-4	2.7 E-2
27	Condenser Leakage	5.3 E-2	2.6 E-3	3.8 E-2
28	Miscellaneous Leakage in Secondary System	8.8 E-2	5.9 E-3	6.4 E-2
29	Sudden Opening of Steam Relief Valves	3.9 E-2	8.9 E-4	3.0 E-2
30	Loss of Circulating Water	6.3 E-2	2.7 E-3	4.7 E-2
31	Loss of Component Cooling	1.5 E-2	8.8 E-2	5.1 E-5
32	Loss of Service Water System	1.0 E-2	1.8 E-4	5.9 E-3

Table 16 (Continued)

BASELINE FREQUENCIES FOR PWR TRANSIENT INITIATORS (NUREG-1050)

Init.	PWR Transient Cate	Mean	Variance	Median
33	Turbine Trip, Throttle Valve Closure, EHC Problems	1.6 E+0	6.6 E-1	1.3 E+0
34	Generator Trip or Generator Caused Fault	4.1 E-1	8.3 E-2	3.2 E-1
35	Total Loss of Offsite Power	1.3 E-1	6.4 E-3	1.1 E-1
36	Pressurizer Spray Failure	3.8 E-2	7.8 E-4	2.9 E-2
37	Loss of Power Necessary to Plant Systems	1.1 E-1	1.1 E-2	7.5 E-2
38	Spurious Trips - Cause Unknown	1.3 E-1	1.4 E-2	9.5 E-2
39	Auto Trip - No Transient Condition	1.2 E+0	6.4 E-1	9.8 E-1
40	Manual Trip - No Transient Condition	5.8 E-1	3.0 E-1	3.9 E-1
41	Fire Within Plant	2.3 E-2	4.3 E-4	1.6 E-2

Table 17

BASELINE FREQUENCIES FOR BWR TRANSIENT INITIATORS (NUREG-1050)

Int.	BWR Transient Categories	Mean	Variance	Median
1	Electric Load Rejection	7.0 E-1	1.9 E-1	5.7 E-1
2	Electric Load Rejection with Turbine Bypass Valve Failure	1.1 E-2	4.7 E-4	5.2 E-3
3	Turbine Trip	1.2 E+0	5.9 E-1	9.2 E-1
4	Turbine Trip with Turbine Bypass Valve Failure	1.1 E-2	4.7 E-4	5.2 E-3
5	Main Stream Isolation Valve Closure	5.7 E-1	2.0 E-1	4.3 E-1
6	Inadvertent Closure of One MSIV (Rest Open)	2.1 E-1	3.4 E-2	1.5 E-1
7	Partial MSIV Closure	1.2 E-1	1.2 E-2	8.1 E-2
8	Loss of Normal Condenser Vacuum	4.8 E-1	1.0 E-1	3.9 E-1
9	Pressure Regulator Fails Open	1.8 E-1	2.7 E-2	1.2 E-1
10	Pressure Regulator Fails Closed	1.7 E-1	2.8 E-2	1.1 E-1
11	Inadvertent Opening of a Safety/Relief Valve (Stuck)	2.5 E-1	4.8 E-2	1.7 E-1
12	Turbine Bypass Fails Open	6.1 E-2	3.0 E-3	4.5 E-2
13	Turbine Bypass or Control Valves Cause Increase Pressure (Closed)	4.8 E-1	1.4 E-1	3.6 E-1
14	Recirculation Control Failure-Increasing Flow	2.5 E-1	4.7 E-2	1.8 E-1
15	Recirculation Control Failure-Decreasing Flow	1.3 E-1	1.4 E-2	8.4 E-2
16	Trip of One Recirculation Pump	8.8 E-2	6.2 E-3	6.5 E-2
17	Trip of All Recirculation Pumps	2.1 E-2	5.0 E-4	1.3 E-2
18	Abnormal Startup of Idle Recirculation Pump	1.4 E-2	8.0 E-2	7.2 E-5
19	Recirculation Pump Seizure	1.1 E-2	4.7 E-4	5.2 E-3
20	Feedwater-Increasing Flow at Power	1.8 E-1	2.6 E-2	1.2 E-1
21	Loss of Feedwater Heater	4.0 E-2	1.2 E-3	2.8 E-2
22	Loss of All Feedwater Flow	1.3 E-1	1.1 E-2	1.0 E-1
23	Trip of One Feedwater or Condensate Pump	1.7 E-1	2.4 E-2	1.2 E-1
24	Feedwater-Low Flow	5.8 E-1	1.7 E-1	4.5 E-1
25	Low Feedwater Flow During Startup or Shutdown	2.3 E-1	3.5 E-2	1.7 E-1
26	High Feedwater Flow During Startup or Shutdown	7.5 E-2	3.8 E-3	5.7 E-2
27	Rod Withdrawal at Power	2.1 E-2	5.2 E-4	1.3 E-2

