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Categorization of Reactor Safety Issues From a Risk Perspective

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ABSTRACT

This report presents the results of an effort to identify and rank reactor safety and risk issues identified from past Probabilistic Risk Assessments (PRAs) and other safety analyses. Because of the varied scope of these analyses, the list of issues may be incomplete. Nevertheless, those studies comprised ordered analyses to whatever their respective depths; hence, they warranted scrutiny for whatever insights they could reveal with respect to issue importance. The top ranked issues in terms of their contribution to the uncertainty in risk are described in some detail. All of these risk issues are compared to the "generic safety issues" for completeness and omissions.

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PREFACE

This is a new direction to the prioritization of research. Past work has focused on trying to prioritize research areas directly. This has led to some useful results; however, deficiencies have existed in the approach because it was cumbersome and failed to incorporate the risk importance of issues directly into the prioritization.

The current work is designed to identify the important reactor risk issues based on the PRAs and other analyses that have been done to date. The information base will be periodically updated. After identification, the issues were rated as to their contribution to the overall uncertainty in risk.

Because the issues are based upon analyses which were performed to various depths, the list of issues is probably incomplete. Our intent is to continue to refine and update the list of issues and the rankings for each issue. Although in this report the rankings are categorical in nature, we will provide interval estimates of the uncertainty in subsequent reports as time and resources allow.

The list of issues can be useful for many other regulatory activities besides the prioritization of research. Because of the flexibility of the method and the ability to quickly manipulate the data base with decision rules and criteria, the issues can be used to provide insights into the risk effectiveness of regulations and into the completeness of regulations.

Finally, the sole responsibility of this initial ranking is that of the Division of Risk Analysis and Operations in the Office of Nuclear Regulatory Research. Other divisions and offices have not concurred in this effort. Comments and suggestions concerning the list of issues or associated ranking and potential uses of this data base are invited.

Categorization of Reactor Safety Issues from A Risk Perspective

1. INTRODUCTION

In mid-September 1984, the Division of Risk Analysis and Operations (DRAO) initiated a task to identify and categorize technical issues that have the potential for impacting our understanding of the risk from light water reactor power plants. This categorization would provide useful information in planning research initiatives for the upcoming fiscal year.

Since limited time was available, the results stem from the subjective judgments of specialists with broad backgrounds in light water reactor safety who drew their knowledge from existing risk-related studies. These studies included both past probabilistic risk assessments (PRAs) and research results and analyses that had been completed for specific groups of issues. For convenience, two groups were formed. One addressed the systems analysis, events, and phenomenology contributing to core melt frequency from internal events and three external events: fire, seismic, and flooding. The second group addressed those issues contributing to offsite consequence given that a core melt accident had occurred. The groups worked separately, interfacing occasionally to achieve consistency in format, when appropriate, and where possible, consistency in criteria for categorization. Fundamental differences in the nature of the issues, the level of understanding, and the amount of data precluded absolute consistency between the two groups.

In this study, broad areas were identified that were important in the understanding of risk and uncertainty in risk. These areas were then subdivided into more specific issues for categorization and ranking. The specific issues were categorized with respect to such criteria as their contribution to the uncertainty in risk and the researchability of the issue. A "small", "medium", and "large" categorization was used. The issues were then ranked in order to provide an overall evaluation of the importance of a particular issue from a risk perspective.

As with any study of this breadth, considerable caution is advisable in utilizing the results. This is particularly true of this study in that limited time was available to conduct the study and review of the results was limited. The document was briefly reviewed by a small set of technical experts and personnel from DRAO. Complete consensus among the reviewers was not achieved in all cases, and some final changes were made to the categorizations by the DRAO staff. The potential error bounds in the results should be treated as large, and it should be emphasized that the categorization, although achieving a reasonably high degree of consensus among those working on the project, should not be regarded as a consensus of any broader group. The ranking also reflects a singular perspective, that is the contribution of the issue to the uncertainties in our understanding of the risk from LWR power plants. Issues judged to have a small contribution to these uncertainties may well be judged important for other reasons relevant to the NRC.

Finally, it must be recognized that there are limitations in our understanding in LWR risk stemming from limitations in the risk analyses and risk-related studies that have been performed. The risk information base is most comprehensive for the internal events. Fewer plant risk studies currently exist in which external events have been thoroughly analyzed. Additionally, the accident phenomenology (containment loads, containment response, and source term definition) are still areas of wide diversity of opinion.

Within the internal events area, it should also be recognized that although many issues have been explicitly treated in past PRAs, many have not. Judgments contained in this categorization addressing those issues which have not been explicitly treated in past PRAs are most speculative and should be used with increased caution. Some areas not explicitly addressed in many past PRAs are listed below. No particular ordering of this group is intended.

Partial Failures

Design Adequacy

Adequacy of Test and Maintenance Practices

Effect of Aging on Component Reliability (also burn-in phenomena)

Adequacy of Equipment Qualification

Equipment Operability in Sequence Environment

Diagnostic Human Errors

Environmentally-Related Common Cause

Similar Parts-Related Common Cause

Sabotage

Long-Term Accident Response (beyond approximately 24 hours)

Innovative Operator Accident Response Actions

Effects of Training and Operator Experience/Conditioning on Operator Response

Equipment Installation Problems/Manufacturing Defects

Cold-Shutdown and Non-Full-Power-Operation Events.

Many of the items in the above list can be classified as root causes of failure. Root causes have been addressed only to a limited extent in past PRAs. In general, root causes do not affect the core melt frequency, but they affect the usability of the PRA results. The important root causes can be identified and factored into plant reliability programs, inspection programs, etc.

The remainder of this report contains the results of this effort. Chapter 2 presents a listing and discussion of those issues considered to be most important to the uncertainty in risk. These issues were selected from the more complete list contained in Appendices A and B. The issues in Appendices A and B were categorized using the approaches described in Chapters 3 and 4, respectively. They were then ranked using the method presented in Chapter 6. Chapter 7 presents the complete results of the ranking and the rationale for selecting the issues identified in Chapter 2. Chapter 5 identifies relationships between the risk issues of Chapter 2 and the NRC's generic safety issues. Appendix D is similar to Chapter 5, except that it includes the complete set of risk issues from Appendices A and B. Chapter 8 presents some thoughts concerning followup work, and Appendix C contains some of the sequence information from past PRAs that was used to guide this effort.

2. DISCUSSION OF TOP-RANKED ISSUES

This chapter contains a discussion of those issues that have been selected as being the largest contributors to the uncertainty in risk. The details of how those issues were selected are contained in Chapters 6 and 7. The information presented here is intended to provide a brief technical summary and explanation of the important issues. A complete set of risk issues is contained in Appendices A and B. Many of the issues contained in Appendices A and B, but not discussed here, are also considered to be important, although somewhat less so on a relative scale. Table 2-1 below lists the issues that were selected for presentation in this chapter. The identifiers are the same as used in Appendices A and B. No rank ordering of the issues in Table 2-1 is intended; rather, they should be treated as a group of important issues.

2.1 Systems Issues

In this section, the 19 most important issues dealing with the sequence likelihood portion of the risk equation are described in detail (see Chapters 6 and 7 for a description of the importance ranking process). The most important issues are those that contribute the most to the uncertainty in the core-melt frequency. Reasons for the high importance ranking of each issue are presented along with some ideas concerning the type of research that would be required to resolve the issue.

2.1.1 Internal Event Issues

I.A.i Hardware Issues - Equipment Failure Reporting Accuracy

Inaccurate reporting of equipment failures can lead to errors in assessing component failure rates and this, in turn, can cause large deviations in estimated core-melt frequency.

One type of equipment failure reporting problem is the recording of component demands. The number of component demands is typically not reported in the open literature and are often based on information given in the plant technical specifications. The estimates obtained from this method have been compared with estimates based on actual plant experience. These comparisons have typically shown that the actual number of demands are greater than the estimate based on technical specifications testing requirements. The denominator that is used in the calculation of the failure rate is therefore underestimated and the calculated failure rates are too high.

Table 2-1 List of Important Risk Issues

Internal Event Issues

I.a.i	Hardware Issues - Equipment Failure Reporting Accuracy
II.a.iv.	Issues Affecting Human Behavior - Ability to Assess Common-Cause Failures Caused by Humans
III.b.i	Infrequent Initiators - Modeling of Interactions Between Initiating Events and Mitigating Systems
IV.c.i	Common Cause - Reporting Accuracy
IV.c.iii	Common Cause - Inadequate Plant Procedures
IV.c.iv	Common Cause - Common Physical Cause
IV.d.i	Information in Analyses - Design Information
IV.d.ii	Information in Analyses - Operational Information
V.a	Sequence Analysis - Definition of Event Tree Sequences
V.c.ii	Modeling System Interactions on Event Trees - Nonhardwired Interactions (Common-Cause, Corrosion, etc.)

Fire Issues

F.IV.ii	Suppression Effectiveness - Secondary Detrimental Effects
F.VI.i	Damage Thresholds and Mechanisms - Component Fragilities for Temperature, Smoke, Moisture, and Corrosion
F.VII.ii	Scenario-Related Systems Response - Remote Shutdown System Effectiveness
F.VII.iv	Scenario-Related Systems Response - Earthquake-Induced Fires or Suppression Actuation

Seismic Issues

S.I.A	Hardware Issues - Relay Chatter and Locking Circuits
S.I.D	Hardware Issues - Ductility Effects in Structural Failures
S.I.G	Hardware Issues - Aging Effects on Seismic Fragilities
S.I.H	Hardware Issues - Correlation of Component Fragilities
S.IV.A	System Response - Local Amplification of Ground Motion

Table 2-1 List of Important Risk Issues (Continued)

Flooding Issues

NONE

Containment/Consequence Issues

R.1.a	In-Vessel Issues - Natural Convection
R.1.d	In-Vessel Issues - Steam Explosion Induced Containment Failure
R.1.f	In-Vessel Issues - Alternate Primary System Failures
R.1.g	In-Vessel Issues - Fuel Melt Progression
R.1.h	In-Vessel Issues - Debris Transport and Interactions with Primary System Structures
R.2.d	Ex-Vessel Issues - Gas Transport
R.2.f	Ex-Vessel Issues - Flame Acceleration and Detonation
R.2.m	Ex-Vessel Issues - Direct Heating
R.3.a	In-Vessel Fission Products - Release of Fission Products from Fuel
R.3.c	In-Vessel Fission Products - Transport and Deposition with Primary System
R.3.d	In-Vessel Fission Products - Chemical Transformations of Fission Products within the Primary System
R.3.g	In-Vessel Fission Products - Revolatilization of Fission Products In-Vessel
R.4.c	Ex-Vessel Fission Products - Release from Core Concrete Interactions
R.5.e	Health and Economic Consequences - Modeling of Emergency Response
R.6.a	Equipment Issues - Detection and Monitoring Systems
R.6.c	Equipment Issues - Essential Equipment Performance During Severe Accidents
R.7.f	Containment Performance - Response to Static Over-pressurization and Increased Temperatures
R.8A.c	Operations - Operator Training and Performance

A second type of equipment failure reporting problem is the recording of component failures. In a recent study (5), LER records reported in Nuclear Power Experience (NPE) and in-house utility records with respect to diesel generator problems were compared. A significant discrepancy was found; for a particular plant, in-house records indicated more problems than were reported in NPE. The discrepancy was thought to occur due to different criteria used to collect in-house data in comparison with the requirements for filing LERs. In this case, the numerator that is used in the calculation of failure rates is underestimated and the calculated failure rates are too low.

A third reporting inaccuracy problem comes from subjective definitions of component failures. There have been several cases where degraded performance events have been called complete component failures. Other examples of misclassification of events have also occurred.

A final reporting problem is derived from inadequate descriptions of root causes of component failures. The root cause is the specific reason why the component failure occurred, e.g., inadequate procedures, wearout, corrosion, etc.

The problem could be alleviated by the reporting of more accurate information and the development of standards in this regard. Steps recently taken by the Institute for Nuclear Power Operations (INPO) will go far to improve this situation.

II.A.iv Issues Affecting Human Behavior - Ability to Assess Common-Cause Failures Caused by Humans

One of the highest contributors to risk and risk uncertainty is the potential for a human to commit multiple errors which cause multiple components or systems to fail. Common-cause failures are in general one of the dominant contributors to risk uncertainty, and human actions are often the sources of these common-cause failures. The common-cause failure potential of the human and the ability (or inability) to assess this potential is the area in human reliability analysis where some of the largest uncertainties and the largest potential risk impacts exist. Some PRAs attempt to be conservative in estimating common-cause human error probabilities but, in general, the common-cause probabilities could be significantly larger than estimated.

Research could address this issue by evaluating in-plant data and simulator performance to relate common-cause potentials to basic performance and procedure attributes.

III.B.i Initiating Event Issues - Infrequent Initiators that Occur During Power Operation involving interactions between initiating events and mitigating systems

Several different initiating events caused by a loss of a support system have been found to be important by past PRAs. This type of initiating event is particularly important because it can trip the plant and cripple mitigating systems at the same time. This issue is a large contributor to the uncertainty in core melt frequency because the interactions that occur between these initiators and the rest of the plant are sometimes subtle and difficult to find. An example of this issue would be the loss of a DC bus. In many cases, it is difficult to know whether loss of a particular bus will trip the plant and what mitigating systems will be left in a degraded state (e.g., the Power Conversion System). The plant trip may occur after several minutes, making it very difficult for the operator to know what is going on and what mitigating systems are available.

Resolution of this issue would appear to require plant specific studies of the interactions between support system initiating events and the safety systems necessary to shut the plant down under abnormal conditions.

IV.C Common-Cause Issues

Nuclear power plant safety systems employ redundancy as a means of achieving high reliability. However, redundant systems may still be vulnerable to single events involving multiple failures. Such events have come to be called common-cause failures (CCF), and because of their potential impact on plant safety and availability, they are a concern in the nuclear industry.

In order to achieve high reliability/availability, systems are designed with redundancy such as a m-out-of-n (m/n) system for which only m trains are required for mission success. Although redundant systems are designed to tolerate multiple individual failures, it is generally recognized that such systems are vulnerable to CCFs. As the degree of redundancy increases, the reliability/availability of the system will be limited by CCFs. For example, in a recent study (1) of several related residual heat removal systems with various degrees of redundancy, it was concluded that the (1/4) system is only slightly better than the (2/4) system because of the assessed importance of two potential CCFs (premature shut-off of RHR pumps and failure of sump level signal). This example is illustrative of the importance of properly including CCF system failures in evaluating safety system reliability.

The important CCF issues are described below. These issues are related to 1) the quantification of CCFs (IV.c.i), and 2) identification of the more important causes of CCFs at plants (IV.c.iii and IV.c.iv).

IV.C.i Common-Cause - Reporting Accuracy

The problem here deals more with interpretation of CCF reports. If the reports were more accurate in definition and explanation, this problem would not be as significant. The primary reason that a problem exists with interpretation of CCF reports is because the term CCF has come to mean many different things to different people and organizations. Attempts to define CCF have failed to gain universal acceptance because the definitions are often ambiguous and tend to be too inclusive or too exclusive, depending on one's viewpoint. The existing ambiguity with the CCF phenomenon is quantitatively apparent in two recent analyses of diesel generator failure data. In one study (2) the mean fraction of events with "some common-cause potential" is reported to be 0.40, while in the other study (3), the fraction of total diesel generator failures attributable to dependent failures is over an order of magnitude lower and is estimated to have median value of approximately 0.02. Although these analyses do not reflect identical data bases, it is clear that the major difference is due to the differences in the analysts' assessment of CCFs and associated data parameterization. Assessment requires identification of the root causes of CCFs, as well as deciding which root causes potentially affected multiple components. As discussed in issue I.A.i, root cause information is often inadequate. Resolution of this problem would be aided by

1) a suitable definition of CCFs with respect to different classes of CCFs and

2) a demonstration that CCF data can be extracted from event reports in a consistent and reproducible way using that definition.

IV.C.iii Common-Causes Because of Inadequate Plant Procedures

Procedures delineate the primary interface that plant personnel have with components and systems. If these procedures are in error, redundant system trains and components are potentially susceptible to CCF. For example, the Salem nuclear reactor recently experienced a failure to scram because redundant trip breakers failed to function properly. The cause of the breaker failures was attributed to an inadequate procedure for lubrication of the breakers. Identification of inadequate procedures can be very difficult; one must be thoroughly familiar with the components the procedure affects as well as the cause/effect relationship between the procedural steps and component reliability. Some components and systems are so complex that even the manufacturer of them may have a hard time identifying all subtle component and system failure modes.

IV.C.iv Common Physical Cause

Nuclear operating experience has shown that adverse environments can degrade system and component reliability. For example, a recent study (4) has indicated that a significant percentage of BWR CCFs were due to exposure of redundant components/systems to adverse environments such as moisture and corrosion. These physical causes represent important "root causes" of failure.

Techniques are currently being developed that will be capable of identifying physical causes that may lead to system CCFs. These techniques use fault trees and other PRA methods. Improved quantification requires better environmental failure data.

IV.D Issues Associated with Accuracy of Information Used in Safety Analysis (IV.D.i, IV.D.ii)

In order to conduct a safety or risk analysis, plant design and operations information must be gathered and assimilated. The amount of information required will depend on the scope and purpose of the analysis being performed. The quality of safety analysis results is dependent on the accuracy of the input information. For example, comparison of the Calvert Cliffs RSSMAP PRA (FSAR information) and the Calvert Cliffs IREP PRA (more accurate information) indicates a factor of 3 difference in estimated core damage frequency. However, it is conceivable that poor or inadequate information could cause large deviations in estimated core damage frequencies.

Resolution of this problem requires a close working relationship between the owner of the plant being analyzed and the organization responsible for performing the analysis.

Issue V.a -- Issues Related to Accident Sequence Analysis - Uncertainties Associated with the Definition of Accident Sequences

Uncertainties associated with the definition of accident sequences have the potential to cause large uncertainties in the estimated core melt frequency. The sequences to be analyzed in a PRA are defined based on the current understanding of accident phenomenology and system interrelationships. If either the understanding of the accident phenomenology or the understanding of the systems interactions changes, a significant portion of the set of sequences as defined by the original study could change. This could cause large differences in the calculated core melt frequency.

Two examples of this are given below: 1) WASH-1400, the Grand Gulf RSSMAP, and the Browns Ferry IREP PRAs all have found the long-term loss of residual heat removal sequence to be a major contributor to the core melt frequency at BWRs. This sequence may not be as important as originally thought because the operator has a relatively long period (~one day) to find alternate means of containment heat removal and emergency injection; 2) Several of the early PRAs did not identify the sequence in which core melt is caused by battery depletion after loss of all AC power at the plant. The inclusion or exclusion of potentially high frequency sequences like the ones described above can cause large differences in calculated core melt frequency.

V.C.ii Modeling System Interactions on Event Trees - Non-hardwired Interactions (Common Cause, Corrosion, etc.)

Issues IV.C.iii and IV.C.iv discussed previously dealt with the identification of single common-cause events that can cause failure of a redundant system. This issue is similar to those issues except that it is concerned with the identification of common-cause events that fail more than one mitigating system, i.e., the systems depicted on PRA event trees. The uncertainties associated with this issue are similar to those associated with IV.c.iii and IV.c.iv. For discussion, see those issues.

2.1.2 Fire Issues

F.IV.ii Suppression Effectiveness - Secondary Detrimental Effects

In order to protect redundant safety trains from the effects of fire, many utilities have installed automatic or manual fire suppression systems which are designed to extinguish fires quickly, thereby limiting fire damage to as few safety components as possible. Often the redundant safety components being protected are located in close proximity in the same room, and therefore, although protected from direct burning, the components may be damaged by common suppression environments. A similar situation can occur even when redundant trains are located in separate rooms, if smoke or heat from a burning area can actuate suppression systems in both of the separated rooms.

Current design practice appears to lack the necessary technical basis to resolve the issue. In particular, the following information appears to be generally lacking:

- the degree to which suppression systems can be actuated by smoke and heat transported from one fire area to another

- tests or analyses demonstrating the effectiveness of spray shields or equipment shrouds in preventing suppression damage to equipment

- fire fighting techniques to prevent damage to redundant equipment located near, adjacent to, or along the path to a fire

- the sensitivity of suppression systems to seismic common-mode actuation

One or more of these information needs apply to virtually every power plant; however, they have not been treated in fire risk assessments to date. In fact, all fire risk assessments have treated suppression systems as positive design features which always improve the level of plant safety, while in reality, suppression activities may potentially damage redundant systems.

F.VI.i Damage Thresholds and Mechanisms - Component Fragilities for Temperature, Smoke, Moisture, Corrosion

Because many power plants have redundant equipment in close proximity (often in the same cabinet), damage can occur as a result of fire environments. However, little is known about the damage thresholds or failure modes of equipment by mechanisms other than burning. The effects of smoke, corrosion, humidity, water sprays, and high temperature are largely unknown and have not been treated in power plant fire analyses to date. Unlike equipment designed and tested for seismic or LOCA environments, equipment expected to function in a fire environment appears never to be tested for its functionality under expected conditions. Equipment not located in close proximity to a fire may be susceptible to damage without actually burning. This has been demonstrated by the recent fire experience of the British Navy during the Falkland Islands Campaign and by the U.S. Navy. Both navies have found certain electronics equipment to be easily and quickly damaged by smoke and corrosion to the point of loss of operational capability. This experience even involved equipment located in rooms separated by walls from the fire sources.

Fire risk analyses to date have assumed that fire damages equipment only by burning or extreme heating. If other elements of a fire environment can damage equipment, then the risk of fire could be even higher than the already significant values estimated to date (e.g., Indian Point, Zion, Limerick, Millstone 3, Seabrook, and Big Rock Point PRA's). This increased fire risk could apply to all power plants, although it would be more significant for power plants relying on spatial separation versus passive barriers.

F.VII.ii Scenario-Related Systems Response - Remote Shutdown Effectiveness

Nuclear power plants generally use redundant trains of safety components which are physically separated to prevent common-mode failure vulnerability. For control circuits this philosophy of train separation usually breaks down in control rooms, alternate control panels, and sometimes in cable spreading rooms. Under these situations, train separation is often replaced with duplication of certain train functions at other locations (e.g. two valve position switches for the same valve - one in the control room and one on an alternate control panel). Because duplicated components within the same safety train must eventually interface with common train components, electrical independence must be assured to prevent fault propagation and spurious signals from damaging both duplicated components or from creating plant states for which an operator has little or no control.

Fires involving control room panels, relay panels, alternate control panels, or cable spreading rooms represent a credible mechanism for damaging redundant trains of control equipment and thereby placing reliance on duplicate control capability. Many power plants and all fire PRA's to date have assumed that duplicate control capability has been assured by current designs. However, there is evidence that, at least in a probabilistic sense, this assumption is not always true because:

- the assumption is sometimes made that no circuit damage occurs prior to transferring control to a duplicate location

- the analyses usually consider the occurrence of only one spurious signal at a time as a result of a control panel fire

- duplicate control capability may be provided for a select group of front-line and support systems, while assuming that other safety and nonsafety components during a fire remain as is, take their benign failure positions, or respond to proper automatic commands.

The ability of operators to cope with spurious signals constitutes a key element in assessing the safety significance of control system fires and the effectiveness of remote shutdown systems.

The ability of operators to cope with spurious signals is most questionable under circumstances where operators have the least diagnostic and control capability. This is likely to occur whenever operators take control of a plant from remote shutdown areas, after evacuating the control room during a fire. If control room panels remain electrically activated and unisolated under these conditions, an operator at a remote station may need to override or correct a variety of spurious signals.