Table 17 (Continued)

BASELINE FREQUENCIES FOR BWR TRANSIENT INITIATORS (NUREG-1050)

Int.	BWR Transient Categories	Mean	Variance	Median
28	High Flux Due to Rod Withdrawal at Startup	9.7 E-2	6.7 E-3	7.2 E-2
29	Inadv rtent Insertion of Rod or Rods	1.4 E-1	1.6 E-2	9.6 E-2
30	Detected Fault in Reactor Protection System	9.8 E-2	9.1 E-3	6.6 E-2
31	Loss of Offsite Power	1.2 E-1	6.0 E-3	9.2 E-2
32	Loss of Auxiliary Power (Loss of Auxiliary Transformer)	1.1 E-1	4.7 E-4	5.1 E-3
33	Inadvertent Startup of HPCI/HPCS	1.1 E-1	4.7 E-4	5.2 E-3
34	Scram Due to Plant Occurrences	4.7 E-1	1.7 E-1	3.3 E-1
35	Spurious Trip Via Instrumentation, RPS FAULT	1.3 E+0	6.1 E-1	1.1 E+0
36	Manual Scram - No Out-of-Tolerance Condition	8.1 E-1	4.4 E-1	5.9 E-1
37	Cause Unknown	1.4 E-1	1.9 E-2	9.3 E-2

Table 18
PWR CORE MELT PROBABILITIES (NUREG/CR-2906)

Sequences*	Brookhaven Re-evaluation (Point estimate)	Reactor Safety Study (Median Value)
LARGE LOCA (A)		
AB	1.00×10^{-9}	1.0×10^{-9}
AH	8.87×10^{-7}	1.3×10^{-6}
AF	8.02×10^{-9}	1.0×10^{-8}
AHF^C -	1.71×10^{-10}	2.3×10^{-10}
AD^C	1.89×10^{-6}	1.87×10^{-6}
AG	1.0×10^{-8}	8.5×10^{-9}
ADF	4.7×10^{-11}	1.87×10^{-10}
ACH	2.31×10^{-9}	3.12×10^{-9}
ACG	2.7×10^{-11}	2.04×10^{-11}
ACF	2.09×10^{-11}	2.4×10^{-11}
ACD^C	4.9×10^{-9}	4.5×10^{-9}
A Probabilities	2.8×10^{-6}	1.88×10^{-6}
SMALL LOCA (S_1)		
S_1	3.0×10^{-9}	3.0×10^{-9}
S_1H	3.52×10^{-6}	2.7×10^{-6}
S_1G	3.11×10^{-8}	2.55×10^{-8}
S_1F	2.41×10^{-8}	3×10^{-8}
S_1HF	5.83×10^{-10}	5.7×10^{-10}
S_1D	2.29×10^{-6}	2.85×10^{-6}
S_1DF	1.84×10^{-10}	2.85×10^{-10}
S_1CH	9.17×10^{-9}	6.48×10^{-9}
S_1CD	5.97×10^{-9}	6.84×10^{-9}
S_1K	1.93×10^{-8}	1.08×10^{-8}
S_1KC	5.02×10^{-11}	2.59×10^{-11}
S_1 Probabilities	5.9×10^{-6}	5.63×10^{-6}

*Sequence symbol key follows the table.

Table 18 (continued)

Sequences	Brookhaven Re-evaluation (Point estimate)	Reactor Safety Study (Median Value)
SMALL SMALL LOCA (S_2)		
S_2B	1.0×10^{-8}	1×10^{-8}
S_2H	1.01×10^{-5}	6×10^{-6}
S_2F	8.02×10^{-8}	1×10^{-7}
S_2HF	1.81×10^{-9}	1.6×10^{-9}
S_2D	5.92×10^{-6}	8.6×10^{-6}
S_2G	1.04×10^{-7}	8.5×10^{-8}
S_2DG	6.15×10^{-10}	7.31×10^{-10}
S_2C	2.6×10^{-6}	2.4×10^{-6}
S_2CD	1.54×10^{-8}	2.06×10^{-8}
S_2L	3.53×10^{-8}	3.7×10^{-8}
S_2LC	9.19×10^{-11}	8.88×10^{-11}
S_2K	6.43×10^{-8}	3.6×10^{-8}
S_2KC	1.67×10^{-10}	8.64×10^{-11}
S_2 Probabilities	1.9×10^{-5}	1.72×10^{-5}
TRANSIENT EVENTS (T)		
TML	3.15×10^{-6}	6×10^{-6}
TKQ	6.43×10^{-6}	3.6×10^{-6}
TKMQ	1.29×10^{-6}	7.2×10^{-7}
TMLB'	1.58×10^{-6}	3×10^{-6}
T Probabilities	1.24×10^{-5}	1.33×10^{-5}
INTERFACE SYSTEM (V)	4×10^{-6}	4×10^{-6}
REACTOR VESSEL RUPTURE (R)	1×10^{-7}	1×10^{-7}
TOTAL CORE MELT PROBABILITY	4.4×10^{-5}	4.35×10^{-5} (6×10^{-5})*

*WASH-1400 core melt probability using Monte Carlo sampling and smoothing technique.

NOTE: Sequences with suffix 'c' contain common mode contribution among its constituent events.

PWR Accident Sequence Symbol Key For Table 18

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of off-site AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system (ECI).

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system (ECR).

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system (SSR & AFWS).

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

Table 19
BWR CORE MELT PROBABILITIES (NUREG/CR-2906)

	Sequences	Brookhaven Re-evaluation (Point Estimate)	Reactor Safety Study (Median Value)
Large LOCA (A)	AE	1.21(-7)	2. (-7)
	AJ	6.38(-9)	1. (-8)
	AHI	1.10(-8)	1. (-8)
	AI	1.45(-13)	1. (-8)
	AGJ	3.28(-11)	7. (-11)
	AEG	6.20(-10)	8.1(-10)
	AGHI	5.67(-11)	6.7(-11)
A Probabilities		1.38(-7)	2.30(-7)
Small LOCA (S_1)	S_1E	2.39(-7)	2. (-7)
	S_1J	1.91(-8)	3. (-8)
	S_1I	4.36(-13)	4. (-8)
	S_1HI	3.31(-8)	4. (-8)
	S_1C	2.40(-9)	3.8 (-9)
	S_1GJ	9.85(-11)	2.3 (-10)
	S_1GI	2.25(-15)	2.3 (-10)
	S_1EG	1.61(-10)	1.2 (-10)
	S_1GHI	1.70(-10)	2.2 (-10)
S_1 Probabilities		2.94(-7)	3.14(-7)
Small-Small LOCA (S_2)			
	S_2J	6.38(-8)	1. (-7)
	S_2I	1.45(-12)	1. (-7)
	S_2HI	1.10(-7)	1. (-7)
	S_2E	1.67(-8)	5. (-8)
	S_2C	7.99(-9)	1. (-8)
	S_2CG	4.11(-11)	7. (-11)
	S_2GHI	5.67(-10)	6.7 (-10)
	S_2G	8.59(-11)	3.5 (-10)
	S_2GJ	3.28(-10)	7. (-10)
	S_2GI	7.48(-15)	2.3 (-10)
S_2 Probabilities		1.99(-7)	3.60(-7)
Transients (T)			
	TW	1.22(-5)	2. (-5)
	TC	7.99(-6)	1. (-5)
	TQUV	4.59(-7)	5. (-7)
T Probabilities		2.07(-5)	3.60(-5)
Pressure Vessel Rupture	PRVO	1 (-8)	
	PRV	1 (-7)	
PRV Probabilities		1.10(-7)	
Total Core-Melt Probability		2.14(-5)	3.14 (-5)

*Sequence symbol key follows the table.
Note: 1.0(-1) represents 1.0×10^{-1}

BWR Accident Sequence Symbol Key for Table 19

A	Large LOCA
B	Failure of electric power to ESF's
C	Failure of RPS
E	Failure of ECI
F	Failure of emergency cooling functionability
G	Failure of containment isolation to limit leakage to less than 100 volume percent per day
H	Failure of core spray recirculation system
I	Failure of low pressure recirculation system
HI	Failure of LPCRS (emergency service water system)
J	Failure of HPSWS
Q	Failure of feedwater system to provide core makeup water.
S_1	Small LOCA
S_2	Small-Small LOCA
T	Transient
U	Failure of HPCIS and RCICS
V	Failure of LECCS during transient
W	Failure to remove decay heat
PVRO	Failure of vessel rupture in oxidizing environment
PVR	Failure of vessel rupture in non-oxidizing environment

Table 20

PWR CONTAINMENT FAILURE MODE PROBABILITIES (WASH-1400)