F.VII.iv Scenario-Related Systems Response - Earthquake Induced Fires or Suppression Actuation

Nuclear power plants have not, in general, designed fire protection systems to withstand earthquakes; and they have not considered the likelihood of fires resulting from seismic events. There are regulatory requirements for fire suppression systems to be "capable of delivering water to manual hose stations located within hose reach of areas containing equipment for safe plant shutdown" in the event of a Safe Shutdown Earthquake (SSE). However, other fire protection system components (e.g., detectors, sprinkler heads, fire pumps, flow sensors) normally only meet commercial standards and are not designed or tested to withstand earthquakes. Earthquakes may reasonably be assumed to cause fires, particularly in nonsafety systems not designed for earthquakes. Under these circumstances, fire protection systems may fail either by not being available for fire extinguishment or by becoming a threat to nearby equipment by spraying or flooding.

To address this issue, an analysis could be made first to assess the physical likelihood that a seismic event could cause a fire, particularly in nonseismically-designed systems, and then to evaluate the manner in which fire protection system components could be affected by an earthquake. Such an analysis could aid in the determination of whether earthquake-induced fires or suppression actuation can be expected to occur.

2.1.3 Seismic Issues

S.I.A Relay Chatter and Locking Circuits

Fragilities for electrical components represent a special problem due to the wide variety of electrical gear found within a plant. Relay chatter and inadvertent trip of circuit breakers are potentially the weakest failure modes in terms of fragilities. Relay chatter is the weakest failure mode and, if included in a risk analysis, could be be the dominant failure in seismic sequences. Because, in most cases, chatter of relays would not cause a change in the state of a system being controlled, past PRAs assumed that relay chatter was not a problem, and included only circuit breaker trip as the failure mode for electrical gear. Before continuing to make this assumption, however, one should carefully investigate whether or not there are certain locking circuits within the plant for which momentary chatter of a relay could cause changes in the configuration of the safety systems. If these conditions exist, core melt frequencies could increase. A better understanding of the effects of relay chatter or inadvertent circuit breaker trip on the reactor protection system can help reduce this uncertainty.

S.I.D Ductility Effects in Structural Failures

Local structural failures have been found to dominate most of the seismic PRAs to date. In each case, in predicting the failure of these structures, consideration is taken of the ductility available to absorb ground shaking energy. This ductility factor is used in all the fragilities of the structures as well as the fragilities of the major components and pipes to account for nonlinear energy absorption. Examination of the fragilities shows that the ductility factor has a strong effect on the final median value of failure in each case. Thus, the failure probabilities are quite sensitive to the assumed value of the ductility, and uncertainties in the computed core melt frequencies are large. Although ductility is a widely used concept in failure-prediction methodologies, its background is based on only a limited number of studies by Newmark and his co-workers (6). Most of these studies dealt with single-degree-of-freedom systems.

S.I.G Aging Effects on Seismic Fragilities

Another limitation that has not been considered is aging effects. Aging effects on fragilities could be significant, and could increase core melt frequencies. We are not aware of any data on how the fragility of nuclear components due to seismic excitation changes as equipment ages. Equipment testing could aid in our understanding of this issue.

S.I.H Correlation of Component Fragilities

Although seismic failure of a component is a random event, seismic failures of like components could be highly correlated. This would imply that the like components (experiencing the same base excitation) could tend to fail together, so that the failures of two like components would not be statistically independent. Thus, the probability of failure of the two like components could be much higher than the joint failure probabilities if the component failures were independent.

Correlation between fragilities of components in the same generic category has been shown to be important. It appears that there are no existing data concerning the question of correlated fragilities. Indeed, this is an area that can only be examined experimentally. Fragility correlation (or lack thereof) might increase the seismic core melt probability, and thus, represents a large uncertainty.

S.IV.A System Response - Local Amplification of Ground Motion

At sites with 60-150 feet of soft soil over bedrock, significant amplification of the earthquake motion (over that seen at a nearby rock site) is observed. This local site amplification results in a higher earthquake hazard curve and larger ground accelerations input to the plant foundation (increased by up to 200%).

The local site amplification is a strong function of the soil depth and soil properties, as well as the change in soil properties with the large strains due to large earthquakes. In addition, sites usually have sloping bedrock and soil strata which affects the amplification.

At least 30% of the US power plants are affected to some extent by such local site effects. Research is needed to develop consistent and accurate methods for including these local site effects in seismic PRAs.

2.2 Containment and Consequence Related Issues

In this section important containment/consequence issues affecting risk are discussed. The basis for determining the importance of these issues is discussed in Chapter 4. A complete set of containment/consequence risk issues is provided in Appendix B. The issues discussed in this section are those that are perceived to directly affect risk. Other areas considered to be very important to our capability to understand risk and its associated uncertainty, such as code development and analysis, are identified in Appendix B, but are not discussed here.

R.1.a In-vessel Issues - Accident Progression - Natural Convection

Natural convection in the primary reactor system affects a wide range of phenomenological and risk issues. The type of natural convection considered here is the convection of gases and suspended aerosols between the time the core is uncovered and vessel breach. Natural convection controls the transport of mass (including fission products) and energy around the primary system, and consequently, the heatup of primary system components and the cooling of the core. Heatup of the primary system could lead to pressure-boundary failure at various locations, and therefore depressurization, prior to meltthrough of the lower head. This depressurization could make direct heating much less likely and reduce the probability of early containment failure for some scenarios, particularly in PWRs. Alternatively, the high-temperature failure of steam generator tubes or isolation valves could provide early fission product release paths, and thus, lead to increased risk.

In addition to the effects of natural convection on the pressure boundary, gas transport in the core region is very important. Gas flow patterns through the core will control the hydrogen production rate and total amount of hydrogen produced. The rate and amount of hydrogen release will control the threat to containment for many scenarios. The concerns discussed above have led us to conclude that this issue contributes significantly to the uncertainty in risk. There is also a potential for the estimated risk to increase significantly if unfavorable primary system failure modes leading to high fission product releases, direct heating, or large and rapid hydrogen releases are determined to occur frequently.

R.1.d In-vessel Issues - Steam Explosion-Induced Containment Failure

This issue deals with the direct failure of containment due to a large in-vessel steam explosion (alpha-mode failure). Containment failure could occur if a large fraction of the core explosively interacted with water in the lower plenum of the reactor vessel, producing a slug of water and debris that propelled the vessel head into the containment structure. The probability of such an event is considered to be low, but the potential consequences are high because the event would involve direct ejection of core material into the atmosphere. Current data are limited to relatively small scales, thus leaving significant uncertainties in the potential for such an occurrence.

There are a variety of subissues that are important in understanding the risk from alpha-mode failures. For example, the manner in which the fuel melts and falls into the lower plenum, the conditions within the vessel, and the configuration of structures within the vessel provide important initial and boundary conditions. Phenomenology dealing with mixing of fuel and coolant and conversion ratios is largely unknown for large scales. The potential for and risk from multiple explosions, with a small explosion triggering a larger one, or explosions due to injecting coolant on top of the melt are also highly uncertain in large-scale interactions.

R.1.f In-vessel Issues - Alternate Primary System Failures

As discussed above for issue R.1.a, the location and type of primary system failure can control the potential for direct heating, the fission product release path, and the location and timing of hydrogen release into containment. The ability of the operator to subsequently terminate the accident may also be affected. There are basically three failure mechanisms considered here: 1) pressure-induced, 2) temperature-induced, and 3) failure induced by direct attack of debris. Failures from operator actions, corrosion, thermal shock, etc., are considered as part of other risk issues.

For many scenarios, pressure-induced failures will be prevented by the opening of relief valves. However, overpressure may still occur for some rapid pressure transients, such as small- to intermediate-scale fuel-coolant interactions. Dynamic, as well as static, pressure loads are important. Pressure effects are also important in combination with thermal loads. Temperature-induced loads relate directly to issue R.1.a above. Given hot steam, hydrogen, and fission products being transported around the primary system at degraded-core temperatures, induced failures are certainly plausible. Direct attack by core debris has previously been considered for the lower head. However, transport of debris around the primary system is also important. Debris can be transported from the core and lower plenum regions either by entrainment in coolant fluid or by ejection from fuel-coolant interactions.

The plant and scenario specific nature of this problem will make final quantitative resolution of all possibilities difficult. It may be reasonable to lump the possibilities into groups of subissues that can be treated together.

R.1.g In-vessel Issues - Fuel Melt Progression

Fuel melt progression deals with the core degradation process through the point where most of the core has slumped into the lower plenum. Fuel melt progression provides the boundary conditions for many other issues, such as hydrogen production, fission product release, fuel-coolant interactions, etc. Currently, fuel melt progression is not well understood and is usually treated parametrically, with conservative treatments possible, depending on the issue being considered.

Existing data are limited to experiments involving limited numbers of fuel pins and, generally, small scales. Further, a relatively small set of accident conditions has been considered.

Significant uncertainties exist with regard to such things as heat transfer, fission product behavior, channel blockage, hydrogen production, and mode and timing of slumping.

R.1.h In-vessel Issues - Debris Transport and Interactions with Primary System Structures

This issue deals with the location and mode of primary system failure. This issue is different from Issue R.1.f, dealing with alternate primary system failures, in that it is confined to core-debris-induced failures and includes the normally-considered failure location, i.e., the lower head. The manner in which the lower head fails will influence the likelihood that direct heating will occur. Uncertainties exist regarding the time required to cause failure, the precise failure location, the size of the opening, and the rate of growth of the opening. Least desirable are failure modes that result in rapid corium release at high pressures. Other concerns consider the possibility that debris transported around the primary system due, for example, to fuel-coolant interactions could cause primary system failures in undesirable locations, such as steam generator tubes.

R.2.d Ex-Vessel Issues - Gas Transport

The major area of concern for this issue is the transport of combustible gases throughout the containment and the potential for formation of combustible mixtures, with detonable mixtures being of particular concern. The formation of such mixtures has been identified as a potential problem in ice condensers, and a detonation is one of the few ways to cause early failure of a large-dry PWR containment or a Mark III BWR drywell. An additional area of uncertainty involves the transport of radionuclides around containment and to potential leak locations.

The importance of local hydrogen detonations hinges on the resolution of this issue. If detonable concentrations of sufficient magnitude to threaten containment are shown to be likely, then the importance of research to study the phenomenology of hydrogen detonations will increase significantly. If containment atmospheres are generally well mixed, then only global detonations need be considered.

The uncertainties in this issue are large because of the complex geometries within containment and because of uncertainties in the phenomenology and systems behavior. The latter areas include such things as condensation effects, spray-induced turbulence, break-flow momentum effects, the effects of fans and fan coolers, etc. While some calculations have been performed in this area, large calculational mesh sizes and simplifying assumptions are always used, leaving the results open to question. Code development and application, as well as additional experimental efforts, could assist in the resolution of this issue. Issues related to fission product transport and equipment response may require more accurate analyses.

R.2.f Ex-vessel Issues - Flame Acceleration and Detonations

For hydrogen concentrations in containment above about 13%, either locally or globally, detonations become possible. Since hydrogen production begins during the core degradation phase, early threats to containment are possible. Concerns have already been identified for ice-condenser containments. Detonations may be one of the few ways to fail large-dry containments or BWR Mark III drywells for many scenarios. This issue will increase or decrease in importance, depending on the resolution of issue R.2.d, dealing with gas transport. There are significant uncertainties regarding the likelihood of a detonation, given that detonable mixtures exist. In most cases, there will not be ignition sources present that are sufficiently strong to directly initiate a detonation.

However, a deflagration can undergo a transition to a detonation under certain conditions. While these conditions are not well understood, turbulence due to obstacles and fans is known to be capable of producing such a transition.

Uncertainty also exists in the response of containment and equipment to detonations. The complexity of containment geometries and the uncertainty in ignition location result in significant uncertainties in predicted shock wave behavior.

R.2.m Ex-Vessel Issues - Direct Heating

If vessel failure occurs at high pressure, the high velocity steam-hydrogen gas stream which would follow the melt out of the vessel and into containment could entrain the molten debris in the cavity region, and transport it into the larger containment volume. This scenario was first proposed in the Zion Probabilistic Safety Study. Recent experiments at Sandia and Argonne indicate that such a process can occur with reasonable efficiency and, moreover, that significant fragmentation of the melt occurs in the process. An issue of recent concern is the possibility that if such finely fragmented debris is thrown into the containment building atmosphere, it could give up its heat directly to the gas, resulting in far higher pressurization than a steam spike with the same amount of heat transferred. However, more important for the direct heating scenario is the fact that the unoxidized metal in the debris can react chemically with the oxygen and/or steam in the atmosphere, resulting in a significant increase in the total heat transferred. If a significant fraction of the core mass participates in such a combined chemical reaction/heat transfer process, the calculated loads far exceed the failure pressures of any containment.

There are a number of uncertainties in the phenomenology involved in the direct heating issue. It is only recently that the scenario has been identified as a concern. Six questions important to our understanding of this issue are:

1. Is it likely that the vessel will fail at high pressure? This is currently a hotly-contested issue, which is discussed in more detail under issue R.1.f.

2. How much of the core is released at vessel failure time? It appears that experimental tests are unlikely to resolve this question. Advanced melt progression codes may be more useful.

3. How much melt is entrained and transported from the cavity? The processes are complex and, in experiments at UK Winfrith, Argonne and Sandia, cavity geometry effects have been shown to be crucial. However, results to date indicate that substantial ejection (near 100%) is possible for some cavity geometries.

4. How much of the ejected debris can be further transported to the large gas volume regions in containment? This issue is considered by many analysts to be the critical question. Unless the gas-debris mixture can traverse the often tortuous path to the open regions in the containment, oxygen starvation will be the limiting factor. The transport process is complex since it may involve curved trajectories (due to the curved gas streamlines) and also "bouncing" of droplets, or the re-entrainment of deposited debris.

5. What is the characteristic particle size of the ejected debris-gas jet? This size determines the rates at which oxidation and heat transfer occurs, and can be important if the residence time of the particles before they drop out of the atmosphere is short.

6. How effective is oxygen transport into the plume? Assuming the answers to the previous questions result in a substantial plume of finely particulated hot debris, there is a possibility that the interior of the plumes will be oxygen-starved, limiting the total chemical reaction.

R.3.a Release of Fission Products from Fuel (In-Vessel)

The timing of release of fission products from fuel can be as important as the total quantity released. Fission products released in-vessel may be much more likely to be retained on surfaces in the near neighborhood of release than those released ex-vessel. Although the BMI-2104 and IDCOR analyses indicate that most of the volatile fission products (noble gases, iodine, and cesium) would be released in-vessel, the QUEST analyses indicate that substantial uncertainty exists even for these fission product groups. Indeed, results of the PBF in-pile experiments indicate that the release rate of volatile fission products may be substantially lower than the BMI-2104 empirical models predict. Uncertainties in the release rates of the less volatile fission products are greater than for the volatiles. Some of the principal contributing uncertainties are: the time-temperature history of the fuel, release mechanisms (e.g., by steam oxidation of the fuel), fission product chemistry, potential for reaction with cladding (e.g., tellurium), and rate limiting mass transport steps.

R.3.c Transport and Deposition within Primary System

One of the major differences between WASH-1400 analyses and more advanced source term analyses (BMI-2104, IDCOR) is the credit taken for the deposition of vapors and aerosols within the reactor coolant system. The methods of analysis used for RCS transport are highly uncertain, however, and to date have very limited experimental validation. The principal sources of uncertainty divide into three groups: fission product chemistry (including reactions with surfaces), aerosol transport behavior, and thermal-hydraulic behavior. The issues associated with RCS transport and deposition are closely coupled with other difficult issues including R.1.a. Natural Convection (In-Vessel), R.1.g. Fuel Melt Progression, R.3.a. Release of Fission Products from Fuel, R.3.b. Aerosolization of Inert Materials and R.3.d. Chemical Transformations of Fission Products Deposited In-Vessel.

R.3.d Chemical Transformations of Fission Products within the Primary System

The chemical forms of fission products within the primary system affect how they will evolve, transport, condense, and react with structural and aerosol surfaces. It is recognized that a particular element (such as iodine) will be present within the primary system in a variety of chemical forms determined by the temperature, hydrogen-oxygen ratio, mix of elements present, and kinetic behavior. In practice, this extremely complex problem has been treated by examining simplified systems involving a few elements using separate chemical thermodynamic analyses to identify a predominant chemical species (e.g., CsI for iodine). The single species is assumed to characterize the transport and deposition behavior of the element. Significant uncertainties exist, however, as to the adequacy of the simplified systems analyzed, the potential for reaction with control materials (e.g., B_4C) and the effect of high radiation levels.

R.3.g In-Vessel Fission Products - Revolatilization of Fission Products In-Vessel

In many of the severe accident sequences analyzed in the Source Term Reassessment Study, large fractions (in some cases >85%) of the core inventory of volatile fission products are predicted to be retained on surfaces of the reactor coolant system. Associated with these fission products is a significant component of the decay heat of the fuel (~20% which is equivalent to approximately 20 MW during the time frame of an accident). Depending on how the fission products are distributed in the RCS and how the surfaces are cooled, this quantity of heat is capable of heating these surfaces to temperatures at which the fission products will be reevolved or the structures will melt. This issue is considered to be particularly important because of the potential for under-estimation of the environmental source term of fission products using existing methods. Not only can the primary system retention factor be altered by this issue (e.g., changed from 0.15 to 1.0 in the case of total revolatilization in a sequence where the initial deposition involves 85% of the core inventory) but the timing and character of release to containment can be changed. A delayed release to containment is more likely to occur close to the time of containment failure or subsequent to containment failure thus reducing the containment retention factor. The conditions in the primary system can also be substantially different at the time of revolatilization than during initial period of RCS transport. As a result, fission products released to the containment in the revolatilization phase are more likely to appear in vapor form or as smaller (more persistent) aerosols. Air ingress to the vessel can also lead to the oxidation of fission products and the release of different chemical forms.

A simple treatment of revolatilization has been included in the IDCOR study. The results indicated a significant potential for revolatilization in some plant designs and sequences. Considerable uncertainties exist in the treatment of volatilization, however. The most fundamental uncertainties are associated with the sparse data base that exists regarding the reaction of fission product species with RCS surfaces and the contaminants on these surfaces, and the potential for volatilizing fission product species when the surfaces are heated. The other major area of uncertainty involves the behavior of buoyancy driven flow patterns in the RCS both prior to meltthrough of the lower head and subsequent to meltthrough. These flow patterns will determine how hot the RCS surfaces will become, how fission products will be redistributed within the RCS and whether they will be convected out of the RCS. The amount of heat loss from the RCS is also an important source of uncertainty. The extent of degradation of RCS insulation material in the accident environment and (for BWRs) the gas temperature in the drywell are contributing uncertainties.

This issue is closely coupled to a number of other issues that have been identified, in particular: Issue R.3.c - Transport and Deposition within Primary System; Issue R.1.a - Natural Convection (In-Vessel); and Issue R.2.b - Radiant Heating of the Atmosphere/Concrete Above the Melt.

R.4.c Ex-Vessel Fission Products - Release from Core-Concrete Interactions

After the molten core penetrates the lower head of the vessel and begins to attack concrete, fission products may be released from the melt either by mechanical means, as bubbles break the surface of the melt, or by the vaporization of fission products into the gases as they sparge through the melt. The latter process is believed to have the greater effect and be the one that can potentially lead to large releases. Release of fission products during core-concrete attack is potentially important because the period of release may extend close to or beyond the time of containment failure. Furthermore, it is less likely that material released during this stage will be deposited in the near neighborhood of the release point, as in the case of in-vessel release. As gases released from the concrete (water and carbon dioxide) pass through the melt, chemical reactions occur that can change the chemical forms of the fission products and, in some cases, make them more volatile (e.g., volatile oxides and hydroxides). In some of the sequences analyzed in the BMI-2104 study, the releases of the strontium and lanthanide groups are predicted to be substantially higher than in WASH-1400. The QUEST results also indicate ex-vessel releases to be a major area of uncertainty.

The uncertainties in ex-vessel releases begin with uncertainties in the initial conditions for core-concrete attack. The fractions of initial core inventory of the volatile fission products remaining with the melt, the oxidation state of the melt, the mass of molten material and the initial temperature of the melt can each have a major influence on the subsequent release.

The mechanics of core-concrete attack also affect the predicted release. The time-temperature history of the core debris is particularly important to the release of the less volatile species. Uncertainties about the rate of concrete attack, mass flow rate of sparging gases, degree of separation of oxidic and metallic layers, and the freezing behavior of the melt are large. State-of-the-art computer codes have had very limited success in predicting the details of core-concrete attack as simulated in experimental programs at Sandia and at KfK in West Germany. Indeed, there are significant discrepancies in the observed quantities of aerosols produced in core-concrete experiments at the two laboratories that have not been completely resolved.

Finally, the chemical kinetics of the reactions occurring among the wide variety of elements in the molten pool cannot be predicted accurately. In addition to determining the concentrations of important species in the pool, it is also important to predict the transport of these species into the bubbles passing through the pool.

R.5.e Health and Economic Consequences - Modeling of Emergency Response

For accidents in which significant offsite releases of fission products occur, emergency response will largely determine the early consequences. Current treatments of this issue are generally conservative, and some progress could be achieved in this area; however, our ability to model emergency response is limited due to the complex institutional and social issues involved. Prediction, for example, of evacuation times depends on the decision-making process, the response of the population to the evacuation instructions, and other factors, such as weather and road conditions. This issue is also affected by Issues R.6.a: Equipment Issues - Detection and Monitoring Systems and R.8A.C: Operator Training and Performance. Clearly, assumptions regarding the adequacy of information available to the operator and his ability to properly interpret that information and report appropriately to decisionmakers will influence the emergency response modeling.

R.6.a Equipment Issues - Detection and Monitoring Systems

The accident at Three Mile Island indicated the sorts of problems that can arise from failed or misleading monitoring systems, particularly when dealing with off-normal situations. For accidents that have progressed to core damage or beyond, the operator will be dealing with information and instrument readings not normally observed and, possibly, with failed or misleading instrumentation. Proper termination of a severe accident and emergency response instructions will depend on the operator's ability to obtain appropriate data concerning the state of the plant. Many severe accidents will result in plant environments more severe than considered as a design basis.

The functionability of this equipment in such environments is largely unknown. Additionally, the overall response of monitoring systems is important. There is uncertainty in how the operator can or should correlate the information from a variety of sources, any one or all of which may be suspect.

R.6.c Equipment Issues - Essential Equipment Performance During Severe Accidents

The functionability of safety grade equipment in environments which exceed the qualification environment and of non-safety grade equipment during severe accident conditions is largely unknown. The lack of this information creates two problems. First, probabilistic risk assessments (PRAs) currently assume such equipment will function properly, which could lead to nonconservative predictions of risk. Second, accident management and emergency response procedures to mitigate the consequences and minimize the risk of a severe accident cannot be devised with any degree of certainty.

The equipment considered in this issue is that equipment needed to mitigate and respond to an accident that has proceeded to core damage or beyond. This equipment will be contained in systems to provide containment heat removal, containment isolation, power, control room environment, etc. A variety of pressure, temperature, humid, and corrosive environments may be encountered during severe accidents. Phenomena affecting the environment include hydrogen burns, diffusion flames, high temperature steam, direct heating and debris and aerosol collection. These phenomena can readily produce environments more severe than those normally considered as a design basis.

R.7.f Containment Performance - Response to Static Overpressurization and Increased Temperatures

The containment is the final engineered barrier preventing the off-site release of fission products during a severe accident. The point at which containment integrity is compromised, the amount of leakage before failure, and the failure mechanism are very important in determining the off-site consequences. Our current understanding of these issues is very limited. Containments are designed to the American Society of Mechanical Engineers (ASME) code and the American Concrete Institute (ACI) code. These codes are based on essentially elastic models, precluding extrapolation to determine the containment ultimate capacity. A limited set of experiments has been performed to examine these issues, but the data base is very small at this time.

The response of concrete containments is different from the response of free-standing steel containments. It is possible that cracks can propagate through the shell in steel containments, resulting in very large leak paths. For very rapid decompression of containment, resuspension of fission products inside containment may be important. Also of importance are the responses of containment structures and penetrations to high pressures accompanied by high temperatures. Computer codes exist that can examine the response of structures to various loadings; however, there are no codes that can adequately factor in the response of all containment penetrations and provide a complete analysis of the response to all possible environments. Some uncertainties will always remain in this area, particularly with regard to the quality of construction and leak testing.