Sequence	α* CRVSE	β CL	γ CR-B	δ CR-OP	ε CR-MT
AR	0.01	**	0	0	0.99
AHI	0.01	**	0	0	0.99
AG	0.01	**	0	1.00	0
AGI	0.01	**	0	1.00	0
AHG	0.01	**	0	0.49	0.49
AHGI	0.01	**	0	0.49	0.49
AF	0.01	**	0	1.00	0
AFI	0.01	**	0	1.00	0
AHF	0.01	**	0.12	0.08	0.78
AHFI	0.01	**	0.12	0.08	0.78
AE	0.01	**	0	0	0.99
AEI	0.01	**	0	0	0.99
AEG	0.01	**	0	0.39	0.60
AEGI	0.01		0	0.39	0.60
AEF	0.01	**	0	0	0.99
AEFI	0.01	**	0	0	0.99
AD	0.01	**	0	0	0.99
ADI	0.01	**	0	0	0.99
ADG	0.01	**	0	0.39	0.60
ADGI	0.01	**	0	0.39	0.60
ADF	0.01	**	0	0	0.99
ADFI	0.01	**	0	0	0.99
AC	-	**	-	-	-
ACI	-	**	-	-	-
ACH	0.01	**	0	0	0.99
ACHI	0.01	**	0	0	0.99
ACG	0.01	**	0	1.00	0
ACGI	0.01	**	0	1.00	0
ACHG	0.01	**	0	0.60	0.39
ACHGI	0.01	**	0	0.60	0.39
ACF	0.01	**	0	1.00	0
ACHF	0.01	**	0.24	0.56	0.19
ACE	0.01	**	0	0	0.99
ACEI	0.01	**	0	0	0.99
ACEG	0.01	**	0	0.39	0.60
ACEGI	0.01	**	0	0.39	0.60
ACEF	0.01	**	0.12	0.04	0.82
ACD	0.01	**	0	0	0.99
ACDI	0.01	**	0	0	0.99
ACDG	0.01	**	0	0.49	0.49
ACDGI	0.01	**	0	0.49	0.49

*See next page
**$\sim 10^{-4}$

Table 20 (Continued)

Sequence	α CRVSE	β CL	γ CR-B	δ CR-OP	ε CR-MT
ACDF	0.01	**	0.12	0.04	0.82
AB	0.01	**	0.12	0.04	0.82
S_2C	0.01	**	0	1.00	0
TMLB'	0.01	**	0.24	0.56	0.19

*α - Containment rupture due to a reactor vessel steam explosion.
β - Containment failure resulting from inadequate isolation of containment openings and penetrations.
γ - Containment failure due to hydrogen burning.
δ - Containment failure due to overpressure.
ε - Containment vessel melt-through.

**$\sim 10^{-4}$

Table 21
BWR CONTAINMENT FAILURE MODE PROBABILITIES (WASH-1400, Mark I)

Sequence	Containment Failure Mode Probabilities* α	β	γ	$\varepsilon\zeta$	$\varepsilon\eta$	$\varepsilon\theta$	ε	$\delta\zeta$	$\delta\eta$	$\delta\theta$	δ
A	--	--	--	--	--	--	--	--	--	--	--
AJ	0.01	0	1	0	0	0	0	0	0	0	0
AI	0.01	0	1	0	0	0	0	0	0	0	0
AH	--	--	--	--	--	--	--	--	--	--	--
AHJ	0.01	0	1	0	0	0	0	0	0	0	0
AHI	0.01	0.09	0.90	0	0	0	0	0	0	0	0
AG	--	--	--	--	--	--	--	--	--	--	--
AGJ	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
AGI	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
AGH	--	--	--	--	--	--	--	--	--	--	--
AGHJ	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
AGHI	0.01	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.90
AF	0.04	0.18	0.78	0	0	0	0	0	0	0	0
AFG	0.04	0.01	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
AE	0.01	0.07	0.92	0	0	0	0	0	0	0	0
AE	0	0.00	1	0	0	0	0	0	0	0	0
AEG	0.01	0.05	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
AEG	0	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.91
AD	--	--	--	--	--	--	--	--	--	--	--
ADJ	0.01	0	1	0	0	0	0	0	0	0	0
ADI	0.01	0	1	0	0	0	0	0	0	0	0
ADH	--	--	--	--	--	--	--	--	--	--	--
ADHJ	0.01	0	1	0	0	0	0	0	0	0	0
ADHI	0.01	0	1	0	0	0	0	0	0	0	0
ADF	0.01	0	1	0	0	0	0	0	0	0	0
ADE	0.01	0	1	0	0	0	0	0	0	0	0
ADE	0	0	1	0	0	0	0	0	0	0	0
AC	0.04	0.18	0.78	0	0	0	0	0	0	0	0
ACG	0.04	0.01	0	0.02	0.00	0.00	0.02	0.04	0.00	0.00	0.86
ACD	0	0	1	0	0	0	0	0	0	0	0
AB	0	0.00	1.00	0	0	0	0	0	0	0	0
ABG	0	0.00	0	0.02	0.00	0.00	0.02	0.05	0.00	0.00	0.91
ABD	0	0	1	0	0	0	0	0	0	0	0
TC	0.01	0	0.99	0	0	0	0	0	0	0	0
TW	0.01	0	1	0	0	0	0	0	0	0	0
TQUV	0.01	0	0.99	0	0	0	0	0	0	0	0