R.8A.c Operations - Operator Training and Performance

Operator performance is a significant contributor to risk and a key ingredient of accident and emergency management during severe accidents. Currently operators receive little training in how to handle severe accident mitigation. Mitigation is complicated by our current uncertainty in scenario progression, but simplified by the time available to take action and the relatively few number of actions which can be taken. In addition to potential in-plant mitigation of the accident, the operator is important to offsite response, including evacuation. The operator should be able to correctly determine conditions within the plant and, if possible, be cognizant of the possible outcomes of the accident.

3. SYSTEMS RELATED ISSUES

3.1 Overview of Tables in Appendix A

Appendix A contains a list of issues dealing with the system unavailability portion of the risk equation for light water reactors. The contribution of each issue to the uncertainty in total core melt frequency is subjectively evaluated and presented in Tables A-1, A-2, A-3, and A-4. As can be seen in the tables, the resolution of an issue may reduce the total uncertainty in core melt frequency, and furthermore may either raise or lower our estimate of the core melt frequency.

Each table consists of six columns:

1. The first column lists the issues that impact uncertainty in core melt frequency.

2. In the second column, the effect that an issue can have on the overall uncertainty in core melt frequency is rated as large, medium, or small. This effect was assessed by estimating the maximum impact that the issue could have on the upper and lower core melt frequency bounds. (See Section 3.2 for a more detailed description.)

3. In the third column, the potential for research resolving this issue is rated as large, medium, or small. For example, for issues receiving a large rating, it is believed that further research has a large or significant potential for reducing the uncertainty currently associated with the issue.

4. In the fourth column, the potential for each issue to decrease the core melt frequency is rated as large, medium, small, or none. The impact on the lower bound of core melt frequency was used to assess this potential. (See Section 3.2 for a more detailed description.)*

5. In the fifth column, the potential for each issue to increase the core melt frequency is rated as large, medium, small, or none. The impact on the upper bound of core melt frequency was used to assess this potential. (See Section 3.2 for a more detailed description.)*

* It should be noted that the expected value of the core-melt frequency or any other best estimate value may not change necessarily when the lower or upper bounds change.

6. The sixth column includes comments and discussion. In this column, the rationale for assigning large, medium, small, or none in Columns 2, 3, and 4 is given. Also, if clarification of an issue is needed, it is provided in this column--usually by use of an example.

Ratings were assigned on the basis of engineering judgment, past experience, and information provided by the PRA catalog developed by NRC's Accident Sequence Evaluation Program (ASEP). The ASEP PRA catalog was used to compile a list of sequences from past PRAs to help identify issues that were important to the uncertainty in core melt frequency caused by internal events. This list is presented in Appendix C.

Although all judgements were subjective, the analysts attempted to assign "large" to those uncertainties that could affect either core melt frequency bound by greater than an order of magnitude, "medium" to those uncertainties that could affect either bound by a factor less than an order of magnitude, but greater than a factor of two, and "small" to those uncertainties which could affect either bound by less than a factor of two.

It must be stressed that these tables represent the perspective of a small group of analysts. Not all perspectives on risk are included. Any user of these tables may want to consider other sources of information that approach solutions to the risk problem from other perspectives.

3.2 Discussion of Uncertainty Definitions

This section describes what the ratings in columns 2, 4, and 5 of the Appendix A tables represent.

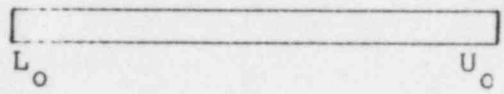
The effect an issue can have on the uncertainty in core melt frequency is the effect that the issue is believed to have on the upper bound or lower bound which is associated with the core melt frequency. A systematic, quantitative measure of the effect is obtained by taking the maximum of the issue's believed impact on the upper bound and lower bound for the core melt frequency.

Let U and L be the upper bound and lower bound, respectively, for the core melt frequency with the issue unresolved (i.e., with the uncertainty for that issue included). Let U_o and L_o be the upper and lower bounds with the issue completely resolved (i.e., no uncertainty due to that issue). In terms of ratio changes, the effect (E) of the issue on the core melt frequency is then:

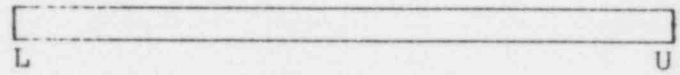
$$E = \max\left(\frac{U}{U_o}, \frac{L_o}{L}\right)$$

where "max" denotes the maximum is to be taken. The ratios are defined such that they are greater than 1 if the issue increases the upper bound or decreases the lower bound. An illustration of the above measure is shown below:

Uncertainty range on the core melt frequency with the issue resolved



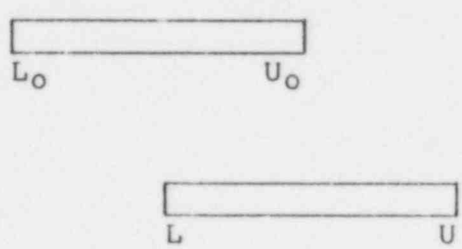
Uncertainty range on the core melt frequency with the issue unresolved



Effect of this issue

$$\max\left(\frac{U}{U_o}, \frac{L_o}{L}\right)$$

The above definition of the effect on an issue is quite general and is applicable to any type of situation. For example, if the issue can only increase the core melt frequency then both the upper bound and lower bound will be increased. This translation of the uncertainty range is shown below:



In this translation case, L_o/L will be less than 1 since $L > L_o$.

The above measure is not tied to any specific statistical framework. It is applicable to either a Bayesian or classical interpretation.* For the present, the effects will be estimated as being "small," "medium," or "large," depending upon the

*For example, a Bayesian viewpoint may interpret (L, U) and (L_o, U_o) as approximate 5 percent and 95 percent bounds. A classical viewpoint may interpret (L, U) and (L_o, U_o) as end points of plausible core-melt frequency ranges.

sizes of the ratio changes. The numerical interpretation for these categories was discussed in Section 3.1.

To provide more information on the impacts of an issue, the separate ratios U/U_0 and L_0/L are also given. The upside ratio U/U_0 (column 5) is the potential for increasing the core-melt frequency. The downside ratio L_0/L is the potential for decreasing the core melt frequency. These potentials are given as "small," "medium," or "large." If there is no increase in the ratio (such as in the downside ratio in the translation case), then "none" is given.

Table 3-1 Summary of Uncertainty Definitions

(2nd Column)	Effect an issue can have on the uncertainty in core melt frequency	=	Maximum of the impacts on the upper and lower bounds for the core melt frequency
(4th Column)	Potential for this issue to decrease the core melt frequency	=	The impact on the lower bound for the core melt frequency
(5th Column)	Potential for this issue to increase the core melt frequency	=	The impact on the upper bound for the core melt frequency

4. CONTAINMENT AND CONSEQUENCE-RELATED ISSUES

As stated earlier, this assessment has been divided into research topics associated with the "front end" and the "back end" of probabilistic risk assessment (PRA). Front end topics relate to the likelihood and definition of accident sequences, and back end topics refer to the subsequent in-plant accident progression and ex-plant consequences with the dividing line being the onset of core degradation. This section describes approach used to assess issues related to back end analyses. It should be noted that in the division that has been made between front end and back end analyses, success criteria are considered as front end issues. Thus, many issues related to the effectiveness of emergency core cooling systems are evaluated as front end issues (see Appendix A). The issues identified as being containment and consequence-related are listed in Appendix B.

The assessment of containment/consequence issues is based on insights developed in various NRC and industry programs. <u>It should be recognized, however, that the assessments are subjective and that they do not necessarily represent a consensus of any particular group</u>.

The first step in the assessment was to develop a list of issues to evaluate and to categorize the list appropriately. An attempt was made to categorize the issues according to the following research areas:

R.1. In-Vessel Issues - Accident Progression

R.2. Ex-Vessel Issues - Containment Loads

R.3. Fission Product and Aerosol Release from Fuel and Transport in Primary System

R.4. Fission Product and Aerosol Generation and Transport Ex-Vessel

R.5. Health and Economic Consequences

R.6 Equipment Functionability and Vulnerability under Severe Accident Conditions

R.7. Containment Performance

R.8. Accident Management

R.9. Severe Accident Analysis Tools

R.10. Safety, Risk, and Application Studies.

Under each heading a number of issues are listed. The starting point for developing a list of issues was the NRC/IDCOR list of Severe Accident Issues, which has been the focus for discussions between the NRC and IDCOR. This list was augmented to provide better coverage of the issues. Even at this level of detail, many of the issues that are identified are quite general and may more appropriately be considered research areas than specific issues. Typically, in order to identify specific issues, it is necessary to go to another level of detail, which was not practical for this exercise. Notes that identify some of the considerations underlying the ratings have been provided in Appendix B for each issue area. These notes often indicate the specific issues that were of principal concern in the rating of an issue area, but in no way should be considered a comprehensive or consistent treatment of the issues. When going through the results, the reader will note that all issues have not been broken down to a consistent level, i.e., some issues are much broader in scope that others. It was not possible in the time available (and not necessarily desirable) to achieve across-the-board consistency.

In developing an approach to rating the issues it was recognized that a number of different attributes would be of interest to NRC decision-makers. Further, the attributes should be a function of the nature of the research area and would probably differ from the attributes used in the evaluation of front end issues. In all cases the attributes were assessed on the basis of a three-level rating scheme: large (L), medium (M), and small (S). These ratings are subjective; no precise algorithm was constructed for assigning ratings. In some cases, several issues are interrelated, with some issues providing boundary conditions affecting other issues. For example, direct heating and steam explosions are affected by several different in-vessel issues. Generally, we have not rated issues which are "links in a chain" as being as important as the dominant question unless these issues are links in several important chains or are exceptionally critical to a particular chain of events. These interrelationships are very important, however, and groups of issues will rise and fall in importance together as the dominant question rises and falls in importance. With the above guidelines in mind, we have defined the ranking levels for each attribute as discussed below.

The first five areas involve phenomenology, which, with some exceptions, could impact the consequences of almost any severe accident sequence. These issues were rated according to the following three attributes:

Attribute I: To what extent do the uncertainties in this area contribute to the overall uncertainty in risk based on conventional wisdom?

Large: These issues contribute significantly (potentially an order of magnitude or more) to the uncertainty in risk. Generally, these issues involve early containment failure and/or high fission product releases.

Medium: These issues are also important contributors to the uncertainty in risk, but are not as dominant as those rated "large". If an issue does not affect early fission product releases or massive resuspension of fission products for late releases, its maximum rating is medium.

Small: These issues are not major contributors to the uncertainty in risk (for example, less than a factor of 2).

Attribute II: What is the likelihood that near-term research in this area could be successful in reducing the associated uncertainty significantly?

Near-term research is defined as covering a time frame of 2 to 3 years and is not restricted to projects currently planned. However, the research effort required has to be of reasonable scope.

Large: There is a significant potential for reducing the uncertainty associated with this issue with near term research. A large rating does not necessarily imply a significant effect on the uncertainty in overall risk, but merely that the uncertainty related to this issue can be reduced.

Medium: Some knowledge can be gained regarding this issue, but easy research may have already been completed, and/or uncertainties are likely to remain.

Small: The remaining research is very complex and/or unlikely to substantially reduce the uncertainty.

Attribute III: Is there a potential that the resolution of this issue could result in a large increase in estimated risk?

This attribute is particularly important to the risk averse decision-maker. It focuses on the upside of the uncertainty in risk.

Large: If conventional wisdom is incorrect, current central estimates of risk may prove to be significantly nonconservative.

Medium: Current estimates of risk could be nonconservative due to these issues, but the upward potential is not as large as for those issues rated "large". If an issue does not affect early fission product releases or massive resuspension of fission products for late releases, its maximum rating is medium.

Small: These issues could not increase current risk estimates significantly. In some cases this is an issue which we are currently treating conservatively.

Research areas R.6 and R.7 are somewhat different than the first five areas. Research area R.6, dealing with equipment, includes phenomenology, but also includes such concerns as design adequacy, qualification, installation, and maintenance. Both research areas R.6 and R.7 include many issues that are more sequence dependent. For example, the importance of the containment response to tornado-generated missiles is dependent on the probability that a tornado will occur. With the above differences in mind, research areas R.6 and R.7 were evaluated according to the same three attributes as research areas R.1 through R.5.

Research area R.8 is divided into two parts: R.8A) Operations, which deals with current perceptions of information needs, procedures, and operator performance and R.8B) Accident Management, which deals with research to address the problems of R.8A and thereby reduce risk. Area R.8A is evaluated according to the previous three attributes. The issues up through R.8A are similar in that they generally impact risk directly. All of these issues were evaluated according to the Attributes I through III. The remaining issues deal with methods for understanding risk or actions to reduce risk and were evaluated according to different attributes. Only the issues through R.8A were considered for inclusion in Chapter 2.

Research area R.8B, accident management, is primarily motivated by the intent to respond to severe accidents and directly reduce risk. For this reason, R.8B was evaluated according to Attribute II and the following fourth attribute.

Attribute IV: What is the likelihood that the product of research when implemented would result in a significant reduction in risk?

Note that while the first three attributes deal with improved understanding of risk, this attribute deals with an actual change in risk (at least as perceived by the assessment team).

Large: Risk could be significantly reduced due to actions taken. For example, containment failure could be prevented or a population could be safely evacuated before the off-site release.

Medium: Reductions in risk are possible. For example, operation of containment systems could reduce the fission product release or prevent late containment failures.

Small: No significant risk reduction envisioned.

Although there is a parallel research area to the accident management area which would develop and evaluate improved hardware, e.g., vent-filter systems, and core catchers, this area was intentionally omitted. These needs are important but cannot be adequately addressed until more information is available and additional research is completed in the other areas.

Research area R.9 covers integrated computer codes being developed for the analysis of severe accident sequences. Here, "integrated" implies that several phenomena or systems are modeled by the same code. The needs for detailed models for a particular phenomena or system are addressed with each individual research issue. The integrated codes generally consist of a framework for combining the individual models. The attribute considered here is:

Attribute V: How much additional code development is required to permit a practical and adequate quantification of risk and its associated uncertainty?

In evaluating what is meant by "required", consideration was given to the ultimate needs of NRC-NRR, accuracy goals in predicting severe accident behavior, and inherent irreducible uncertainties in the prediction of severe accident behavior and risk. In some cases additional model development is needed, for example for direct heating in containment, but a basic framework will exist for incorporating such models when they are completed. A small rating should not be interpreted as indicating that continued maintenance and upgrading of these codes is not required.

Large: A practical framework for performing integrated analysis does not exist.

Medium: A framework exists which permits certain analyses, but additional models need to be incorporated into the framework and/or the code or group of codes is not very practical to use.

Small: An adequate integrated framework exists for performing these analyses.

Research area R.10, safety, risk and application studies, was also felt to require a separate criterion. This is the research area in which the results of the other research are integrated and interpreted. The products of this area provide the input to regulatory implementation. The attribute considered is:

Attribute VI: To what extent is additional work required to support the mission of the NRC to protect the public from severe accidents?

Large: There are a large number of outstanding issues that need to be addressed. Also, there is a need to reassess issues as particular areas of research are completed.

Medium: A great deal has already been accomplished; however, a significant amount of analysis remains.

Small: The work is largely complete or the reduction in uncertainty is already as much as can be reasonably achieved.

5. GENERIC SAFETY ISSUES AND TOP RISK ISSUES

The table presented below indicates relationships between the top risk issues identified in Chapter 2 and the issues identified in Table II of NUREG-0933 (7). For the purposes of this report, all of the issues in Tables II and III of NUREG-0933 are referred to as "Generic Safety Issues." The identifiers for the Generic Safety Issues are mostly those used in NUREG-0933; however, the Task Action Plan items are identified by "TAP" and the New Generic Issues are identified by "NGI". Table III of NUREG-0933 identifies 506 generic safety issues. Only 237 were used for the cross cut with the risk issues. They include the issues in the columns labeled I, USI, HIGH, MEDIUM, Note 4 and Note 5. The remaining 269 issues were not considered because they are in various stages of resolution, are covered in the other issues, or are ranked as low or drop. No details concerning the particular relationships are presented; we merely show that the issues are related and that the resolution of certain risk issues could impact the resolution of related generic safety issues. Complete tables containing all of the risk issues identified in Appendices A and B are presented in Appendix D.

It is particularly important to note that the lists of risk issues and generic safety issues were developed from different perspectives. The perspective for the development of risk issues was discussed in the first four chapters, while the generic safety issues were developed from a variety of perspectives, which deal with such things as operations, worker safety and licensing, as well as risk. Also, the reader should recognize that both lists of issues contain issues of varying breadth. The interrelationships between a few broad issues, e.g., TAP A-45, can be just as important or more important than the interrelationships between large numbers of more narrowly defined issues.

Table 5-1. Generic Safety Issues Related to Top Risk Issues

RISK ISSUES	ISSUES FROM NUREG-0933
Internal Events	
I.A	I.F.1, II.C.1, II.C.2, II.C.4, II.D.1, II.E.6.1, II.F.5, II.K.3, TAP A-3, TAP A-4, TAP A-5, TAP A-30, TAP A-42, TAP B-55, TAP C-11, NGI-23, NGI-70
II.A	I.A.1.1, I.A.1.2, I.A.1.3, I.A.2.1(1), I.A.2.1(2), I.A.2.1(3), I.A.2.2, I.A.2.3, I.A.2.6(1), I.A.2.6(4), I.A.2.7, I.A.3.1, I.A.3.3, I.A.3.4, I.A.4.2(1), I.A.4.2(4), I.B.1.1(1), I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(5), I.B.1.1(6), I.B.1.1(7), I.C.1(1), I.C.1(2), I.C.1(3), I.C.2, I.C.3, I.C.4, I.C.5, I.C.7, I.C.9, I.D.1, I.D.2, I.D.3, I.D.4, I.D.5(5), I.E.2.2, I.E.3.1, I.G.1, I.G.2, II.C.1, II.C.2, II.C.4, II.K.1(4), II.K.1(6), II.K.1(7), II.K.3, III.A.1.1(1), III.A.1.1(2), III.A.1.2(1), III.A.1.2(2), III.A.1.2(3), TAP B-17
III.B	I.F.1, II.C.1, II.C.2, II.C.4, II.F.5, TAP A-3, TAP A-4, TAP A-5, TAP A-9, TAP A-10, TAP A-11, NGI-51, NGI-61, NGI-65, NGI-68
IV.C	II.C.1, II.C.2, II.C.4, II.E.1.1, TAP A-9
IV.D	II.C.1, II.C.2, II.E.1.1, TAP A-9, TAP A-30, TAP A-44
V.A	II.C.1, II.C.2, TAP A-3, TAP A-4, TAP A-5
V.C	II.C.1, II.C.2, II.C.4, II.E.4.2, II.k.3, TAP A-1, TAP A-2, TAP A-3, TAP A-30, TAP A-31, TAP A-43, TAP A-44, TAP A-47, TAP B-55, NGI-12, NGI-70
Fire	
F.IV.ii	TAP A-40, TAP A-41, TAP A-45, TAP A-46, NGI-57, NGI-77
F.VI.i	I.D.4, TAP A-24, TAP A-29, TAP A-45, NGI-57
F.VII.ii	I.D.1, I.D.4, II.E.3.2, II.E.3.3, II.F.5, TAP A-17, TAP A-29, TAP A-30, TAP A-45, TAP A-47, NGI-83
F.VII.iv	TAP A-40, TAP A-41, TAP A-46, NGI-57, NGI-77
Seismic	
S.I.A	II.C.1, II.C.2, II.C.3, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-56, TAP B-57, NGI-55
S.I.D	II.C.1, II.C.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, TAP A-12, TAP A-18, TAP A-22, TAP A-40, TAP A-41, TAP A-45, TAP A-46, TAP B-4, TAP B-5, TAP B-6, TAP B-50, TAP B-51

Table 5-1. Generic Safety Issues Related to Top Risk Issues (Continued)

RISK ISSUES	ISSUES FROM NUREG-0933
Seismic **(Continued)**	
S.I.G	II.C.1, II.C.2, II.C.3, II.D.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-52, TAP B-56, TAP B-57, NGI-29
S.I.H	II.C.1, II.C.2, II.C.3, II.D.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-21, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-55, TAP B-56, TAP B-57, NGI-29, NGI-55, NGI-70
S.IV.A	I.C.1, II.C.2, II.E.1.1, II.E.3.3, II.E.3.4, TAP A-12, TAP A-31, TAP A-40, TAP A-41, TAP A-45, TAP B-4, TAP B-52
Containment/Consequence	
R.1.A	II.B.1
R.1.D.	II.B.5(2)
R.1.F	II.B.1, II.B.5(2), TAP A-1
R.1.G	II.B.5(1), II.B.5(2)
R.1.H	II.B.5(2), TAP A-11
R.2.F	II.B.5(3), II.B.7, II.B.8, TAP A-48
R.2.M	II.B.1, II.B.5(2)
R.3.A	II.B.5(2), II.B.7
R.3.C	II.B.5(2)
R.2.D	II.B.1, II.B.5(3), II.B.7, II.B.8, TAP A-48, TAP B-14
R.3.D	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.G	II.A.1, II.B.5(1), II.B.8, II.H.3
R.4.C	II.A.1, II.B.5(2), II.B.8
R.5.E	II.A, III.A, IV.E.5, NGI-88
R.6.A	TAP A-2, TAP A-8, TAP A-24, TAP A-30, TAP A-34, TAP A-48, TAP B-50, TAP B-76, TAP B-85, TAP B-87, TAP B-91, TAP B-93, II.B.2, II.D.3, II.E.1.2, II.F.1, II.6.1
R.6.C	TAP A-2, TAP A-8, TAP A-24, TAP A-30, TAP A-48, TAP B-32, TAP B-50, TAP B-56, TAP B-58, TAP B-21, TAP B-41, TAP B-49, TAP B-55, TAP B-70, TAP B-71, II.B.2, II.E.3.1
R.7.F	TAP A-23, TAP B-5, TAP B-[illegible], TAP B-10, TAP B-26, TAP B-54

Table 5-1. Generic Safety Issues Related to Top Risk Issues (Continued)

RISK ISSUES	ISSUES FROM NUREG-0933
R.8A.C	I.A.21(1), I.A.2.1(2), I.A.2.1(3), I.A.2.2, I.A.2.3, I.A.2.6(1), I.A.2.6(3), I.A.2.6(4), I.A.2.7, I.A.3.1, I.A.3.3, I.A.3.4, I.A.4.2(1), I.A.4.2(4), I.B.1.1, I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(6), I.B.1.1(7), I.B.1.2(1), I.B.1.2(3), I.B.1.2(3), I.E.8, II.B.4, IV.E.5, HF.01.1.2, HF.01.1.3, HF.01.1.5, HF.01.2.1, HF.01.2.2, HF.01.3.1, HF.01.3.2, HF.01.4.2, HF.01.5.1

6. DESCRIPTION OF THE RANKING TECHNIQUE

To obtain an overall ranking for each risk issue, the scores or ratings for each issue were combined in a linear, weighted fashion with respect to defined criteria. The criteria are the sizes of uncertainties associated with the issues and the potential for research solving the issues (the column headings of the tables in Appendices A and B).

For each risk issue, the scores under the different criteria are combined to arrive at an overall score for the risk issue. The overall score for the risk issue is obtained by linearly weighting the individual scores for the different criteria.

The Lightyear program is a commercially available program for the IBM personal computer that implements this technique. Other programs could also be used to perform this analysis. To use Lightyear, four basic steps are taken:

1. Alternatives are defined which are to be ranked. The alternatives here are the risk issues.

2. Criteria are defined which are to be used for the prioritization. For the present application, the criteria can include any of the column headings on the issue uncertainties and the research potential.