*Definition of containment failure mode is given in the following page.

BWR Containment Failure Mode Description

α	:	Containment failure due to steam explosion in vessel
β	:	Containment failure due to steam explosion in containment
γ	:	Containment failure due to overpressure - release through reactor building
γ'	:	Containment failure due to overpressure - release direct to atmosphere
δ	:	Containment isolation failure in dry well
ε	:	Containment isolation failure in wet well
ζ	:	Containment leakage greater than 2400 volume percent per day
η	:	Reactor building isolation failure
θ	:	Standby gas treatment system failure

Table 22

PWR AND BWR RELEASE CATEGORIES (WASH-1400)

RELEASE CATEGORY	PROBABILITY PER REACTOR-YR	TIME OF RELEASE (HR)	DURATION OF RELEASE (HR)	WARNING TIME FOR EVACUATION (HR)	ELEVATION OF RELEASE (METERS)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI of WASH-1400. Background on the isotope groups and release mechanisms is found in Appendix VII of WASH-1400.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI of WASH-1400.

Table 23

PWR DOMINANT ACCIDENT SEQUENCES
vs. RELEASE CATEGORIES (WASH-1400)

	RELEASE CATEGORIES			Core Melt				No Core Melt	
	1	2	3	4	5	6	7	8	9
	AB-α 1×10^{-11}	AB-γ 1×10^{-10}	AD-α 2×10^{-8}	ACD-β 1×10^{-11}	AD-β 4×10^{-9}	AB-ε 1×10^{-9}	AD-ε 2×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
LARGE LOCA	AF-α 1×10^{-10}	AB-δ 4×10^{-11}	AH-α 1×10^{-8}		AH-β 3×10^{-9}	AHF-ε 1×10^{-10}	AH-ε 1×10^{-6}		
A	ACD-α 5×10^{-11}	AHF-γ 2×10^{-11}	AF-δ 1×10^{-8}			ADF-ε 2×10^{-10}			
	AG-α 9×10^{-11}		AG-δ 9×10^{-9}						
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}.
	S_1B-α 3×10^{-11}	S_1B-γ 4×10^{-10}	S_1D-α 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9}	S_1DF-ε 3×10^{-10}	S_1D-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
SMALL LOCA	S_1CD-α 7×10^{-11}	S_1B-δ 1×10^{-10}	S_1H-α 3×10^{-8}		S_1D-β 6×10^{-9}	S_1B-ε 2×10^{-9}	S_1H-ε 3×10^{-6}		
S_1	S_1F-α 3×10^{-10}	S_1HF-γ 6×10^{-11}	S_1F-δ 3×10^{-8}			S_1HF-ε 4×10^{-10}			
	S_1G-α 3×10^{-10}		S_1G-δ 3×10^{-8}						
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}

Step 7

Table 23 (Continued)

	RELEASE CATEGORIES			Core Melt				No Core Melt	
	1	2	3	4	5	6	7	8	9
	S_2B-α 1×10^{-10}	S_2B-γ 1×10^{-9}	S_2D-α 9×10^{-8}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8}	S_2B-ε 8×10^{-9}	S_2D-ε 9×10^{-6}		
	S_2F-α 1×10^{-9}	S_2HF-γ 2×10^{-10}	S_2H-α 6×10^{-8}		S_2H-β 1×10^{-8}	S_2CD-ε 2×10^{-8}	S_2H-ε 6×10^{-6}		
SMALL LOCA S_2	S_2CD-α 2×10^{-10}	S_2B-δ 4×10^{-10}	S_2F-δ 1×10^{-7}			S_2HF-ε 1×10^{-9}			
	S_2G-α 9×10^{-10}		S_2C-δ 2×10^{-6}						
	S_2C-α 2×10^{-8}		S_2G-δ 9×10^{-8}						
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
	RC-α 2×10^{-12}	RC-γ 3×10^{-11}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
REACTOR VESSEL RUPTURE - R		RF-δ 1×10^{-11}							
		RC-δ 1×10^{-12}							
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		

Step 7

Table 23 (Continued)

	RELEASE CATEGORIES			Core Melt				No Core Melt	
	1	2	3	4	5	6	7	8	9
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 3×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
	(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY								
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-8}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO PWR ACCIDENT SEQUENCE SYMBOLS FOR TABLE 23

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment vessel melt-through.