3. Weights are assigned to the criteria to indicate their relative importance. The weights which were assigned are described in Chapter 7.

4. The alternatives are then ranked according to each criterion.

For a given criterion, the scoring of the alternatives can be done in various ways. For a criterion measuring a quantitative output, the score is the output value associated with an alternative. The output is measured on a linear scale from a defined minimum value (representing 0 percent) to a defined maximum value (representing 100 percent of the potential).

The scoring can also be done in a graphical manner with the score represented by a point chosen on a line from 0 to 100 percent. The relative position of the point indicates the relative score for the criterion.

The scoring is done in a categorical manner. For a given criterion, categories such as "small," "medium," and "large" can be used with a category assigned to each alternative. Each category is then translated to relative value from 0 to 100 percent. The relative values of the alternatives are then the scores of the alternatives for the given criterion. The relative values assigned to the categories are described in Chapter 7.

As part of the criteria, rules can also be defined which serve as goals or constraints. A goal rule defines a goal value for the score under a given criterion. If an alternative satisfies the goal, it is given the extra weight assigned to the goal. The constraint rules define minimum or maximum acceptable scores for a given criterion. Alternatives are eliminated if they do not satisfy the constraints.

The rules are thus quite useful for refining the criteria. Constraint rules are applied to the risk issues to sort out those which do not have at least "medium" category scores under the criteria. The rules which were applied are more fully described in the next section.

7. RESULTS OF THE ANALYSIS

The information in the tables in Appendices A and B was entered into the Lightyear computer program. Two separate computer files were made, one for the systems issues and one for the containment and consequence issues. The results of these analyses are discussed below.

7.1 Systems Analysis Issues for Research Prioritization

The analysis of the systems issues is discussed in this section. The alternatives are the issues themselves. In the figures that follow, the issue numbers are shown and not the issue names. The criteria correspond to Columns 2 through 5 in the tables in Appendix A.

For research prioritization, it is important to identify those issues that contribute significantly to the uncertainty in risk. Issues that can cause the risk to increase are also important. One of the goals of research is to reduce uncertainty. Research can also help us to better understand the source of the uncertainties. Various uncertainty contributors are candidates for resolution through research involving new experiments or the development of new analysis techniques. Other issues are best addressed through regulatory or enforcement actions.

Another important aspect of identifying issues that are important for research is the "researchability" of the issue. This corresponds to Column 3 in the tables in Appendix A. A "large" researchability does not imply that the research necessarily needs to be performed by the Office of Research, as opposed to other organizations either inside or outside NRC.

These two criteria, along with the other two columns, were input into the computer. The weights of the criteria that were assigned are shown in Figure 7-1. Research potential was weighted highest because this is a research prioritization effort. Contribution to overall uncertainty was assigned the next highest weight. Increase potential of the uncertainty was assigned the lowest weight because it is correlated with the uncertainty criterion. Decrease potential was assigned a weight of zero for consistency with the containment and consequence issues analysis.

The screening rules used for this initial identification of the important issues are shown in Figure 7-2, and they correspond to elimination rules. This means that if a criterion corresponding to an issue violates the rule, the issue is flagged. These issues are shown in the figures by the nonsolid lines. Issues that do not violate any screening rules are shown by the solid bar. The results of this initial analysis using the first two rules and the first two criteria are shown in Figure 7-3. Notice that the first issue that violates a rule is ranked 24th in the figure. Also notice that 14 issues have a total score equal to the maximum possible.

A third rule was added for the increase potential of the uncertainty. This is shown in Figure 7-4. The same weights for the criteria were used so that the rankings would remain the same. Notice in Figure 7-5 that the first issue to violate a rule is now ranked 16th. Adding the additional rule can help in cutting down the number of issues to be considered. Issues with total scores greater than or equal to 175 and that passed the screening rules were selected for discussion in Section 2.1.

Sensitivity studies were performed by changing the values of the weights and varying the rules. The results were all very consistent with those shown in Figure 7-5. For example, when the weights of the four criteria are set equal to 100 and using the rules in Figure 7-4, the 19 issues identified in Figure 7-5 were in the top 29. When the weight of the decrease potential criterion is changed from 100 to 50, the 19 issues are in the top 25 issues in the sensitivity study. Additional analyses were performed with similar results. The 19 issues identified correspond to those that have large in all three criteria or two large and a medium in one (increase potential). If the issues were binned according to their rankings for the criteria, the same issues would have been identified.

7.2 Containment and Consequence Issues Important for Research from a Risk Prospective

The containment and consequence issues were analyzed in the same manner as the systems issues. The same three criteria from Figure 7-1 were used and were given the same weights. The initial analysis considered only the uncertainty and research potential criteria which were coded as screening rules shown in Figure 7-2. The results are shown in Figure 7-6.

A second analysis was performed with the inclusion of a third rule for increase potential as shown in Figure 7-4. Adding this rule eliminates additional issues since they violate the new rule. The results of this analysis are shown in Figure 7-7. Issues 1-14 and 17-20 had scores greater than or equal to 135 and did not violate any of the screening rules and, thus, were selected for discussion in Section 2.2.

The top issues correspond to those that are ranked all large, two larges and a medium, or a large and two mediums. Sensitivity studies were run for the containment and consequence issues. The sensitivity studies changed the order of the top issues to some extent, but the ones identified were always in the top 25.

7.3 Other Applications of the Issues Data Base

The issues data base can be used to give a risk perspective to other questions. Some examples of these are the following:

1. Risk Relevance of Rules - Can some of the regulations be eliminated or relaxed from a risk perspective? Here the contribution to uncertainty should be low and the contribution to reduce risk (e.g., core melt frequency should be large).

2. Completeness of Rules - Are the important issues that are known to contribute to risk covered by rules? Here the contribution to uncertainty should also be low and the increase potential for risk (e.g., potential to increase core melt frequency) should be large.

An example of "Completeness of Rules" is given below for the containment and consequence issues. All three criteria were used with the same weights as assigned in Figure 7-1. This was done to keep the ranking the same for reader convenience. The research potential criterion could have been given a zero weight. The rules for this analysis are shown in Figure 7-8. Notice that the uncertainty rule says that the uncertainty criterion must be at most small. It can have a value of none or small. The increase rule says that the increase potential criterion must be large or medium. The results of this run are shown in Figure 7-9. The issues that are identified as satisfying these rules are the ones with the solid bars. The first one is ranked Number 45. In this analysis, ten issues are identified.

Additional analyses can be performed by changing the rules. For example, when the uncertainty rule was changed from "must be at most small" to "must be at most medium," a total of 31 issues were identified. These issues can then be evaluated by comparing them to the existing rules to see if they are covered.

Lightyear

SUBJECT: System Categor.
VERSION: March 1985

CRITERIA	MODE	WEIGHT
Uncert. Effect	V	75
Research	V	100
Decrease Pot.	V	0
Increase Pot.	V	50

Figure 7-1 Criteria and Criteria Weights

Lightyear

SUBJECT: System Categor.
VERSION: March 1985

RULE NAME	RULE
Uncertainty Rul	Uncert. Effect MUST BE AT LEAST Medium (ELIMINATION RULE)
Research Rule	Research MUST BE AT LEAST Medium (ELIMINATION RULE)

Figure 7-2 Rules for Initial Ranking

Lightyear

SUBJECT:System Categor.
VERSION:March 1985

RANK		SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
1	I.a.i		225
2	III.b.i		225
3	IV.c.i		225
4	IV.c.iii		225
5	IV.c.iv		225
6	IV.d.i		225
7	IV.d.ii		225
8	V.a		225
9	V.c.ii		225
10	F.IV.ii		225
11	F.VI.i		225
12	S.I.A		225
13	S.I.G		225
14	II.a.iv		225
15	S.I.D		200
16	I.d.i		185
17	S.I.C		185
18	L.III.A		185
19	IV.e		185
20	F.VII.ii		175
21	F.VII.iv		175
22	S.I.H		175
23	S.IV.A		175
24	V.b		165
25	III.a.ii		162

POSSIBLE = 225

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Figure 7-3 Results for the Initial Ranking for the Systems Issues

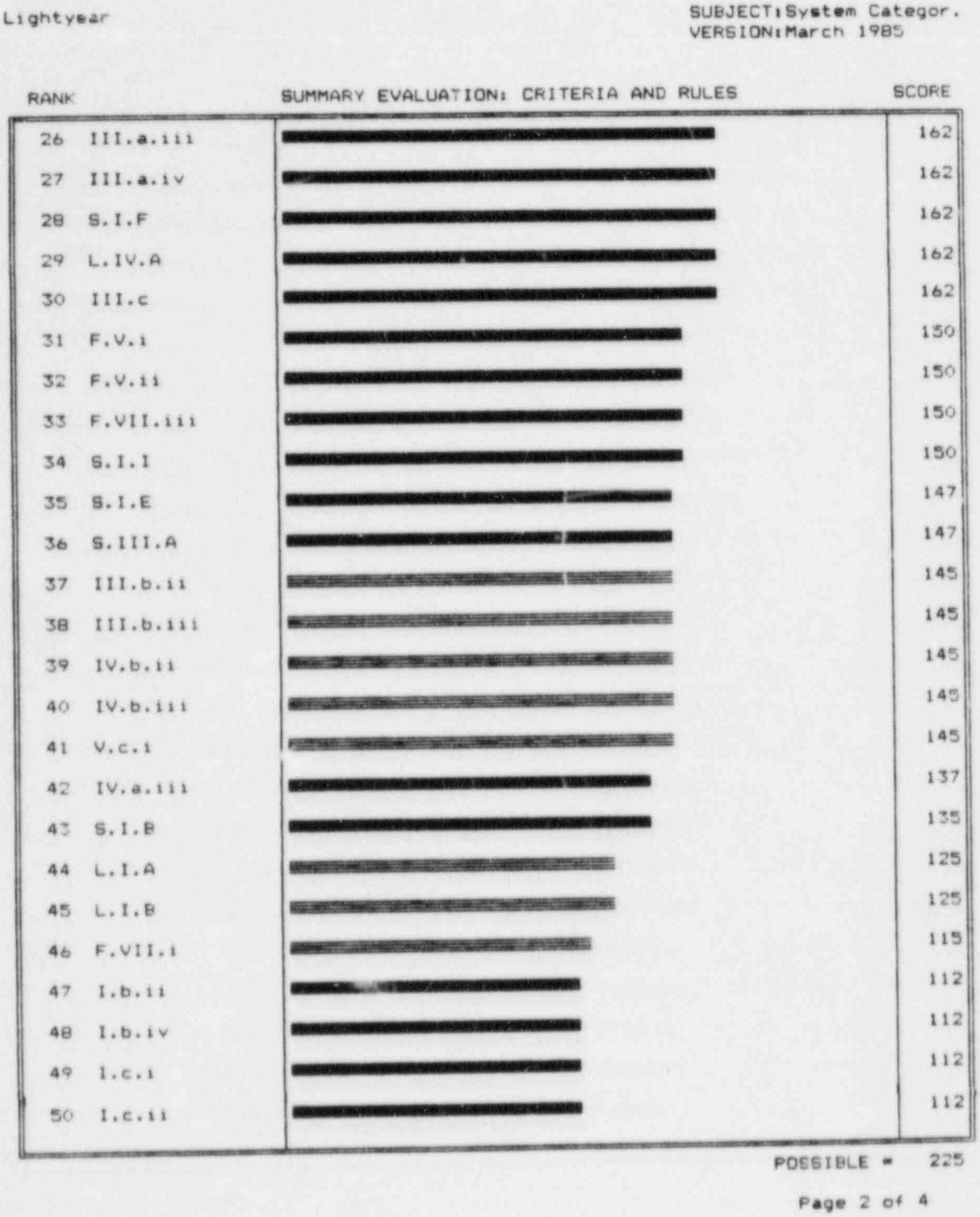
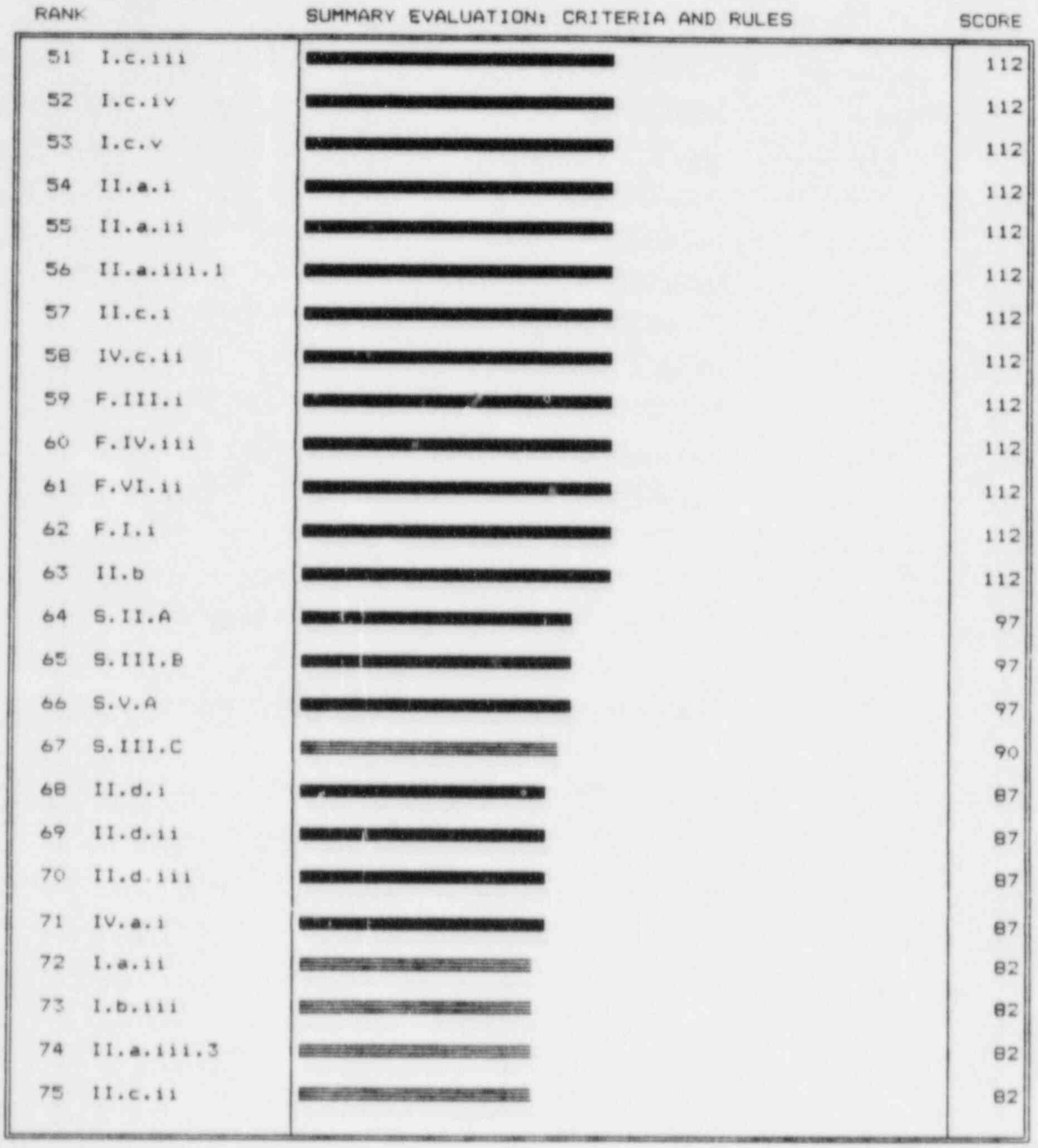


Figure 7-3 Results for the Initial Ranking
for the Systems Issues (Cont.)

Lightyear

SUBJECT:System Categor.
VERSION:March 1985



POSSIBLE = 225

Page 3 of 4

Figure 7-3 Results for the Initial Ranking for the Systems Issues (Cont.)

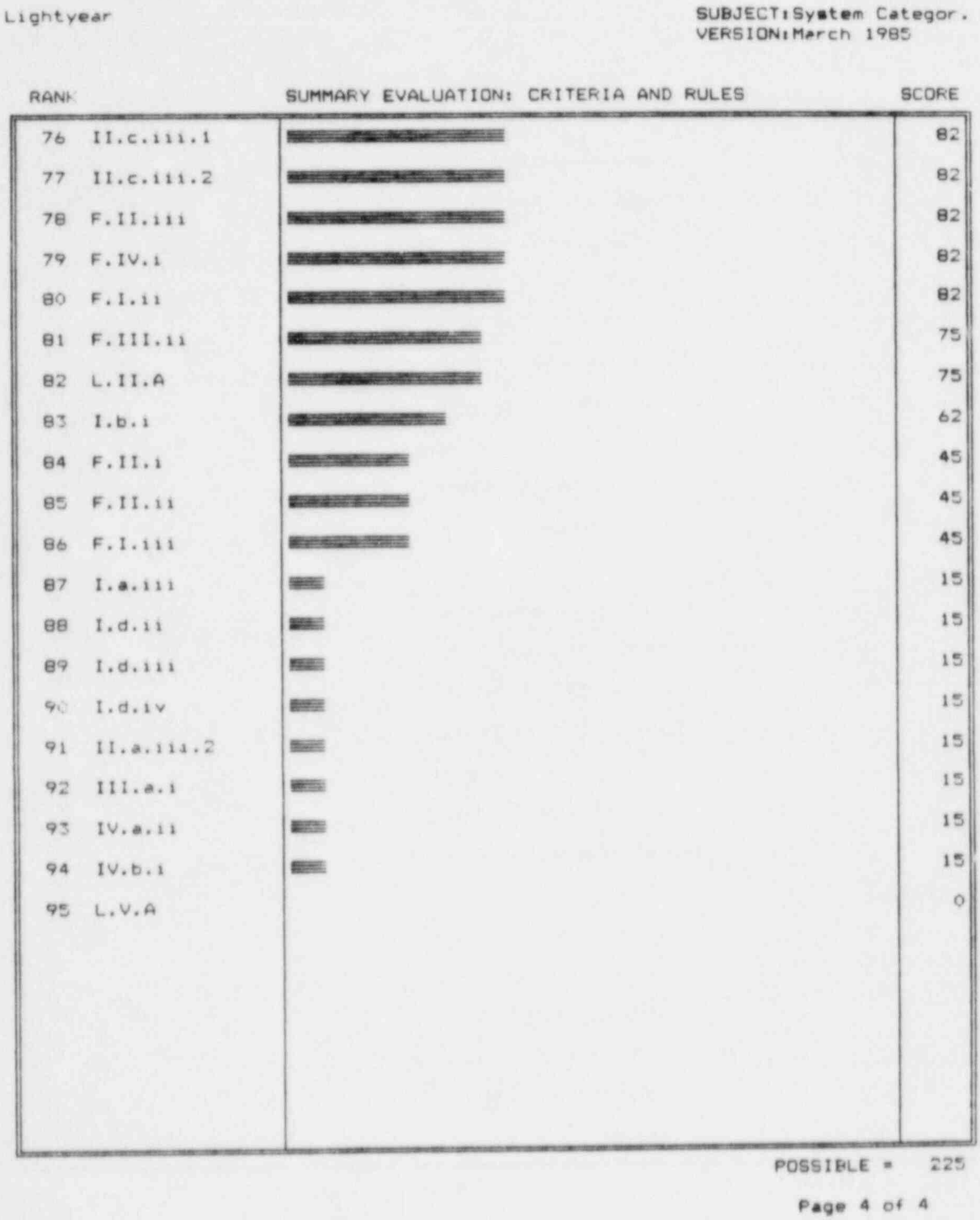


Figure 7-3 Results for the Initial Ranking for the Systems Issues (Cont.)

Lightyear

SUBJECT:System Categor.
VERSION:March 1985

RULE NAME	RULE
Uncertainty Rul	Uncert. Effect MUST BE AT LEAST Medium (ELIMINATION RULE)
Research Rule	Research MUST BE AT LEAST Medium (ELIMINATION RULE)
Increase Rule	Increase Pot. MUST BE AT LEAST Medium (ELIMINATION RULE)

Figure 7-4 Rules for the Second Ranking

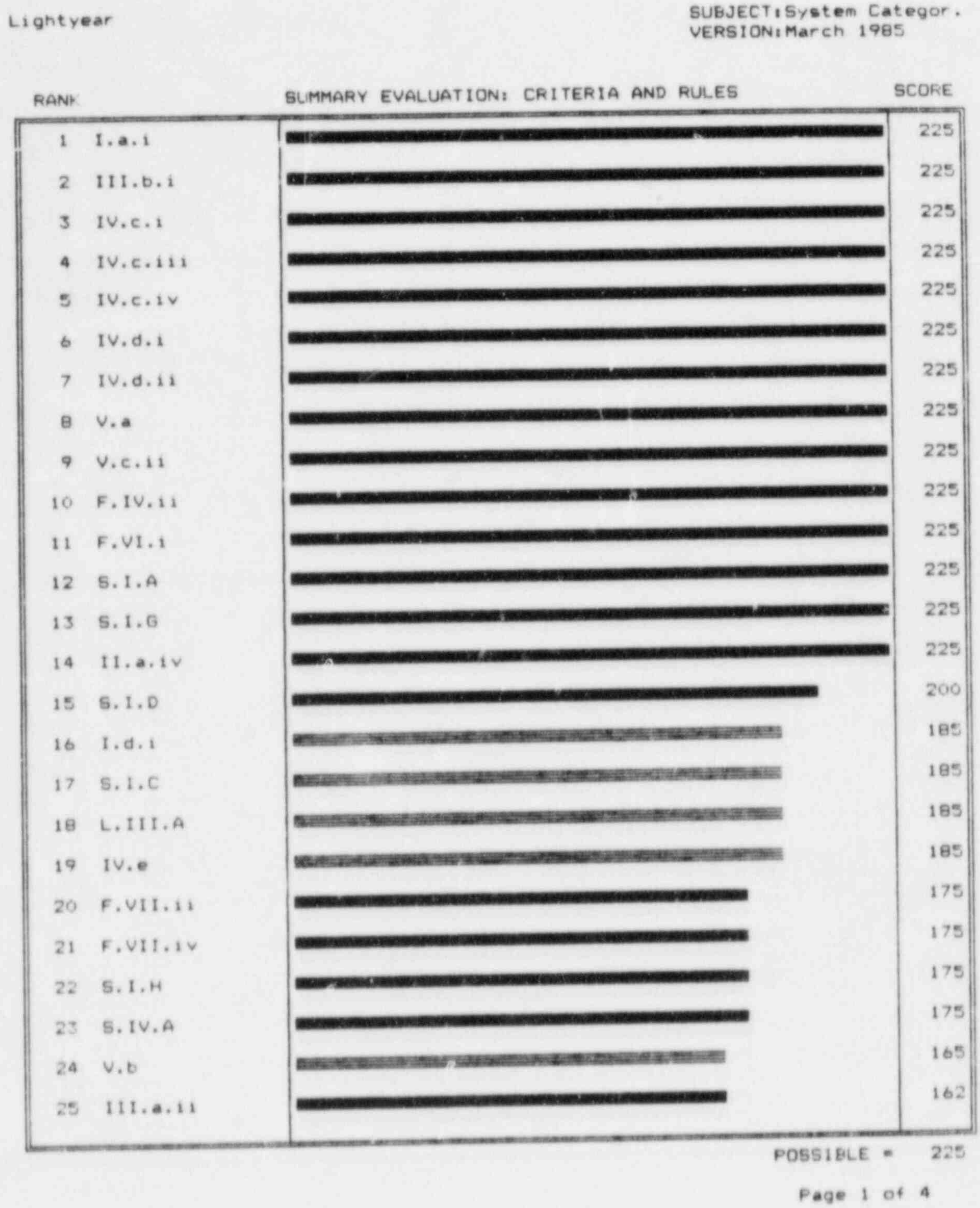


Figure 7-5 Results of the Second Ranking for the Systems Issues

Lightyear

SUBJECT:System Categor.
VERSION:March 1985

RANK		SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
26	III.a.iii		162
27	III.a.iv		162
28	S.I.F		162
29	L.IV.A		162
30	III.c		162
31	F.V.i		150
32	F.V.ii		150
33	F.VII.iii		150
34	S.I.I		150
35	S.I.E		147
36	S.III.A		147
37	III.b.ii		145
38	III.b.iii		145
39	IV.b.ii		145
40	IV.b.iii		145
41	V.c.i		145
42	IV.a.iii		137
43	S.I.B		135
44	L.I.A		125
45	L.I.B		125
46	F.VII.i		115
47	I.b.ii		112
48	I.b.iv		112
49	I.c.i		112
50	I.c.ii		112

POSSIBLE = 225

Figure 7-5 Results of the Second Ranking for the Systems Issues (Cont.)

RANK		SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
51	I.c.iii		112
52	I.c.iv		112
53	I.c.v		112
54	II.a.i		112
55	II.a.ii		112
56	II.a.iii.1		112
57	II.c.i		112
58	IV.c.ii		112
59	F.III.i		112
60	F.IV.iii		112
61	F.VI.ii		112
62	F.I.i		112
63	II.b		112
64	S.II.A		97
65	S.III.B		97
66	S.V.A		97
67	S.III.C		90
68	II.d.i		87
69	II.d.ii		87
70	II.d.iii		87
71	IV.a.i		87
72	I.a.ii		82
73	I.b.iii		82
74	II.a.iii.3		82
75	II.c.ii		82

POSSIBLE = 225

Figure 7-5 Results of the Second Ranking for the Systems Issues (Cont.)

Lightyear

SUBJECT:System Categor.
VERSION:March 1985

RANK		SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
76	II.c.iii.1		82
77	II.c.iii.2		82
78	F.II.iii		82
79	F.IV.i		82
80	F.I.ii		82
81	F.III.ii		75
82	L.II.A		75
83	I.b.i		62
84	F.II.i		45
85	F.II.ii		45
86	F.I.iii		45
87	I.a.iii		15
88	I.d.ii		15
89	I.d.iii		15
90	I.d.iv		15
91	II.a.iii.2		15
92	III.a.i		15
93	IV.a.ii		15
94	IV.b.i		15
95	L.V.A		0

POSSIBLE = 225

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Figure 7-5 Results of the Second Ranking for the Systems Issues (Cont.)

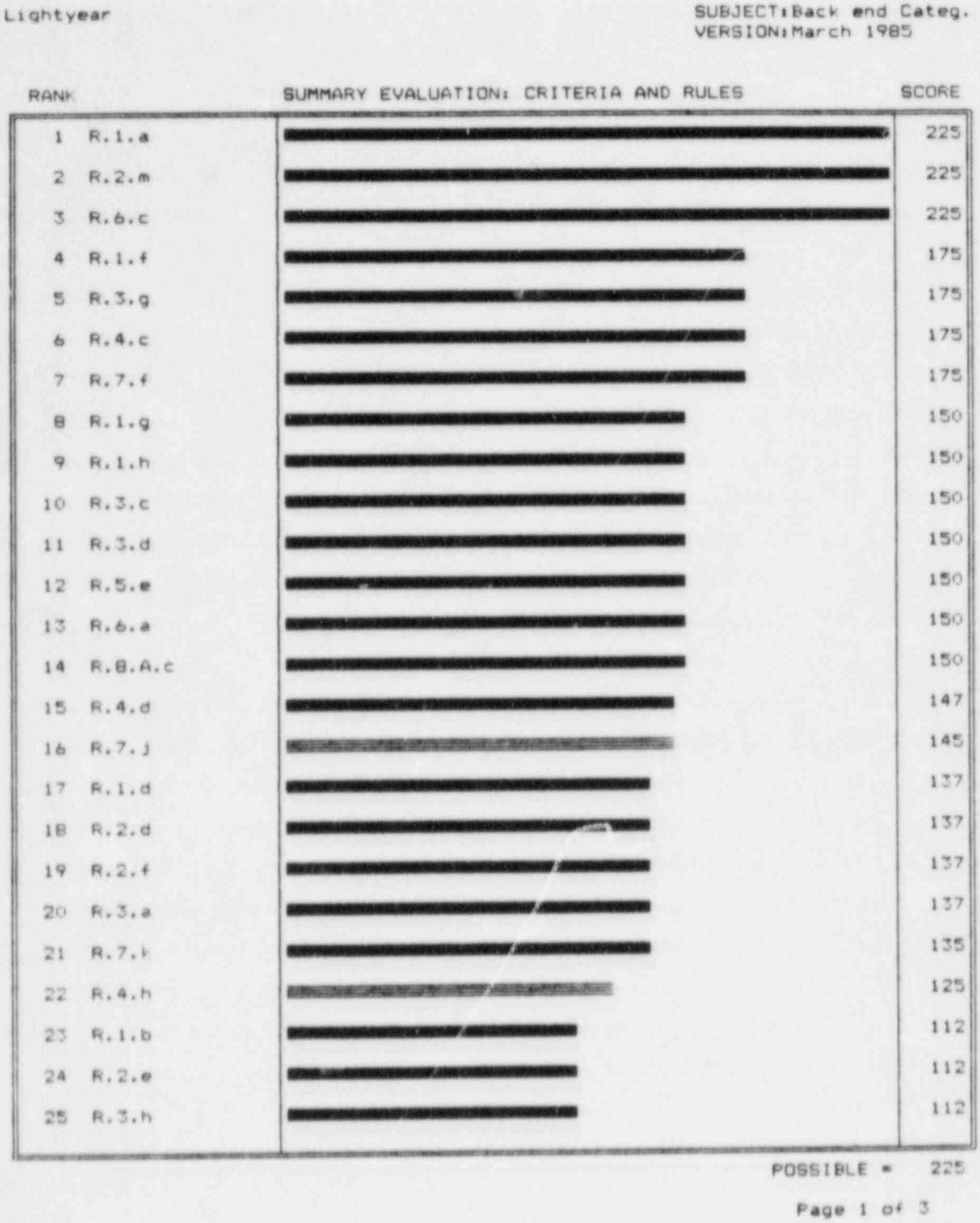

Figure 7-6 Results of the Initial Ranking for the Containment and Consequence Issues

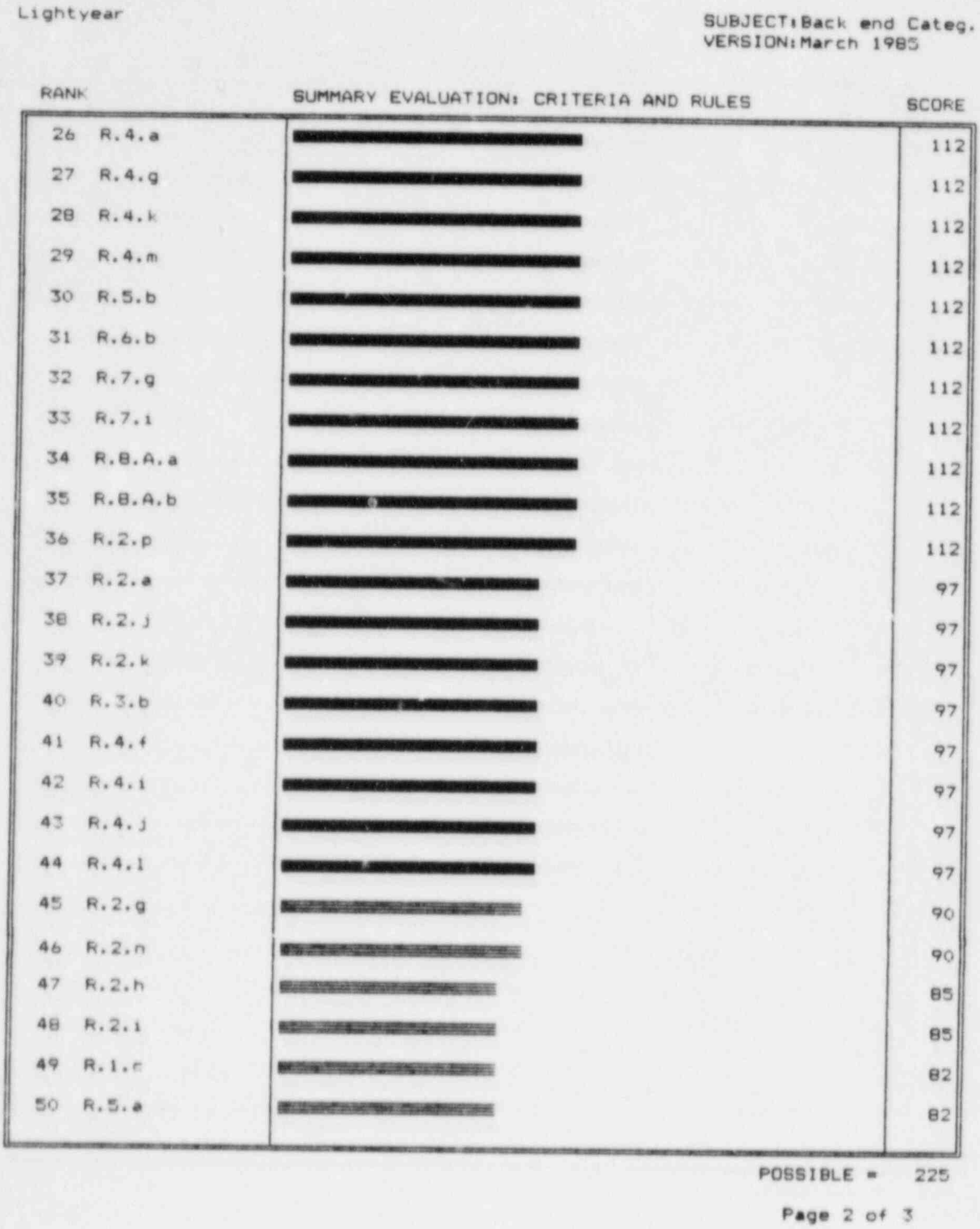

Figure 7-6 Results of the Initial Ranking for the Containment and Consequence Issues (Cont.)

Lightyear

SUBJECT:Back end Categ.
VERSION:March 1985

RANK		SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
51	R.5.g		82
52	R.3.e		75
53	R.3.f		75
54	R.4.b		75
55	R.4.e		75
56	R.5.f		67
57	R.1.e		60
58	R.2.b		60
59	R.2.c		60
60	R.5.d		60
61	R.5.h		60
62	R.7.h		60
63	R.2.1		45
64	R.2.o		45
65	R.5.c		45
66	R.7.a		45
67	R.7.b		45
68	R.7.c		45
69	R.7.d		45
70	R.7.e		45

POSSIBLE = 225

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Figure 7-6 Results of the Initial Ranking for the Containment and Consequence Issues (Cont.)

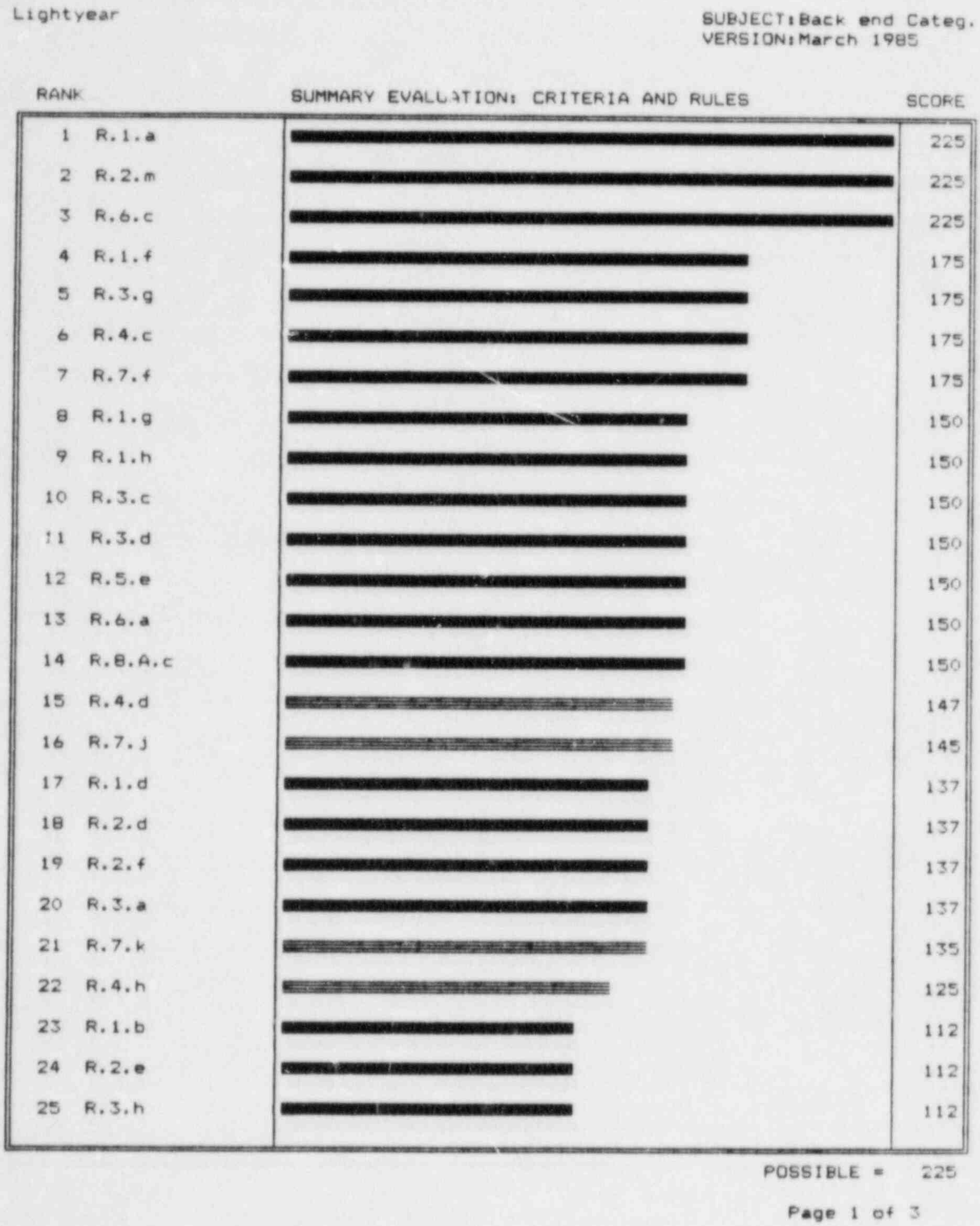

Figure 7-7 Results of the Second Ranking for the Containment and Consequence Issues

Lightyear

SUBJECT: Back end Categ.
VERSION: March 1985

SUMMARY EVALUATION: CRITERIA AND RULES

RANK		SCORE
26	R.4.a	112
27	R.4.g	112
28	R.4.k	112
29	R.4.m	112
30	R.5.b	112
31	R.6.b	112
32	R.7.g	112
33	R.7.i	112
34	R.8.A.a	112
35	R.8.A.b	112
36	R.2.p	112
37	R.2.a	97
38	R.2.j	97
39	R.2.k	97
40	R.3.b	97
41	R.4.f	97
42	R.4.i	97
43	R.4.j	97
44	R.4.l	97
45	R.2.g	90
46	R.2.n	90
47	R.2.h	85
48	R.2.i	85
49	R.1.c	82
50	R.5.a	82

POSSIBLE = 225

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Figure 7-7 Results of the Second Ranking for the Containment and Consequence Issues (Cont.)

Lightyear

SUBJECT: Back end Categ.
VERSION: March 1985

SUMMARY EVALUATION: CRITERIA AND RULES

RANK		SCORE
51	R.5.g	82
52	R.3.e	75
53	R.3.f	75
54	R.4.b	75
55	R.4.e	75
56	R.5.f	67
57	R.1.e	60
58	R.2.b	60
59	R.2.c	60
60	R.5.d	60
61	R.5.h	60
62	R.7.h	60
63	R.2.l	45
64	R.2.o	45
65	R.5.c	45
66	R.7.a	45
67	R.7.b	45
68	R.7.c	45
69	R.7.d	45
70	R.7.e	45

POSSIBLE = 225

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Figure 7-7 Results of the Second Ranking for the Containment and Consequence Issues (Cont.)

Lightyear

SUBJECT: Back end Categ.
VERSION: March 1985

RULE NAME	RULE
Uncertainty Rul	Uncert. Effect MUST BE AT MOST Small (ELIMINATION RULE)
Increase Rule	Increase Pot. MUST BE AT LEAST Medium (ELIMINATION RULE)

Figure 7-8 Rules for the Example "Completeness of Rules" Analysis

Lightyear

SUBJECT: Back end Categ.
VERSION: March 1985

RANK	SUMMARY EVALUATION: CRITERIA AND RULES	SCORE
1	R.1.a	225
2	R.2.m	225
3	R.6.c	225
4	R.1.f	175
5	R.3.g	175
6	R.4.c	175
7	R.7.f	175
8	R.1.g	150
9	R.1.h	150
10	R.3.c	150
11	R.3.d	150
12	R.5.e	150
13	R.6.a	150
14	R.8.A.c	150
15	R.4.d	147
16	R.7.j	145
17	R.1.d	137
18	R.2.d	137
19	R.2.f	137
20	R.3.a	137
21	R.7.k	135
22	R.4.h	125
23	R.1.b	112
24	R.2.e	112
25	R.3.h	112

POSSIBLE = 225

Page 1 of 3

Figure 7-9 Results for the Example "Completeness of Rules" Analysis for the Containment and Consequence Issues

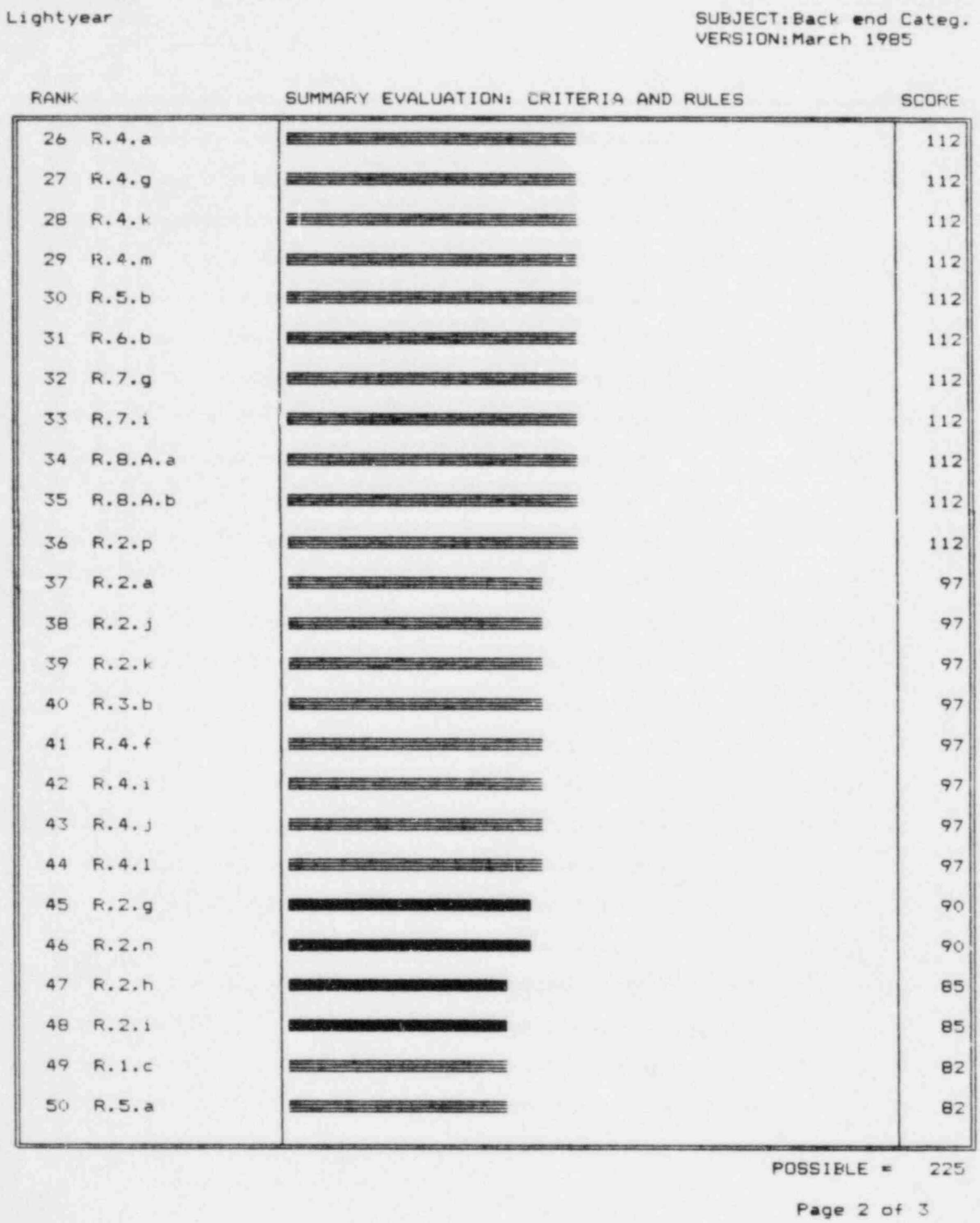

Figure 7-9 Results for the Example "Completeness of Rules" Analysis for the Containment and Consequence Issues (Cont.)

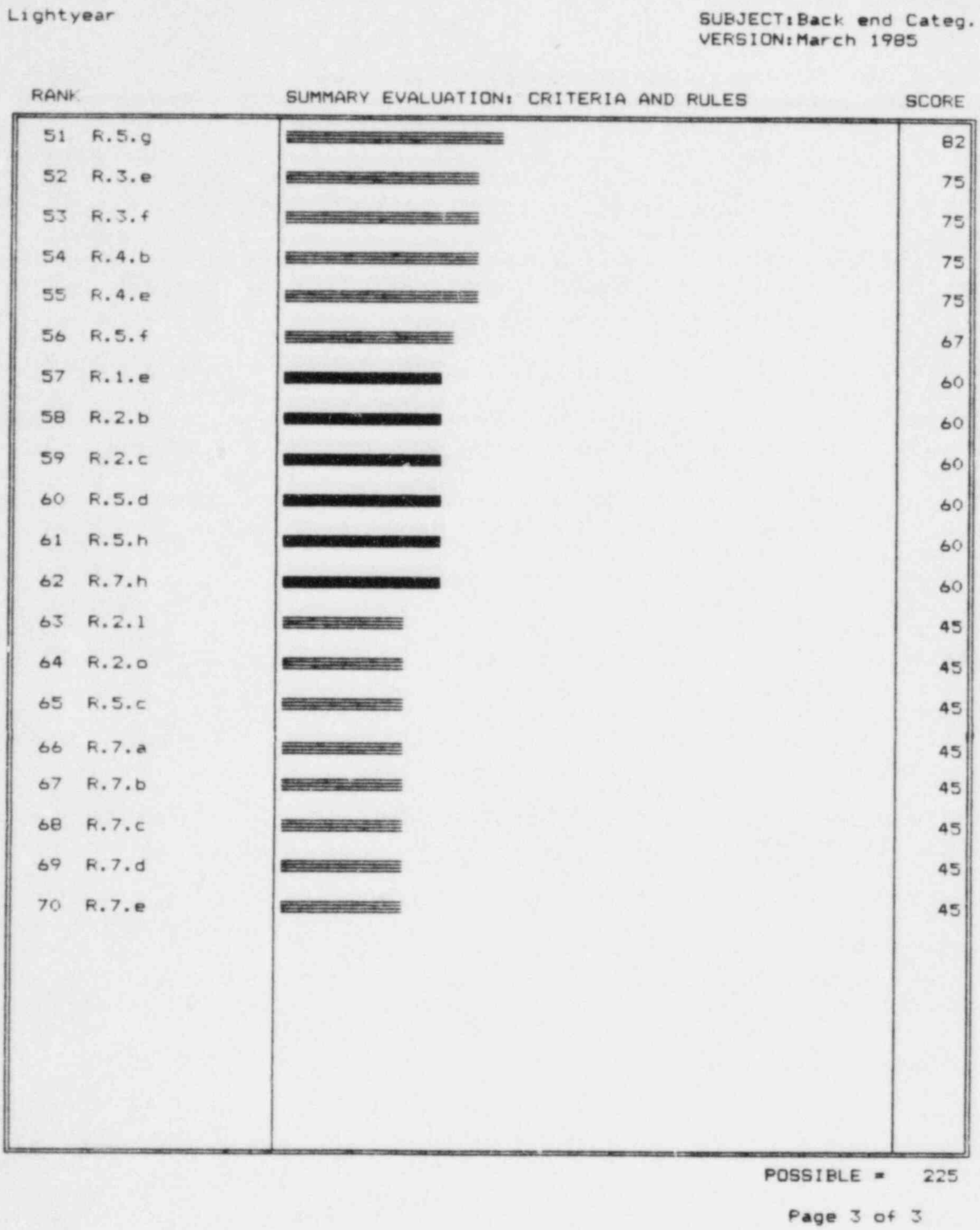

Figure 7-9 Results for the Example "Completeness of Rules" Analysis for the Containment and Consequence Issues (Cont.)