Table 24

BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE vs. RELEASE CATEGORY (WASH-1400)

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	AE-α 2×10^{-9} AJ-α 1×10^{-10} AHI-α 1×10^{-10} AI-α 1×10^{-10}	AE-γ' 3×10^{-8} AE-β 1×10^{-8} AJ-γ' 2×10^{-9} AI-γ' 2×10^{-9} AHI-γ' 2×10^{-9}	AE-γ 1×10^{-7} AJ-γ 1×10^{-8} AI-γ 1×10^{-8} AHI-γ 1×10^{-8}	AGJ-δ 6×10^{-11} AEG-δ 7×10^{-10} AGHI-δ 6×10^{-11}	A 1×10^{-4}
A Probabilities	8×10^{-9}	6×10^{-8}	2×10^{-7}	2×10^{-8}	1×10^{-4}
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S)	S_1E-α 2×10^{-9} S_1J-α 3×10^{-10} S_1I-α 4×10^{-10} S_1HI-α 4×10^{-10}	S_1E-γ' 4×10^{-8} S_1E-β 1×10^{-8} S_1J-γ' 7×10^{-9} S_1I-γ' 7×10^{-9}	SE-γ 1×10^{-7} S_1J-γ 3×10^{-8} S_1I-γ 4×10^{-8} S_1HI-γ 2×10^{-8}	S_1GJ-δ 2×10^{-10} S_1GI-δ 2×10^{-10} S_1EI-ε 1×10^{-10} S_1GHI-δ 2×10^{-10}	

Table 24 (Continued)

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1) (Continued)		$S_1HI-\gamma'$ $6x10^{-9}$	$S_1C-\gamma$ $3x10^{-9}$		
S_1 Probabilities	$1x10^{-8}$	$9x10^{-8}$	$2x10^{-7}$	$2x10^{-8}$	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	$S_1J-\alpha$ $1x10^{-9}$	$S_2E-\gamma'$ $1x10^{-8}$	$S_2E-\gamma$ $4x10^{-8}$	$S_2CG-\delta$ $6x10^{-11}$	
	$S_2I-\alpha$ $1x10^{-9}$	$S_2E-\beta$ $4x10^{-9}$	$S_2J\gamma$ $8x10^{-8}$	$S_2GHI-\delta$ $6x10^{-10}$	
	$S_2HI-\alpha$ $1x10^{-9}$	$S_2J-\gamma'$ $2x10^{-8}$	$S_2I-\gamma$ $9x10^{-8}$	$S_2EG-\delta$ $3x10^{-10}$	
	$S_2E-\alpha$ $5x10^{-10}$	$S_2I-\gamma'$ $2x10^{-8}$	$S_2HI-\gamma$ $9x10^{-8}$	$S_2GJ-\delta$ $6x10^{-10}$	
		$S_2HI-\gamma'$ $2x10^{-8}$	$S_2C-\gamma$ $8x10^{-9}$	$S_2GI-\delta$ $2x10^{-10}$	
S_2 Probabilities	$2x10^{-8}$	$1x10^{-7}$	$4x10^{-7}$	$4x10^{-8}$	
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	$TW-\alpha$ $2x10^{-7}$	$TW-\gamma'$ $3x10^{-6}$	$TW-\gamma$ $1x10^{-5}$		
	$TC-\alpha$ $1x10^{-7}$	$TQUV-\gamma'$ $8x10^{-8}$	$TC-\gamma$ $1x10^{-5}$		

Table 24 (Continued)

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T) (Continued)	TQUV-α 5×10^{-9}		TQUV-γ 4×10^{-7}		
T Probabilities	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	
PRESSURE VESSEL RUPTURE ACCIDENTS (R)		P.V. RUPT. 1×10^{-8} Oxidizing Atmosphere	P.V. RUPT. 1×10^{-7} Non-oxidizing Atmosphere		
R Probabilities	2×10^{-9}	2×10^{-8}	1×10^{-7}	1×10^{-8}	
SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	1×10^{-4}
LOWER BOUND (5% VALUE)	1×10^{-7}	1×10^{-6}	5×10^{-6}	5×10^{-7}	1×10^{-5}
UPPER BOUND (95% VALUE)	8×10^{-6}	3×10^{-5}	8×10^{-5}	1×10^{-5}	1×10^{-3}