8. FUTURE WORK AND OTHER APPLICATIONS

Plans for the future are to refine the data base of risk issues and to extend the applications. With regard to the data base, plans are to further define the risk issues and their uncertainty impacts. The PRAs which have been performed and other safety analyses can be used to obtain more quantitative evaluations of the uncertainty impacts of the risk issues. The uncertainty contributions and risk importances which are obtainable from these PRA sources, while not all-inclusive, are extremely useful for checking and refining the present category rankings of the risk issues.

The available PRA studies also provide an important resource of risk contributors and risk uncertainty issues. The dominant contributors to risk and to risk uncertainty which are identified can be used to refine and extend the present set of risk issues. Issues which have been identified in other efforts, such as the Generic Safety Issues, also will be a resource which will be further investigated for refining the present data base of issues and their impacts.

For the future, plans also are to incorporate the risk issues in a data base system which can be easily assessed by the decision maker. The commercially available data base systems for the IBM PC such as DBASE III are being specifically examined for storing and managing the risk issues. A data base management system will allow risk issues to be easily updated as research progresses.

To improve understanding and implementation, the risk issues will be better organized. Issues will be grouped under general research areas to give insights on these areas. Issues of broad scope will thus be better differentiated from issues of narrow scope. The issues will also be more fully defined and characterized to better indicate their relations to research activities and responsibilities. The planned data base management system will allow the risk issues to be fully described ad will allow the risk issues to be easily cross-cut in different ways.

Plans are under consideration to expand the criteria to better characterize research alternatives and to allow the risk issue data base to be used in other applications. Additional research criteria include the cost to resolve the issue and the time to resolve the issue. The cost criterion can be defined in terms of the degree to which the uncertainty can be resolved for less than \$250K, for less than \$500K, etc. The time to resolve the issue can similarly be defined in terms of the degree to which the uncertainty can be resolved within one year, within three years, etc. These criteria can help to identify the most cost-effective research alternatives and will allow category responses such as "high," "medium," and "low."

To extend the applications, criteria can be defined to more clearly separate the risk level impacts from the risk uncertainty impacts. Issues which have little uncertainty but which impact the risk level could thereby be added to the data base. Identification of these issues would be extremely useful for regulatory applications. Criteria can also be defined to better identify the scope of impact on the issue. Issues can thereby be better identified which have a medium or low generic impact but which can significantly impact one or two outlier plants. This could be useful for regulatory applications. The data base could also be used in helping to identify where regulations have little or no impact on risk and, thus, could be loosened or be removed.

Detailed plans for the improvement of the data base and for extensions of the applications are thus presently being developed and will be implemented in a timely fashion to meet the agency's needs.

9. REFERENCES

1. Dressler, E., and H. Sobottka, "Optimization of Safety Systems Designed by Reliability Analysis," in Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, New York, 1981.

2. McClymont, A., and G. McLagan, "Diesel Generator Reliability at Nuclear Power Plants: Data and Preliminary Analysis," EPRI-NP-2433, June 1982.

3. Atwood, C. L., and J. A. Stevenson, "Common-Cause Fault Rates for Diesel Generators: Estimates Based on Licensee Event Reports at US Commercial Nuclear Power Plants, 1976-1978," NUREG/CR-2099, EGG-EA-5359, Rev. 1, June 1982.

4. Letter from D. J. Campbell (JBF Associates), to M. P. Bohn (Sandia National Laboratories), Subject; RMIEP Dependent Failure Analysis Project - Task 1 Letter Report, April 9, 1984.

5. Fleming, K. N., et al, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," PLG-400, NP-3967, February 1985.

6. Riddell, R., and N. M. Newmark, "Statistical Analysis of the Response of Non-linear Systems Subjected to Earthquakes," Report UILU 79-2016, Department of Civil Engineering, University of Illinois at Urbana-Champaign, August 1979.

7. Emrit, R., et al, "A Prioritization of Generic Safety Issues," NUREG-0933, December 1983, Revision 1, July 1984.

APPENDIX A

SYSTEMS RELATED ISSUES

APPENDIX A

SYSTEMS RELATED ISSUES

This appendix contains tables identifying the systems related issues that impact risk. The issues are rated according to various criteria and notes are included that provide some insights into the evaluation process. These notes should not be considered a comprehensive or consistent discussion of the issues. A complete description of the evaluation methodology is provided in Chapter 3. Four tables are provided in this appendix: Table A-1 deals with internal event issues; Table A-2 deals with fire issues; Table A-3 deals with seismic issues; and Table A-4 deals with flooding issues.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
I. <u>Hardware Issues</u>					
a. Uncertainties associated with components having relatively <u>high</u> failure rates in a normal operating environment (e.g., diesels, pumps, active valves)	Large				An example of issue Ia. is diesel generators demanded in response to a loss of offsite power. Diesel generators are often the dominant contributors to loss of offsite power sequences. Loss of offsite power sequences have been dominant in many PRAs. If the diesel unavailability changes by a factor of 3, the sequence frequency could be affected by a factor of 10 to 30.
i-Equipment failure reporting accuracy	Large	Large	Large	Large	The reporting accuracy has been found to affect both the numerators and denominators used to calculate unavailabilities and could change the sequence frequency by an order of magnitude.
ii-Data analysis techniques	Medium	Small	Medium	Medium	Estimating plant-specific failure rates where data is sufficient is fairly well established. However, generic data techniques can cause medium to large variations in generic results.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
iii-Accelerated equipment failure due to aging	Small				Components with relatively high failure rates that are affected by aging would be detected during periodic tests.
b. Uncertainties associated with components having low failure rates in a normal environment (eg, pipe breaks, vessel ruptures, mechanically caused control rod insertion failures)	Medium				An example of issue Ib. is failure of a RCS pipe that leads to a large LOCA. Large LOCA sequences have been important in some PRAs but not usually dominant. If the large LOCA frequency is off by an order of magnitude (e.g., LOCA frequencies may increase in the future due to intergranular stress corrosion) the affect on the overall core melt frequency should probably be less than an order of magnitude.
i-Lack of available data	Medium	Medium	Medium	Medium	One way to get more data on low failure components is to continue to monitor plants. Some uncertainty reduction could be obtained by collecting non-nuclear data which have applicability.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
i-Lack of available data	Large	Large	Large	Small	Most PRAs have been conservative on failure criteria. Components that are important to core melt frequency have generally been assumed to fail in beyond design basis conditions.
ii-Accelerated equipment failure due to aging	Small				Issues dii and diii are believed to have a second order effect upon issue d compared to issue di. Since such little is known about equipment performance operating in a beyond design basis environment, it would be difficult to assess the individual effects of issues ii and iii. The primary problem in this area is lack of data.
iii-Failure rate prediction methods	Small				

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
d. Uncertainties associated with survival of components operating in a beyond design basis environment	Large				This issue is important for several situations, for example: 1. ATWS sequences involving large pressure spikes 2. Battery operation and reactor coolant pump seal performance during station blackout sequences 3. Loss of decay heat removal in which the suppression pool overheats (BWR) 4. Loss of component air and water cooling systems (e.g., loss of a pump room cooling systems) The four examples listed above can affect core melt frequencies by more than an order of magnitude, e.g., the core melt frequency for the WASH-1400 PWR could increase by a factor of 10 if pessimistic assumptions are made regarding battery depletion and RCP seal performance.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
c. Uncertainties associated with components operating in a LOCA environment (e.g., ECCS actuation system components - transmitters, cables, terminals)	Medium				An example of Issue Ie. is instrumentation located within containment that must operate during a small LOCA or during feed and bleed core cooling in a PWR. This instrumentation may be affected at any time during the release, and certainly with a higher probability in the latter phases of the accident with a significant steam dump to the containment. Local environments and spatial conditions could affect ECCS actuation, but small-break events experienced to date have not prevented ECCS actuation. A complicating factor is that components tend to "fail" suddenly although they may be continuously degrading (undetected); hence, the operator may not have warning of impending failure. An additiinal concern is that the operator could terminate core cooling due to misinterpretation of erroneous control room information. Assuming a relatively high operator error probability yields a medium impact on core melt frequency.
i-Lack of available data	Medium	Medium	None	Medium	
ii-Accelerated equipment failure due to aging	Medium	Medium	None	Medium	
iii-Failure rate prediction methods	Medium	Medium	None	Medium	
iv-Survivability of equipment	Medium	Medium	None	Medium	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
ii-Accelerated equipment failure due to aging	Medium	Medium	None	Medium	As plants get older, the components which are considered highly reliable may become less so without being detected in testing (e.g., vessel response to PTS).
iii-Failure rate prediction methods	Medium	Medium	Medium	Medium	The methods used to estimate failure rates when there is little or no data available generally has less than an order of magnitude effect on the core melt frequency. This is because components with low failure rates are generally not found in dominant sequences. This uncertainty may be reduced somewhat by using methods based on the physics of failures.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
II. Issues affecting Human Behavior					
a. Uncertainties associated with failure of the operator to correctly follow procedures during accident situations (e.g., switchover from injection to recirculation, actuation of feed and bleed cooling)	Medium				An example of issue IIa. is failure of the operators to to correctly follow procedures when establishing "feed and bleed" core cooling. "Feed and bleed" sequences have been dominant in several PWR PRAs. This operator error can influence plant core melt frequencies by a factor of 2 more than depending on the error probability assumed.
i-Lack of human reliability data	Medium	Medium	Medium	Medium	Large uncertainties exist when modeling operator response during an accident. We feel that the current uncertainties can be reduced somewhat by evaluating human reliability models and comparing the results to an increased data source. Though large uncertainties exist, operator errors in following procedures generally only have a moderate impact on total plant core melt frequency.
ii-Human reliability analysis techniques (e.g., Therp models vs. MAPPS, SLIM-MAUD models)	Medium	Medium	Medium	Medium	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
iii-Ability to assess performance shaping factors	Medium	Medium	Medium	Medium	
1. Quality of procedures	Medium	Medium	Medium	Medium	
2. Control room design (layout, alarm configuration)	Small				
3. Operator variability (training, response to stress situations, etc.)	Medium	Small	Medium	Medium	
iv. Ability to assess common-cause failures caused by humans	Large	Large	Small	Large	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
b. Uncertainties associated with accuracy of core cooling related instrumentation in accident situations	Medium	Medium	Small	Medium	PRAs typically assume that critical instrumentation is available. The TMI pressurizer level gauge is a good example of unavailability of critical instrumentation.
c. Uncertainties associated with system and component restoration errors following test and maintenance activities	Medium				The prediction of human behavior always has the potential to cause large uncertainties. However, human reliability estimates can predict behavior for test and maintenance type procedures better than for operator behavior during accident situations. We therefore feel that research will not significantly reduce the uncertainty.
i-Restoration error reporting accuracy	Medium	Medium	Medium	Medium	
ii-Human reliability analysis techniques	Medium	Small	Medium	Medium	
iii-Ability to assess performance shaping factors	Medium	Small	Medium	Medium	
1. Quality of Procedures	Medium	Small	Medium	Medium	
2. Variability among personnel	Medium	Small	Medium	Medium	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
d. Uncertainties associated with failure of operator to perform recovery actions during accident situations (e.g., actions not explicitly described in procedures)	Medium				An example of issue IId. is failure of the operator to manually start the AFWS given failure of automatic actuation. Calvert Cliffs and ANO IREP showed that consideration for recovery for all accident sequences can reduce plant core melt frequencies by a factor of 5 to 10.
i-Lack of human reliability data	Medium	Medium	Medium	None	
ii-Human reliability analysis techniques	Medium	Medium	Medium	None	
iii-Ability to to assess performance shaping factors	Medium	Medium	Medium	None	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
III. Initiating Event Issues					
a. Uncertainties associated with frequent initiators that occur during power operation (e.g., turbine trips, reactor trips)	Medium				Frequent initiators, such as turbine trips and loss of offsite power can typically differ by a factor of 3 among plants. Since frequent initiators usually comprise the dominant sequences, the plant core melt frequency could also vary by the same factor.
i-Modeling of interactions between initiating events and mitigation systems	Small				Interactions between frequent initiators that dominate core melt frequency (transients, small LOCAs) and mitigation systems can be identified.
ii-Initiating event reporting accuracy	Medium	Large	Medium	Medium	Inappropriate grouping of initiators can lead to moderate differences (e.g., factors of three in initiating event frequencies.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
iii-Frequency of transients during plant burn in	Medium	Large	None	Medium	Current PRAs have not differentiated between the initiating event frequency during the first few years and the rest of the plant life. Differentiation could cause moderate increases in event frequencies in early life.
iv-Statistical methods used to estimate initiating event frequencies (includes applicability of generic data to specific plants)	Medium	Large	Medium	Medium	Even though generic initiating event information may be misused, the effect is usually less than an order of magnitude.
b. <u>Infrequent</u> initiators that occur during power operation (e.g., larger LOCAs and support system caused initiating events)	Large				An example of issue IIIb. is an initiating event caused by failure of the component cooling water system in a PWR. If pessimistic assumptions are made, this event could directly lead to core damage. The core melt frequency for a plant could increase by greater than an order of magnitude, depending on assumptions and on the initiating event frequency.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
i-Modeling of interactions between initiating events and mitigating systems	Large	Large	Small	Large	Support system initiating events have been shown to be important by PRAs and the industry because they cause trips and degrade safety systems. These events are responsible for large uncertainties because the interactions can be extremely subtle.
ii-Lack of data	Large	Small	Large	Large	Presently, the frequencies used for these initiators are often subjective and could be significantly inaccurate. Unfortunately, the only way to improve the data base is to record data from industry experience.
iii-Methods used to estimate infrequent quent initiating event frequencies	Large	Small	Large	Large	The methods may affect the core melt frequency by greater than an order of magnitude because several dominant sequences can be affected by a single initiating event. The real problem here is the lack of data, not the methods.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
c. Initiating events that occur during non-full power operation	Medium	Large	None	Medium	This type of event has been typically ignored by PRA. This type of incident did show up in the Accident Sequence Precursor Study report, but was evaluated to be insignificant. However, a more in-depth look at these types of sequences may be warranted.
IV. <u>Issues Related to System Response</u>					
a. Uncertainties associated with system/function success criteria	Medium				An example of issue IVa. is the number of ECCS pump trains required to successfully cool the core following a LOCA.
i-Consideration of system or components which are partially failed	Medium	Medium	Medium	None	PRAs have typically conservatively classified LERs involving partial failure as total failure. Further research may find that system and component failure probabilities have been moderately overestimated.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
ii-Consideration of whether systems are adequately designed to accomplish their designated purpose	Small				PRAs have typically used ECC system success criteria that are believed to be conservative and are supported by FSAR calculations. However, there is still some concern that this criteria may not be conservative due to the various thermal hydraulic modeling assumptions made. We recognize this but believe these assumptions affect the prediction of whether some core damage occurs during an accident rather than whether a full-scale core meltdown occurs.
iii-Consideration of all safety and non-safety systems which can mitigate the sequence	Medium	Large	Medium	None	BWRs have many alternate injection pathways which have typically not been handled by PRAs. The reduction in core melt frequency when considering these alternate pathways is limited by support system interactions. Present day PRAs are considering these alternate success states.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
b. Uncertainties associated with system modeling depth	Large				System modeling depth deals with the completeness of the system model, i.e., have the fault models included all significant system failure modes. If the fault models are incomplete, it is conceivable that the plant core melt frequency could be grossly inaccurate.
i-Inclusion of all important flow paths	Small				This has been handled adequately in the past.
ii-Inclusion of important support system dependencies	Large	Small	Small	Large	The mishandling of support system dependencies can cause wide variation in core melt frequency. However, they can be handled correctly using the current state-of-the-art PRA techniques.
iii-Simplification of support system models	Large	Small	Small	Large	Again, the mishandling of simplified models can cause large uncertainties. However, simplifications can be used effectively using today's techniques if the interdependencies are done correctly.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
c. Uncertainties associated with common cause modeling and data	Large				
i-Reporting accuracy	Large	Large	Medium	Large	Past PRAs have not adequately treated common cause because of problems identified as issues i, iii, and iv. These three issues have the potential for large impact on core melt frequency. Given that these three issues are resolved, various existing common-cause estimation methods can be used to obtain probabilities. These methods can have a moderate effect on core damage frequency estimates.
ii-Methods used to estimate common cause probabilities and frequencies	Medium	Medium	Medium	Medium	
iii-Inadequate plant procedures	Large	Large	Medium	Large	
iv-Common physical cause (e.g., corrosion, moisture, vibration)	Large	Large	Medium	Large	

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
d. Uncertainty associated with accuracy of information used in safety analyses	Large				An example of this issue is a PRA based on FSAR information versus a PRA based on accurate plant design and operation information. Comparison of Calvert Cliffs RSSMAP (FSAR information) and IREP (accurate information) indicates a factor of 3 difference. However, it is conceivable that inaccurate information could influence predicted accident frequencies by several orders of magnitude.
Design information	Large	Large	Large	Large	This depends on the study. Plant specific studies do better on this. However, there always seems to be a problem getting information from the utilities (even on EPRI programs).
Operational information	Large	Large	Large	Large	
e. Uncertainty associated with system operability or recoverability in accident	Large	Large	Large	Small	This issue is similar to issue addressed in I,d.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
V. Issues related to accident sequences analysis					
a. Uncertainties associated with definition of event tree sequences	Large	Large	Large	Large	Sequences may have been identified in PRA's that do not reflect current understanding of plant response. PRAs may have omitted sequences that are known to exist based on proper understanding.
b. Modeling of Interactions between the initiating event and the event tree systems	Small (frequent initiators) Large (infrequent initiators)	Large (infrequent initiators)	Small (infrequent initiators)	Large (infrequent initiators)	This issue is similar to the Issue addressed in III.a.1, and III.b.i.

Table A.1. Relative Ranking of Internal Event Issues that Impact Uncertainty in

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potentia this iss increase core mel frequenc
c. Uncertainty associated with modeling of interactions among event tree systems	Large			
i-Hard wired interactions	Large	Small	Small	Lar
ii-Non hard wired interactions (common cause -- corrosion, etc.)	Large	Large	Small	Lar

Core Melt Frequency (Continued)

for e to the	Comments and Discussion
	Systems that appear an event trees are generally not independent, e.g., they often share support systems. If interactions due to shared support systems are not properly accounted for, core melt frequencies can be affected by greater than an order of magnitude. For example, if two event tree systems have unavailabilities of 10^{-2}/demand, the probability of failing both could be between 10^{-2} and 10^{-4}, depending on the degree of support system sharing between the two.
e	If handled incorrectly, this issue can cause large uncertainties. However, it can be handled with current state-of-the-art techniques.
e	This issue is similar to issue addressed in IV.c.

Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in Core Melt Frequency

ssu s that impact nce tainty in core elt trequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
.I. Selection of Area for Analysis					
i-Area to Area fire spread environmental phenomena (HVAC) effects, smoke, heat)	Medium	Medium	None	Medium	Fire PRA's to date have argued that plant areas in which a fire can damage enough equipment to cause a core melt pose the greatest risk. These single-point vulnerable areas have been analyzed using fire scenario mathematical models. However, many power plants rely on fire barriers to separate safety trains, therefore single area contributions to risk may be insignificant in comparison to area-to-area fire spread. Because heat and smoke from a nuclear power plant fire will not normally flow directly out of the plant through doors or windows, reliance must be placed on barriers and HVAC systems to control combustion products and prevent area-to-area spread. The degree to which these measures will be effective has not been demonstrated. In addition, the adequacy and reliability of barriers and barrier elements (e.g. fire dampers and doors) remains unknown. By excluding all but single fire areas from analysis, risk-important fire scenarios involving multiple fire areas may have been incorrectly ignored.
i-Barrier element reliability (random and human)	Medium	Small	None	Medium	
i i-Deterministic uncertainties of barriers	Small	Small	None	Small	

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Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in Core Melt Fr

Issues that impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Com
F.II. Ignition Probabilities and Mechanisms					
i-Area specific information (occupancy, type of area)	Small	Small	Small	Small	The dep of fir sta wei fic of the The fro ver tud an pro est imp
ii-Function of fuel type and amount	Small	Small	Small	Small	
iii-Growth and ignition phenomena	Medium	Medium	Small	Medium	

ency (Continued)

ts and Discussion

timated risk associated with fire
s directly on the assumed frequency
urrence and magnitude of the ignition
For PRA's already completed, generic
tics on nuclear power plant fires, with
ing factors to account for plant-speci-
tuations, have been used for a variety
nt areas. Large uncertainties surround
ighting factors and generic statistics.
uncertainties can shift a fire scenario
mportant to insignificant or vice-
If the ignition frequency and magni-
f fire occurrence can be quantified on
a and plant-specific basis, the
ilistic significance of fires can be
ted more accurately to focus on truely
ant fire weaknesses.

Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in Core Melt Frequency (Continued)

sues that impact cert inty in core it f equency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
III. Detection Effectiveness					
-Sensitivity and reliability relative to room geometries and environments (manual and auto)	Medium	Medium	None	Medium	Many plants use automatic detectors as the means of giving early warning to fire fighting teams or automatic extinguishing systems. PRA's to date have assumed the adequacy of these detection systems. Numerous installation details may prevent an otherwise adequate detector from functioning (e.g. HVAC flows, ceiling heights). In addition, some detection schemes may be needed to ensure that control cabinet fires do not get out of control. Finally, spurious detector operation may cause suppression systems to activate and damage equipment inside of a fire area.
i -Spurious actuation effects relative to suppression	Small	Medium	None	Small	

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Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in Cor

Is ues that impact un ertainty in core me t frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency
F.IV. <u>Suppression Effectiveness</u>				
i-Manual fighting effectiveness	Medium	Small	Small	Medium
ii-Secondary detrimental effects (floods, eqpt. damage)	Large	Large	None	Large
iii-Spurious actuation	Medium	Medium	None	Medium

elt Frequency (Continued)

Comments and Discussion

Manual fire fighting brigades are required at all nuclear power plants. Their function during a fire ranges from the primary fire control mechanism in areas not having automatic suppression, to a backup or mopup function where automatic suppression is installed. Little is known about how effective manual fire fighting would be in unventilated plant areas or in areas containing sensitive and redundant safety equipment. The ease with which hose sprays could be directed on fragile equipment instead of a fire source may be a significant factor in fire risk calculations.

The use of suppression systems (water, CO_2, Halon) in nuclear power plants has been assumed to always improve safety. It may be that the use (accidental or intentional) of suppression may actually reduce safety. In many nuclear power plants, safety systems having electrical equipment are located in close proximity. Often the equipment has never been tested to the environments caused by suppression systems. As a result, suppression usage may damage equipment not directly involved in a fire.

Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in Core Melt Frequency (Continued)

Issues that impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
F. Fire Environment Definition					
i-Applicability and adequacy of existing models	Large	Medium	Medium	Medium	In many nuclear power plants, spatial separation and fire suppression have been assumed to adequately prevent damage to redundant equipment in the same areas. To demonstrate this, a time dependent understanding of the temperatures and combustion products resulting from a fire in an area must be known. There is a lack of adequate models and data to accurately estimate fire environments. This shortcoming applies to the generic calculational methods now being used. The inability of models to handle hot gas layers, room-to-room fire spread, and feedback to source fires are a few examples. Of particular concern is the lack of relevant parameters in the models for evaluating equipment damage.
ii-Eg. Damage relevant parameter estimates by models	Large	Medium	Small	Medium	

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Table A.2. Relative Ranking of Fire Issues That Impact Uncertainty in C

Iss es that impact unc rtainty in core mel frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research reducing this uncertainty	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency
F.VI. Damage Thresholds and Mechanisms				
i-Component fragilities for temperature, smoke, moisture, corrosion	Large	Large	None	Large
i-Component failure modes unknown	Medium	Medium	None	Medium

Comments and Discussion

PRA's to date have assumed that fire damages equipment only by heat. Smoke, corrosion, humidity and water sprays have been ignored and, except for cabling, even heat has been ignored. Unlike equipment designed and tested for LOCA environments, equipment expected to function in a fire environment is never tested for its functionality. Because many power plants have redundant equipment in close proximity (often in the same cabinet), damage can be expected to occur eventually during a fire. Little is known about the thresholds of failure and failure modes of safety equipment during fire. The extent to which fire poses a risk will depend on how easily equipment is damaged and in the manner (benign or detrimental) in which the equipment fails.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues					
A. Relay Chatter and Locking Circuits	Large	Large	Small	Large	Fragilities for electrical components represent a special problem due to the wide variety of electrical gear found within a plant. The two weakest failure modes are relay chatter and inadvertent trip of circuit breakers. Relay chatter is the lowest failure mode and, <u>if included in a risk analysis, would be the dominant failure</u>. Because, in most cases, chatter of relays would not cause a change in the state of a system being controlled, past PRAs assumed that relay chatter was not a problem, and included only circuit breaker trip as the failure mode for electrical gear. Before continuing to make this assumption, however, one should carefully investigate whether or not there are certain locking circuits within the plant for which momentary chatter of a relay could cause serious consequences in the behavior in the safety systems. This could change computed core melt frequencies by an order of magnitude. In particular, no PRA has ever studied the effects of relay chatter or inadvertent circuit breaker trip on the reactor protection system - and this should be a high priority research item.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues (Cont.)					
B. Pipe Failure Due to Soil Failure or Liquification	Large	Large	Small	Large	There is one generic aspect of PWRs which (when treated conservatively) has been found to be quite important, and that is the possibility of interbuilding pipe failure. This applies primarily to PWRs because of their typically tall containment configurations and the fact that all safety and shutdown system piping runs between the auxiliary building and the containment. If soil failure occurs under the containment during rocking motions, large relative displacements between the two buildings could occur, with the resulting possibility of failure of the interbuilding piping. Soil failure can occur due to lack of soil bearing strength (at higher acceleration levels) or more catastrophically, from liquification due to shallow water table and low density sands. Very little is known about liquification, but a number of apartment buildings tipped over in Japan because of it. No PRA has tried to treat this issue - but it could be very important for affected sites, (more than an order of magnitude on core melt).

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues (Cont.)					
C. Relative Displacement Pipe Failures	Large	Large	Large	Small	Failure of pipes between buildings has been treated very conservatively in past PRAs. However, this type of failure is due to relative motion and is thus a strain-controlled phenomenon. The maximum strain and, hence, stress is limited by the relative movement of the structures. Conventional wisdom would hold that the pipes themselves would not break but rather failure would first occur at the supports. The nature of the failure is a function of the exact configuration of the penetrations and the nearest anchor points in the buildings. Two aspects need further investigation. First, analyses should be made in which the possibility of failing nearby supports should be allowed. Secondly, the calculated moments should be compared with nonlinear stress-strain behavior in the piping. It may also be that proper estimation of the strain-induced failure of piping is amenable to experiment, using appropriately sized pipes and appropriate support configurations.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues (Cont.)					
D. Ductility Effects in Structural Failures	Large	Large	Large	Medium	Local structural failures have been found to dominate most of the seismic PRAs to date. In each case, in predicting the failure of these elements, consideration is taken of the ductility available to absorb ground shaking energy. This ductility factor is used in all the fragilities of the structures as well as the fragilities of the major components and pipes to account for nonlinear energy absorption. Examination of the fragilities shows that the ductility factor has a strong effect on the final median value of failure in each case. Thus, the failure probabilities are quite sensitive to the assumed value of the ductility, and computed core melt frequencies could change by an order of magnitude. Although ductility is a widely used concept in failure-prediction methodologies, its background is based on only a limited number of studies by Newmark and his co-workers. Most of these studies dealt with single-degree-of-freedom systems. We recommend examination of typical multi-degree-of-freedom structures and components to evaluate the appropriate ductility factors in a nonlinear fashion.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues (Cont.)					
E. Seismic Fragilities	Medium	Large	Medium	Small	Most actual fragility test data were taken in the late 1960s. Seismic failure data on more recent models of equipment are needed. Better data could change core melt frequencies by 5 to 10.
F. Flat Bottom Storage Tank Fragilities	Medium	Large	Medium	Medium	Free-standing storage tanks (CST, RWST, etc.) often play important roles in a seismic PRA. Their fragilities have been determined by analysis in the past. Tall tanks designed prior to 1980 were designed by a non-conservative method of analysis. Test data are sorely needed on these important components, and core melt frequencies could change by an order of magnitude.
G. Aging Effects on Seismic Fragilities	Medium	Large	Medium	Medium	Another limitation that has not been considered is aging effects. Aging effects on fragilities could be most significant, and could change core melt frequencies by 5 to 10. We are not aware of any data on aging effects on the fragility of nuclear components due to seismic excitation. This is an area that can only be examined by testing. We recommend that such testing be performed on those important components affected by aging.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.I. Hardware Issues (Cont.)					
H. Correlation of Component Fragilities	Large	Medium	Small	Large	Correlation between fragilities of components in the same generic category has been shown to be quite important. To the best of our knowledge, there are no existing data concerning the question of correlated fragilities. Indeed, this is an area that can only be examined experimentally. Fragility correlation (or lack thereof) can change the seismic core melt probability by an order of magnitude.
I. Correlation of Structural Fragilities	Large	Medium	Medium	Medium	A second area of importance concerning fragility correlation is the possibility that fragilities developed from design analysis using the Newmark approach could be correlated by virtue of the fact that they were all developed using the same method. For example, if we tend to underestimate median values of strength for one structure, we are likely to underestimate them for all structures. This would imply a degree of correlation that may not exist. This question needs further investigation, as it could greatly affect most seismic PRAs performed to date (at least an order of magnitude).

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.II. Human Behavior					
A. Operator Response Following Earthquakes	Medium	Medium	Medium	Small	Most PRAs have been conservative by allowing no human recovery actions within 30 minutes after an earthquake. Additional research could demonstrate the possibility of recovery actions following an earthquake and reduce the core melt probabilities computed in past seismic PRAs (by a factor of 5 to 10).
III. Initiating Event Issues					
A. Seismically-induced system transients	Medium	Large	Medium	Small	In seismic PRAs, a transient required reactor shutdown is usually assumed, provided a LOCA is not probable. This is based on the likely failure of ceramic insulators causing an LOSP. Research is needed to delineate the level of earthquake a plant can withstand without requiring shutdown. This could significantly change the computed risk due to earthquakes (by a factor of 5 to 10).
B. Pipe Break Data	Medium	Medium	Medium	Small	Little data exist on pipe break frequencies, especially for small pipes which could be small LOCA initiators. A defensible means of calculating small break LOCAs is needed. This could significantly alter the computed man-REM risk (up to a factor of 5 to 10).

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
C. Fires/Floods Induced by Earthquakes	Small	Medium	Small	Medium	An area that has not been addressed to date is the possibility of earthquake-induced fires and flooding inside the plant. These are secondary failure effects that were outside the scope of most PRAs. However, fires and flooding are important potential consequences of large earthquakes. If fires were to occur inside a plant, there is the possibility that the seismic ground shaking could damage the fire-detection and protection system, in which case the potential consequences of these fires would be greatly enhanced.
S.IV. Issues Related to System Response					
A. Local Amplification of Ground Motion	Large	Medium	Large	Large	At plant sites with 60-150 feet of soil over bedrock, significant amplification of the earthquake motion is anticipated. This results in a higher earthquake hazard curve (up to an order of magnitude) and input ground accelerations increased by up to 200%. At least 30% of the US power plants are affected to some extent by such local site effects. Research is needed to develop consistent and accurate methods for including these local site effects in seismic PRAs. This issue could change the core melt probabilities by over an order of magnitude.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.V. Issues Related to Accident Sequence Analysis					
A. Statistical Approximations	Medium	Medium	Medium	Small	Most seismic PRA results to date are based on the use of lognormal distributions to represent and the fragilities. The lognormal distribution was found to provide a reasonable fit to the calculated responses and the limited fragilities data available seem to be relatively well represented by lognormal distributions. So, in general, we have no reason to believe that such distributions are not appropriate for this type of risk analysis. However, we have not tested this assumption by employing different types of distributions or using pure Monte Carlo approaches to generate risk numbers without specification of a particular form of distribution. Finally, all the seismic risk results are based on the use of an upper-bound approximation in computing the unions of cut sets for the accident sequences. Recent developments show promise of a means of replacing those upper bounds with more realistic estimates. This issue could change core melt frequencies by factors of 2 to 5.

Table A.3. Relative Ranking of Seismic Event Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
S.V. Issues Related to Accident System Response					
A. Local Amplification of Ground Motion	Large	Medium	Large	Large	At plant sites with 60-150 feet of soil over bedrock, significant amplification of the earthquake motion is anticipated. This results in a higher earthquake hazard curve (up to an order of magnitude) and input ground accelerations increased by up to 200%. At least 30% of the US power plants are affected to some extent by such local site effects. Research is needed to develop consistent and accurate methods for including these local site effects in seismic PRAs. This issue could change the core melt probabilities by over an order of magnitude.

Table A.4. Relative Ranking of Flooding Risk Issues that Impact Uncertainty in Core Melt Frequency

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
L.I. Hardware Issues					
A. Component Flood Susceptibility	Small	Large	Medium	Small	Very little data exist on failure of components given a flood level, or spray or high moisture environment.
B. Flood Sensor Failure Rates	Small	Large	Medium	Small	No data available.
L.II. Issues Affecting Human Behavior					
L.III. Initiating Event Issues					
A. Flood Initiator Issues	Large	Large	Large	Small	Data are needed on pipe break and flange leakage occurrence frequencies. Conservative frequencies are currently used. This issue could change the flood risk by an order of magnitude.
L.IV. Issues Related to System Response					
A. Plant State Effects on Flood Risk	Medium	Large	Medium	Medium	The potential for flooding causing core melt is highly dependent on the state of the plant at the time of the flood. Additional research is needed to consistently include this in flooding risk analyses. This issue could change the flood risk by factors of 2 to 5.

Table A.4. Relative Ranking of Flooding Risk Issues that Impact Uncertainty in Core Melt Frequency (Continued)

Issues that can impact uncertainty in core melt frequency	Effect this issue can have on the uncertainty in core melt frequency	Potential for research resolving this issue	Potential for this issue to decrease the core melt frequency	Potential for this issue to increase the core melt frequency	Comments and Discussion
L.V. Issues Related to Accident Sequence Analysis					
None.					

APPENDIX B

CONTAINMENT AND CONSEQUENCE ISSUES

APPENDIX B

CONTAINMENT AND CONSEQUENCE ISSUES

This appendix identifies the containment and consequence issues that impact risk. The issues are evaluated according to the criteria discussed in Chapter 4. These criteria are different in some cases from those used in Chapter 3 and Appendix A. Notes are provided in Table B.1 below that provide some insights into the evaluation process. However, the notes are not intended as a comprehensive or consistent discussion of the issues.

Table B.1. Accident Progression and Consequence Research Issues

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.1. In-vessel issues -accident progression				
a. Natural convection	L	L	L	I. One major concern is uncertainty in primary system failure mode. Affects direct heating and timing and location of fission product and gas release. Steam generator tube ruptures could be important. Other concerns relate to steam cooling and hydrogen production. Another major concern is effects on issue 3C. Concerns not clear for BWRs, but probably less important. II. Efforts are in progress to treat 2 and 3D convection. The problem appears tractable. III. Could make direct heating likely. Anternatively, could result in unfavorable failure mode leading to containment bypass.
b. Rate and magnitude of H_2 production and release	M	M	M	I. The first uncertainty is whether there is sufficient H_2 generated for large burns to be possible. Assuming that is the case, then the rate of release becomes the key question in most situations. H_2 may be released fast enough at the time of vessel failure to lead to a local detonation or overwhelm ignition systems, if present. The amount of steam released with the H_2 is important (coupled steam spike - H_2 burn). The magnitude is important for MK I & II containments in terms of noncondensible overpressure. II. Planned fuel degradation experiments and code development and analysis efforts should provide some insights into the problem. III. If detonations from rapid releases are prevalent, risk could increase, depending on the location and magnitude of the detonations. Steam spike-H_2 burn problems can be bounded.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
c. Likelihood and magnitude of fuel-coolant interactions	M	S	M	I. Alpha-mode failures not included here. Concerns are H_2 production, bottom vessel failure, debris relocation in primary system. Except for debris relocation and possible influence on direct heating, current MARCH calculations are conservative with respect to H_2 production and release, so most uncertainty is in downward direction. II. Mechanisms of fuel-coolant interactions can and are being studied. Easy research largely complete. III. Debris relocation in the primary system could lead to steam tube or isolation valve leakage. May lead to direct heating even for some "l. -pressure" scenarios.
d. Steam explosion induced containment failure	M	M	L	I. Overall risk can be affected, as off-site consequences can be much higher if this occurs. Probability of alpha-mode failure significantly higher than now believed would be important to risk. II. Research can indicate whether or not such failures are physically possible. Experiments may show that large-scale explosions are impossible, thus resolving the issue. However, if they are possible, then uncertainty in their probability will remain large. III. If alpha-mode failures are found to be likely, then risk will be dramatically increased due to early release by direct ejection of debris up into the atmosphere.

R.1.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
e. Recovery potential prior to vessel failure	S	S	M	I. Time window for recovery after core degradation has begun is generally small. Partial ECC operation could become important. II. Limited potential for analysis with codes under development, but no motivation for such calculations. III. Could increase if operator actions are likely to make the situation worse, for example, by causing an alpha-mode failure.
f. Alternate primary system failures	L	M	L	I. How and when the primary system fails during transients will determine how melt is released from the vessel and whether or not direct heating is likely to occur. Also important for LOCAs that are small enough that the system remains pressurized. Failure of steam generator tubes and isolation valves strongly influences off-site release. Importance of high-pressure sequences for BWRs not clear. II. Improved analysis of primary system failure modes appears feasible with current and planned computational methods. The broad scope of this issue makes final resolution difficult, particularly when dealing with plant-to-plant differences. III. Frequent failure of steam generator tubes and isolation valves or failures conducive to direct heating could significantly increase risk.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.1.				
g. Fuel melt progression	L	M	M	I. Geometric progression of core melt not well understood. Mode of slumping, structural failures, channel blockage, etc. will affect H_2 production, fission product release, and fuel-coolant interactions. II. Experiments and code development will be able to provide some insights. III. Impacts on issues noted under I above could increase risk.
h. Debris transport and interactions with lower plenum, RPV penetrations, steam generator tubes and other structures	L	M	M	I. Impacts mode of melt ejection and fission product release paths. II. Combined analysis of fuel melt progression and an examination of the structures present should provide some insights into the mode of vessel failure. III. Certain failure modes could maximize rapid energy addition to containment or cause containment bypass.
R.2. Ex-vessel issues -containment loads				
a. Core-concrete interactions	M	M	S	I. Fission products and aerosols excluded here. Affects whether late containment failures will occur and timing. Issues are combustible and noncondensible gas generation, steam generation, high temperatures, and basemat penetration. Uncertainty in phenomena associated with distribution and coolability of debris. II. Experiments with corium can address gas generation and penetration questions, given the debris configuration. Codes can be modified appropriately, based on the experiments. III. No major upward trend in risk, as this primarily affects late failures.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
b. Radiant heating of concrete above melt	S	S	M	I. Issues include gas generation and concrete ablation, which could lead to melt dilution, affect combustion, and weaken the vessel supports. Somewhat similar to 2a above, but less important. II. Better modeling of radiation heat transfer possible, but probably not warranted. III. Risk could increase if vessel supports are likely to fail, resulting in containment failure when the vessel falls over.
c. Interactions of Debris with shell and structures	S	S	M	I. Could be important for Mk I plants, but failure will occur by other means anyway for many applicable scenarios. II. Debris distribution will be very plant and scenario specific. III. Risk could increase if interactions with shell and penetrations prove likely to cause failure.
d. Gas transport	M	M	L	I. Key issue is formation of detonable mixtures. Most important for ice-condenser and Mk III containments. Also of concern for large dry containments. II. Existing analysis capability should be able to treat the problem, possibly supplemented by limited experiments. Only crude answers are necessary, as detonable clouds must be fairly large to be of concern. III. If detonable mixtures are prevalent, the risk will be substantially increased.