NOTE: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS FOR TABLE 24

A	-	Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.
B	-	Failure of electric power to ESFs.
C	-	Failure of the reactor protection system.
D	-	Failure of vapor suppression.
E	-	Failure of emergency core cooling injection.
F	-	Failure of emergency core cooling functionability.
G	-	Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H	-	Failure of core spray recirculation system.
I	-	Failure of low pressure recirculation system.
J	-	Failure of high pressure service water system.
M	-	Failure of safety/relief valves to open.
P	-	Failure of safety/relief valves to reclose after opening.
Q	-	Failure of normal feedwater system to provide core make-up water.
S_1	-	Small pipe break with an equivalent diameter of about 2"-6".
S_2	-	Small pipe break with an equivalent diameter of about 1/2"-2".
T	-	Transient event.
U	-	Failure of HPCI or RCIC to provide core make-up water.
V	-	Failure of low pressure ECCS to provide core make-up water.
W	-	Failure to remove residual core heat.
α	-	Containment failure due to steam explosion in vessel.
β	-	Containment failure due to steam explosion in containment.
γ	-	Containment failure due to overpressure - release through reactor building.
γ'	-	Containment failure due to overpressure - release direct to atmosphere.
δ	-	Containment isolation failure in drywell.
ε	-	Containment isolation failure in wetwell.
ζ	-	Containment leakage greater than 2400 volume per cent per day.
η	-	Reactor building isolation failure.
θ	-	Standby gas treatment system failure.

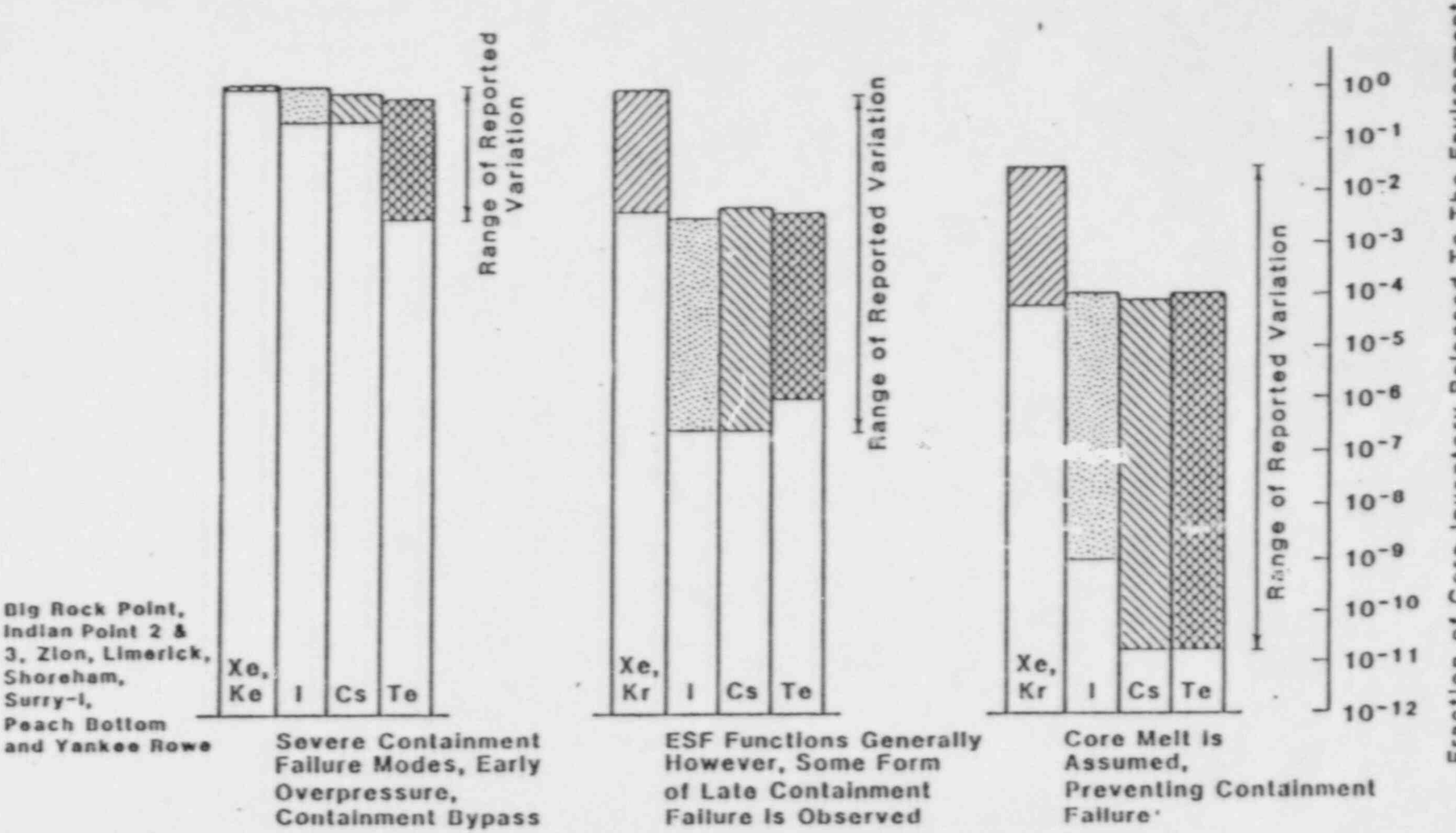

Table 25 Range of Radionuclide Release Fractions from Listed PRA Studies

APPENDIX B

LIST OF ACRONYMS

ADS	- Automatic Depressurization System
AFW	- Auxiliary Feedwater
AFWS	- Auxiliary Feedwater System
ASEP	- Accident Sequence Evaluation Program
ATWS	- Anticipated Transient Without Scram
BWR	- Boiling Water Reactor
CFCS	- Containment Fan Cooler System
CHRS	- Containment Heat Removal System
CRD	- Control Rod Drive
CS	- Containment Spray
CS	- Core Spray (BWR only)
CSI	- Containment Spray Injection
CSIS	- Containment Spray Injection System
CSR	- Containment Spray Recirculation
CSRS	- Containment Spray Recirculation System
CSRS	- Core Spray Recirculation System (BWR only)
CSS	- Containment Spray System
CSS	- Core Spray System (BWR only)
ECCS	- Emergency Core Cooling System
ECI	- Emergency Coolant Injection
ECR	- Emergency Coolant Recirculation
EFWS	- Emergency Feedwater System
ESAS	- Engineered Safeguards Actuation System
ESF	- Engineered Safety Feature
ESFAS	- Engineered Safety Feature Actuation System
ESW	- Emergency Service Water
ESWS	- Emergency Service Water System
FSAR	- Final Safety Analysis Report

LIST OF ACRONYMS (Continued)

FW - Feedwater
HMS - Hydrogen Mitigation System
HPCI - High Pressure Coolant Injection System
HPCS - High Pressure Core Spray
HPI - High Pressure Injection
HPIS - High Pressure Injection System
HPR - High Pressure Recirculation
HPRS - High Pressure Recurculation System
HPSWS - High Pressure Service Water System
ICS - Ice Condenser System
IDCOR - Industry Degraded Core Program
IREP - Interim Reliability Evaluation Program
LOCA - Loss of Coolant Accident
LOOP - Loss of Offsite Power
LPCI - Low Pressure Coolant Injection
LPCS - Low Pressure Core Spray
LPCR - Low Pressure Coolant Recirculation
LPI - Low Pressure Injection
LPIS - Low Pressure Injection System
LPR - Low Pressure Recirculation
LPRS - Low Pressure Recirculation System
LPSWS - Low Pressure Service Water System
MFW - Main Feedwater
MSIV - Main Steam Isolation Valve
NPSH - Net Position Suction Head
NSSS - Nuclear Steam Supply System
PCS - Power Conversion System
PORV - Pressure Operated Relief Valve
PRA - Probabilistic Risk Assessment
PWR - Pressurized Water Reactor
RCIC - Reactor Core Isolation Cooling
RCICS - Reactor Core Isolation Cooling System
RCS - Reactor Coolant System
RHR - Residual Heat Removal

LIST OF ACRONYMS (Continued)

RHR-SPC	- Residual Heat Removal-Suppression Pool Cooling
RHRSW	- Residual Heat Removal Service Water
RPS	- Reactor Protection System
RPT	- Recirculation Pump Trip
RSS	- Reactor Safety Study (WASH-1400)
RSSMAP	- Reactor Safety Study Methodology Applications Program
SLC	- Standby Liquid Control
SRV	- Safety Relief Valve
SWS	- Service Water System
VSS	- Vapor Suppression System