Table B. . Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.2.				
e. Deflagrations	M	M	M	I. Concerns exist for the survival of equipment. Also, there are uncertainties regarding H_2-CO burns and ignition characteristics in the absence of igniters. II. H_2-CO burns can be addressed. H_2-air-steam research largely complete. III. H_2-CO burns or late burns with high baseline pressure and high H_2 concentrations could fail many containments.
f. Flame acceleration and detonations	M	M	L	I. Potential for detonable mixtures to form in ice condensers has been established. One of few ways to fail large dry containments with containment heat removal working. One of the few ways to fail Mk III drywell. II. Mockups of particular geometries could determine if detonations possible. Easy research completed. III. If detonations are prevalent, risk could increase substantially.
g. Diffusion flames	S	M	M	I. No major contribution to overpressure. Some concerns exist with regard to equipment survival (see research area 6) and induced leakage (see research area 7). II. Potential sites can be identified and heating calculations can be performed. Near term experiments planned for MK IIIs. III. Containment leakage or buckling due to temperature effects could increase fission product release. Also, the alteration of accident sequences could result in more severe consequences.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
h. Steam spikes and explosions	S	S	L	I. Conservative treatment of steam spikes shows little threat to containment. Can affect debris dispersal. II. No motivation. III. Potential for large scale ex-vessel steam explosions to knock over the vessel or otherwise cause containment failure.
R.2.				
i. Combined steam spikes and H_2 burns	S	S	L	Simultaneous steam spikes and hydrogen burns following vessel breach can be more severe than either effect when considered individually. I. May be important in both plants with igniters and large dry containments. II. Very little work completed. Scoping analyses feasible. III. Could increase risk for situations where neither H_2 burns nor steam spikes are currently considered important.
j. Effects of aerosols on containment thermalhydraulics	M	M	S	I. Effects of aerosols on heat transfer from melt in cavity may be important for late failures. Direct heating not considered here. Other effects in outer containment can effect timing of late failures. II. Some data possible from core-concrete experiments. III. No apparent potential for upward movement of risk.
k. Debris relocation	M	M	S	I. Relocation (not including direct heating) only important for 2a and 2c above and 21 below. These questions mostly address timing of late failures. II. (Little has been done, but problem is difficult to treat. III. No credible scenarios worse than what's now considered.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
l. Debris coolability	S	S	S	I. Affects potential for and timing of late failures. II. Can be treated reasonably well with existing capabilities given boundary conditions. III. Worst-case scenarios now treated.
m. Direct heating	L	L	L	I. Could cause early failure with high radiological consequences if large amounts of core debris are suspended in the atmosphere. Character and amount of debris that will be suspended is largely unknown. II. Experiments can address the basic mechanisms of the process. Computer models are now nonexistent but could be built to allow treatment of direct heating in integrated containment analyses. III. If significant direct heating is likely for dominant scenarios, then risk estimates will increase significantly due to early failures accompanied by large fission product releases.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.2.				
n. ESF thermalhydraulic performance	S	M	M	I. Fission product removal or equipment survival not considered here. Thermal-hydraulic performances of ESFs reasonably well understood. H_2 igniter performance also fairly well characterized, given boundary conditions. II. Additional H_2 igniter tests are straightforward, if needed. Industry facilities exist to examine performance of other ESFs. III. Potential for ice condenser performance to be degraded due to asymmetric flow and preferential melting on one side. Potential for steam breakthough in BWR suppression pools, given energetic events.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
o. Condensation heat transfer	S	S	S	I. Results are not significantly changed over a wide range of condensation parameters. II. Large scale research is difficult, and small scale information has little applicability. III. Conservative treatments are available now.
p. Containment Release Categories for Seismic Events	M	M	M	One seismic limitation in existing seismic PRAs is the use of the same containment release categories for internal and seismic accident sequences. The internal event containment failure modes may not be appropriate for the seismic situation. In particular, they do not distinguish between the consequences of an accident involving a single piping failure and the case involving failure of both the primary coolant piping and the secondary steam piping. In the latter case, considerably more energy would be released into the containment, and the probability of failure of the containment would be significantly higher. This could change man-REM risk by factors from 5 to 10.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.3. Fission product and aerosol release from fuel and transport in primary system (RCS)				
a. Release of fission products from fuel	M	M	L	Release of volatile fission products (noble gases, cesium and iodine) in-vessel is probably complete and is therefore insensitive to uncertainties in rate. Greatest concern for non-conservatism is with the less volatile materials such as in the WASH-1400 lanthanide group. Existing models do not treat some processes that could have important effects (upward or downward) upon the releases. Some modeling capability exists, but additional experiments will be required to develop and validate. Irreconcilable uncertainties in fuel behavior limit the value of detailed fission product release modeling. Potential for large non-conservatism in current treatment would have to be associated with the enhanced release of low volatility elements such as plutonium.
b. Aerosolization of inert materials	M	M	S	Uncertainty in rate limiting effects of mass transfer. Significant differences exist in treatment of aerosolization of control materials - will they flow away from hot zone? Magnitude of release affects agglomeration and gravitational settling in RCS. A fair amount of separate effects and integral testing will have been completed prior to FY86.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
c. Transport and deposition within primary system	L	M	M	Significant credit has been given in recent studies. Computational technique is reasonably well developed, but data base is sparse. Uncertainties in thermal-hydraulic conditions are large and very important. Question exists as to the adequacy of a control volume approach. Chemical forms and chemical interactions with surfaces and aerosols are not well known. Preliminary agreement between codes and experiments is poor. Processes involved are very complex and difficult to resolve with a high degree of confidence.
R.3.				
d. Chemical transformations of fission products within the primary system	L	M	M	Retention of fission products in RCS can be strongly affected by chemical form and chemical reactions. Growing evidence exists for changing iodine chemistry by interaction with control materials. Chemistry of the lesser studied elements (other than Cs, I and Te) could have surprises. Modeling capability is fairly well developed, but data base is sparse.
e. Release rate and magnitude associated with fuel-coolant interactions	S	M	S	The principal consideration is the potential for enhanced release of ruthenium as predicted in WASH-1400. Cursory evaluation indicates oxidation release of ruthenium is unlikely. If it was felt necessary to better determine the release of ruthenium from a hot corium particle in a steam environment, an experimental program could be developed which would have a high success probability. Increased ruthenium release would make accident consequences more severe, but not dramatically worse.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
f. Release rate and magnitude associated with quench-induced formation	S	M	S	Release would probably be into an aqueous environment and would, as a result, be somewhat mitigated. Threats to containment are not as severe as in uncontrolled accidents (although it is recognized that quench-induced fragmentation can occur in uncontrolled as well as terminated accidents). Further research could improve our understanding.
g. Revolatilization of fission products deposited in-vessel	L	M	L	Significant retention of fission products in-vessel is predicted in recent studies. Considerable decay heat is associated with the volatile fission products. Thermal-hydraulic conditions in the primary system after vessel failure are quite uncertain. Chemical reactions between fission products and primary system or aerosol surfaces are not well known. Delayed release of fission products from the RCS near to or following containment failure could lead to severe consequences. Delayed release of fission products could result in small aerosols that do not deposit rapidly.
R.3.				
h. Resuspension of deposited aerosols at vessel failure	M	M	M	Currently, substantial resuspension is assumed to be improbable but, if it did occur, it could contribute to a large release in an event such as TMLB's with containment failure following shortly after vessel failure. Liquid aerosols (e.g, CsOH) would probably act to prevent resuspension of other aerosols deposited in the same locations. Resuspended aerosols might be large and fall out quickly in the containment.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.4. Fission product and aerosol generation and transport ex-vessel				
a. Release associated with melt ejection	M	M	M	Important issues are aerosol size distributions and combustion of zirconium-containing corium droplets in direct heating scenarios. This could enhance release. Also, substantial releases of volatile species are likely if those are still in the melt at vessel failure time. Melt ejection research could provide some insights. The importance of this issue will depend on whether or not simultaneous containment failure occurs. If simultaneous failure occurs, the release will probably be large in any case, and the details will be less important.
b. Release associated with fuel-coolant interactions (FCI)	S	M	S	Our understanding is that the "oxidation release" of ruthenium assumed in WASH-1400 is now considered very dubious. However, even if it occurs, effects on consequence are small (perhaps 10-30%) compared with other uncertainties concerning both consequences and probabilities of scenarios. Release in FCI scenarios other than "alpha" containment failures seems still less likely to govern consequences. Hence, "S" is assigned to Columns I and III. The aerosol size distribution may be important. The "M" assigned to Column II allows for there being other species involved in addition to ruthenium; if only ruthenium were at issue, an "L" could probably be assigned here. The importance of this issue will depend on whether or not simultaneous containment failure occurs. If simultaneous failure occurs, the release will probably be large in any case, and the details will be less important.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.4.				
c. Release associated with core-concrete interactions (CCI)	L	M	L	Some calculations indicate that it is not possible to rule out large releases of La-group species (including Np and Pu) during core-concrete interactions. If this did occur, implications for consequences of sequences involving containment failure at intermediate or even late times could be serious. Hence, the "L" for Columns I and III. If this issue could be laid to rest, Column III might be dropped to "M" or "S," but it would remain true that core-concrete interactions would likely dominate the releases in any scenario that did not involve early containment failure. (The principal exception to the preceding would be scenarios involving delayed release of volatile species from the RCS due to revolatilization by decay heating.) The aerosol size distribution is important in determining the amount of aerosol that will be suspended in the atmosphere at any particular time. Though the questions involved in this issue can be addressed by research, there are many such questions and an "M" is assigned to Column II.
d. Change in physical and chemical form due to combustion	M	L	S	These issues could strongly affect subsequent behavior within containment; however, a likely time of release is right after the event and the consequences of the release probably would not depend greatly upon change in physical/chemical form (in current consequence codes, anyway). Hence "M" is assigned to Column I and "S" to III. The dominant issue is the effect upon iodine species, which should be resolvable by research; hence the "L" for Column II.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
e. Deposition on structures	S	M	S	Though these processes are important, the dominant mechanisms (diffusiophoresis and gravitational settling) are believed to be reasonably well modeled in current codes, and "S" is therefore assigned to Columns I and III. (Note: in this context, the effect of uncertainty in aerosol dynamic shape factors upon settling rates should be understood to be included in Issue 4j.)
R.4.				
f. Water pool decontamination	M	M	S	In the case of BWR suppression pools, modeling is reasonably good and decontamination factors (DFs) are sufficiently large so that risk is clearly dominated by sequences which bypass the suppression pools; hence "S" assignments would be justified if only suppression pools were considered. However, this issue includes such scenarios as water pools overlying the melt during core-concrete interactions, and "M" seems justified for these cases.
g. Ice-condenser decontamination	M	M	M	Arguments were similar to the suppression pool case, except that it was judged that decontamination is less effective and the modeling is not in as good shape. Hence the "M" assignments.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
h. Effects of sprays on fission product retention	S	L	S	Past results have shown that uncertainties in the effectiveness of containment sprays in collecting aerosolized radionuclides could contribute up to an order of magnitude uncertainty in the source term. Nonetheless, even the most pessimistic estimates yielded a DF of at least an order of magnitude for the containment sprays. Hence, it appears that risk and risk uncertainties will be dominated by sequences in which sprays are ineffective (unavailable, bypassed, have inadequate time to act). "S" was therefore assigned to Columns I and III. There is a common belief that spray modeling is in great shape, which is NOT true: although much theoretical and experimental work has been performed on the basic problem of aerosol collection by falling drops, the representation of this process in current containment codes is quite simplistic. Hence, an "L" is assigned to the likelihood that substantial improvement could be made if serious efforts were undertaken to incorporate the existing knowledge into the codes.
R.4.				
i. Aerosol attenuation and distribution modification along leak path	M	M	S	Here, "leak path" was defined to include other buildings when leakage from containment is into such buildings. An "M" was therefore assigned for Column I.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
j. Aerosol agglomeration	M	M	S	QUEST indicated that, in sequences involving no sprays, uncertainties involving aerosol agglomeration and settling rates (aerosol shape factors, turbulence, etc.) can affect the source term by up to an order of magnitude. The issues involve both modeling uncertainty and uncertainty in the values of the governing parameters. These issues are largely bypassed in "worst case" early containment failures, but are generally important for the scenarios that would be risk-dominating if early containment failure could be precluded; hence "M" is assigned for importance. QUEST showed that the uncertainty is largely in the downward direction, hence the "S" in Column III. Research should prove reasonably fruitful in resolving some of the more important uncertainties, e.g., appropriate values of the shape factors, turbulence levels in containment atmospheres, and modeling of aerosol processes including turbulent and gravitational agglomeration rates. "M" is therefore assigned to Column II.
k. Resuspension associated with energetic events (combustion, depressurization, etc.)	M	M	M	The events considered include catastrophic containment failure and hydrogen burns. Burns might yield substantial resuspension of aerosols deposited on surfaces, and QUEST indicated that significant resuspension upon containment failure is conceivable if and only if failure is catastrophic (openings of 100 sq. m. or more). The issues involved have received only limited study and should yield significantly to appropriate research. Hence, M is assigned to all attributes.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.4.				
l. Liquid-path transport (washdown, etc.)	M	M	S	This issue includes flow to pools and sumps, overflows thereof, transport by pumps when such are operating, etc. Transport of aerosols and radionuclides by liquids arises in several contexts. Washdown of radionuclides on structures, or from pools, is to be assessed (Issue 4k above). Decay heat from radionuclides in pools may generate additional steam sources to the containment, and the timing and severity of hydrogen burns can be rather sensitive to such sources. Aerosol materials in sumps may degrade such ESF components as spray orifices, heat exchangers, and pumps. "M" is assigned for attributes I and II, and "S" for III (the "worst case" early failure scenarios are less likely to be affected by these issues).
m. Chemical reactions other than combustion	M	M	M	Includes radiolysis, organic iodine formation, etc. Conceivably, such effects might convert iodine back into volatile species. Until it is shown that this cannot occur for large fractions of the inventory, "M" is assigned for Columns I and III. "M" is assigned to Column I in order to allow for uncertainties associated with species other than iodine; if only iodine were to be considered an "L" might be justified here.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.5. Health and Economic Consequences				
a. Atmospheric dispersion	M	S	M	Contributes only to the uncertainty in early effects since late effects depend primarily on the magnitude of the release. Further research will probably only add confidence in current results. Self-induced plume rainout may be important.
b. Direct exposure pathways	M	M	M	These are the dominant pathways and include cloudshine, inhalation from the plume and groundshine. The uncertainty is medium for all consequences. Pessimistic assumptions could have a moderate effect on the estimates of latent cancers and economic consequences.
c. Food chain transport	S	S	S	Even with pessimistic assumptions the food chain pathway is a small contributor to risk and uncertainty.
d. Liquid pathways	S	S	M	The most important situations are direct deposition into a water body or washoff of material from land. However, the direct exposure pathways dominate the risk. Basemat meltthrough is not a dominant path. Contamination of an important drinking water supply could be a major problem.
e. Modeling of emergency response	L	M	M	Emergency response is the largest contributor to the uncertainty in early consequences. Some progress can be expected in modeling, but it will always be difficult to quantify institutional and social responses to accidents. Current assumptions and models may be conservative.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
f. Dosimetry	M	S	S	While there is uncertainty in dosimetry, it is not large when compared to that from other sources of uncertainty. The existing computer codes for dosimetric calculations are probably adequate.
g. Health effects	M	S	M	The uncertainty for early health effects and for genetic effects is large and that for latent health effects is moderate. Very little can be done to improve the estimates of latent and genetic effects in humans. Ongoing NRC funded animal studies may improve the estimates of early effects.
R.5.				
h. Economic consequence modeling	S	S	M	Economic effects are dominated by onsite costs. The offsite economic cost uncertainties are generally small compared to those for other types of impact. The loss of the use of a major resource could be important, but is not usually considered.
R.6. Equipment functionability and vulnerability under severe accident conditions*				
a. Detection and monitoring systems	L	M	M	There is considerable uncertainty as to what environments might actually occur and how these systems will respond. Interpretation may be very difficult. Research should be able to work parts of the problem.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
b. Control systems	M	M	M	The ability of control systems to withstand the accident environment is similarly uncertain. The prospects for improving our understanding of parts of the problem are good. Unanticipated behavior and feedbacks between systems could have an impact on accident progression.
c. Essential equipment performance during severe accidents	L	L	L	The operation of various systems is essential for the management of the accident. The ability of equipment associated with these systems to survive in accident environments is a large contributor to risk uncertainty as is the effect of its operation in unanticipated environments. Since little research has been done in this area, it is expected that much progress can be made in a short time. Examples of systems where equipment is important for accident mitigations are: containment heat removal, containment isolation, emergency power, control room environment. Other systems, such as core cooling, will be important for arresting a sequence.

*A variety of phenomena can affect the environment that the equipment is subjected to. Examples of important considerations are: hydrogen burns, diffusion flames, high temperature steam, direct heating, debris impact, etc.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.7. Containment Performance				
a. Response to external missile impact (tornadoes)	S	S	S	I, II, III are scored "S", since probability of containment being struck by a missile is very low and, if hit, the probability of perforating the containment is small.
b. Response to internal missile impact (steam explosions)	S	S	S	I, II, III scored "S", since the probability of containment failure given a missile can be fairly well determined.
c. Response to external detonations (gas clouds, high explosives)	S	S	S	I, II, III scored "S", because the probability of an accidental external detonation is low and furthermore the pressure pulse is a function of the distance of the detonation from the containment and, in most cases, the detonations would occur a distance beyond the site boundary. In addition, the building is designed for a slight overpressure due to tornadoes.
d. Response to airplane impact	S	S	S	II, II, III scored "S", since most power plant sites are not in the path of commercial or military flight paths and hence the probability of an impact is low. There are two sites (Three Mile Island and Seabrook) which are designed for aircraft impact.
e. Transmission of impact loads to equipment	S	S	S	I, II, III Since a through d were scored "S", the transmission of these loads to equipment is also scored low.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
f. Response of containment structure and penetrations to static overpressurization and increased temperature	L	M	L	I is scored "L", since the final engineered barrier for prevention of the release of radioactive material is the containment building and the performance of the containment is not quantified at this time. The amount of leakage before failure and the leakage and failure locations are also uncertain. II is scored "M" rather than "L" because, even though near-term research will reduce the uncertainty, there will remain the uncertainty of the workmanship of the containment and the state of the containment at the time of the accident, e.g., has a valve been left open? III is scored "L" because risk could increase if failures are generally catastrophic, resulting in R.7.
g. Response to dynamic overpressurization and increased temperature (H_2 detonations)	L	M	M	I is considered "L" because the issue has been studied, but the 2- and 3-D effects are difficult to treat. Primarily important for H_2 detonations. Could also be important for steam spikes and explosions. II is scored as a medium in that only a limited number of experiments can be performed to quantify the loading conditions; and due to the randomness of the loading, uncertainties will remain. III is rated as medium since calculations can now provide conservative bounds on the loading conditions and structural calculations will be able to estimate the performance of the containment.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
h. Response to earthquake loading	S	S	M	I & II have been rated small since the containment building is rugged and can withstand an earthquake higher than the Safe Shutdown Earthquake. III is scored medium since pessimistic assumptions, such as the extension of the Charleston, South Carolina, earthquake to the entire east coast region can lead to increases in the estimated risk.
i. Aerosol plugging	M	M	M	I is rated medium, due to effects on timing of release and impact on containment failure. There are also concerns related to leakage in Mk III drywells. II is rated medium in that near term research should be able to somewhat characterize the behavior of aerosol plugging. III is considered medium since the complete absence of plugging or the complete plugging of leak paths will change the sequence of the potential release of material from the containment.
j. Containment bypass	L	S	L	I is given a high rating because, if containment is bypassed, the containment will no longer perform its function, i.e., contain radioactive materials. II is considered as a small because of the variability in system designs and performance, there will remain a residual uncertainty. However, additional study of bypass routes could provide insights. III is rated large since pessimistic assumptions could lead to essentially a condition of complete release.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
R.7.				
k. Secondary containment performance	L	M	S	I is listed as "L" since the amount of retention of material within the secondary containment is essentially unknown, but could reduce the release significantly. Most important for Mk I and II containments. II is considered "M" because almost no work has been done in this area and some improvement appears likely, although plant and scenario-specific questions will make the problem difficult. III is rated "S" because pessimistic assumptions are being considered now, i.e., no retention.
R.8A. Operations				
a. Information needs and procedures for offsite emergency response	M	M	M	The effectiveness of emergency response can affect risk significantly. Existing emergency response plans are based on WASH-1400 methodology. Insights can be gained through a more realistic analysis of severe accidents. Evacuation will always be a difficult issue due to site-specific problems.
b. Information needs and procedures for accident mitigation	M	M	M	TMI2 and other accident experience indicate the importance of the operator. Existing emergency procedures do not extend into severe accident regime. Currently, little credit is taken for operator actions following initiation of core damage. However, taking unplanned action or attempting to follow inapplicable procedures could make the situation worse.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	I Contribution to Overall Uncertainty	II Potential for Reduction of This Uncertainty Through Near Term Research	III Increase in Estimated Risk Associated With Pessimistic Assumptions	Notes and Comments
c. Operator training and performance	L	M	M	TMI2 indicated deficiencies in operator training. The need for technical expertise of operating staff in severe accident behavior is unclear. Operators need more training in responding to realistic severe accident sequences. Computers could provide more assistance to the operators in accident diagnosis and control. There is a possibility that the operator could make the situation worse in some cases.
R.8B. Accident management				
a. Information needs and procedures for offsite emergency response	L		L	Research in other areas should contribute to a knowledge base that will allow more realistic and applicable procedures to be developed.
b. Information needs and procedures for accident mitigation	M		M	Procedures can be developed for those sequences currently not considered and where meaningful actions are possible, i.e., monitoring and control systems of some sort are operational.
c. Operator training and performance	M		M	Improved operator understanding of severe accidents is very important. Since operators currently know very little about what to expect during severe accidents, improvements are possible for many types of accident sequences. However, there is a limit to our ability to eliminate surprises.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	V Need for Additional Development*	Notes and Comments
R.9. Severe accident analysis tools		
a. In-vessel accident progression	M	Modeling efforts in progress will continue for the near term. These models will allow some reduction in the uncertainties. More work is needed for BWRs.
b. Containment loads and thermal-hydraulic response	M	Many of the capabilities needed should be in place in the near term. Improved phenomenological models can be incorporated as they are developed. More work may be required to deal with "local effects."
c. Containment performance	M	Structural analysis codes need additional validation. Leakage models need improvement, but we can currently treat the problem parametrically.
d. Fission product release and transport in-vessel	M	In-pile experiments may lead to better models. Such models are badly needed due to the affect of in-vessel transport on consequences, but residual uncertainties will remain due to the complexity of the problem.
e. Fission product release and transport in containments	M	General capabilities available. Improved models for ESF decontamination factors, particularly for ice condensers, suppression pools and fan coolers, should be incorporated when they are available.
f. Health and economic consequences	S	Unlikely that existing capabilities can be significantly improved.
g. Integrated analysis tools	L	Integrated analysis is important to understand the possible feedbacks and synergisms between various parts of the problem. Also, from a practical standpoint, integrated tools will greatly reduce the time and effort now needed to exercise the "suites" of codes. Work to develop such tools is in progress, but additional development is needed.

*This applies to code development, not code application. Needs for code application are indicated under Item 10.

Table B.1. Accident Progression and Consequence Research Issues (Continued)

Research Issue	VI Need for Study	Notes and Comments
R.10. Safety risk and application studies (including best estimate, sensitivity and uncertainty studies)		
a. Severe accident sequence progression	L	Studies have indicated that in-depth analyses can identify significant differences in sequence behavior from cursory PRA studies. It is necessary to have a realistic understanding of severe accident behavior to develop meaningful accident management strategies. These studies will incorporate the results of much of the research in areas 1 through 9.
b. Source term studies	L	Source term methodology is in a rapid state of advancement. Peer reviewers have identified problems with current methodology. Methods are under development which should be able to resolve many of the problems identified for BMI-2104 methodology. Source term uncertainties are not well characterized; improved characterization of these uncertainties is essential to the proper interpretation.
c. Risk rebaselining and evaluation of alternative severe accident mitigation and regulation options	L	Improved methods are under development which can reduce the uncertainty in risk estimates and cost-benefit tradeoffs. Uncertainties exist in the treatment of generic plant classes or the extension of six plant results to the population of existing plants.

APPENDIX C

SUMMARY OF ACCIDENT SEQUENCES USED FOR SYSTEM RELATED ISSUES

APPENDIX C

SUMMARY OF ACCIDENT SEQUENCES USED FOR SYSTEM RELATED ISSUES

This appendix contains a listing of accident sequences that past PRAs have indicated were important to the core melt frequency caused by internal events.

Table C.1. Important Accident Sequences fr

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)	S R (
I. <u>Hardware Issues</u> a. Uncertainties associated with components having relatively high failure rates in a normal operating environment	DASs involve random hardware failures and/or T&M outages	All	All	T_2MLU T_1MLU T_1MLUO T_1MLUOO'	S_1D S_2D S_3H S_3D T_1B_3MLU T_1MLU T_1MLUO T_2KMLU T_2MQD T_3MLUO T_2MLUO	S_1D S_1H S_2D S_2H S_2HF	A
b. Uncertainties associated with components having low failure rates in a normal environment	DASs involve pipe breaks, reactor vessel rupture, severe bus failures, gross valve ruptures, mechanical control rod insertion failure, etc.	$B(1.66)H_1$ $B(4)H_1$ $T(A3)LD_1$ $T(A3)LD_1C$ $T(A3)LQ\text{-}D_3$ $T(D01)LD_1$ $T(D01)LD_1C$ $T(D01)LD_1YC$ $T(D01)LQ\text{-}D_3$ $T(D02)LD_1YC$	None	B_4H B_4FH B_4D B_4CD	S_1D S_2D S_2FH S_3D S_3FH S_3H V TKMLU	S_1D S_1H S_1HF S_2D S_2H S_2HF V	A A S S S S S V T T
c. Uncertainties associated with components operating in a steam environment	DASs involve a LOCA environment in containment. This occurs following pipe breaks, stuck open relief valves (PWR only), and during feed and bleed core cooling (PWR only)	All	T_1MQD T_1MQH T_1MQFH T_2MQH	All	All	S_1D S_1H S_1HF S_2D S_2H S_2HF	A A S S S S S

Past PRAs

y (… DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs)	Peach Bottom RSS (2 DASs)
	All Except AR S_1R	All	All	T_EUX T_PQUV T_PQUX T_1UX T_MQUX T_TQUX	All	TC TW
	AR S_1R	$T_UQR_BR_A$ $T_PQR_BR_A$	SI $T_{23}C$	None	T_2A	C
J	AR S_1R AI SI S_2R	None	SI		None	None

Table C.1. Important Accident Sequences from Past PRAs

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)	Surry RSS (11 DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs)	Peach Bottom RSS (2 DASs)
d. Uncertainties associated with survival of components operating in a beyond design basis environment	DASs involve components that are operated without their associated air/water cooling system, or are operated in a mode they were not designed (an example of the latter is SRVs passing solid water)	B(1.2)DC B(1.66)H_1 B(4)H_1 T(A3)LQD_3 T(D01)LD_1YC T(D01)LQD_3	T_1MQD T_1MQFH T_1MQH T_2MQH	None	T_2MQ-D T_2MQ-H T_2MQ-FH	S_1D S_2D	S_1D S_2D	$T_{EQ}E$	$T_PQR_BR_A$ $T_KR_BR_A$ $T_PKR_BR_A$ $T_PR_BR_A$ $T_UQR_BR_A$ $T_UR_BR_A$	None	T_EUV T_EUX	None	None
II. <u>Issues affecting Human Behavior</u> a. Uncertainties associated with failure of the operator to correctly follow procedures during accident situations b. Uncertainties associated with failure of operator to perform recovery actions during accident situations	DASs that involve significant interaction between the operators and the mitigating systems. Significant interactions include actions described in the emergency procedures as well as likely recovery actions not described in procedures	All	All except ATWS	B_4H B_4FH B_4CD T_2MLU T_1MLU B_4D	S_2FH S_3FH T_1B_3MLU T_1MLU T_2KMU T_2MLU T_2MQFH T_2MQD S_3H	S_1D S_1H S_2H TML S_2D	AH S_1H TML TMLB' S_2H	S_2R S_1R AR $T_{MF}KPR$ T_TKPR $T_{SI}OR$ $T_{MF}LO$ $T_{MF}EL$ T_REL T_TEL T(LOSP)EL	$T_PQR_BR_A$ $T_PR_BR_A$ TKR_BR_A $T_PKR_BR_A$ $T_UQR_BR_A$ $T_UR_BR_A$	T_1PQE T_1PQI T_1QUV T_1QW $T_{23}C$ $T_{23}PQE$ $T_{23}PQI$ $T_{23}QW$		All	TC TW

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Table C.1. Important Accident Sequences f

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoya RSSMAP (7 DASs)
c. Uncertainties associated with accuracy of core cooling related instrumentation in accident situations	DASs that involve extensive loss of control room instruments due to power failures or involve instrumentation inaccuracies. The latter may occur if the instrument is operating in a design basis environment or is operating significantly away from calibration conditions	All	T_1QD T_1MQH T_1MQFH T_2MQH	All	All	S_1D S_1H S_1HF S_2D S_2H S_2HF
d. Uncertainties associated with system and component restoration errors following test and maintenance activities	DASs that involve unavailable components and systems due to realignment errors following T&M	T(LOP)LDYC	T_2ML T_3ML T_1ML T_1MLOO'	T_1MLU T_2MLU	S_1D S_2D S_3H T_2MQH	S_1HF S_2HF

rom Past PRAs

n	Surry RSS (11 DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs)	Peach Bottom RSS (2 DASs)
	AD AH S_1D S_2C S_2D S_2H	S_2R S_1R AR AI SI $T_{MF}KL$ $T_{MF}KPR$ T_TKPR T_{SI} OR $T_{SI}E$	-	SI	-	-	-
	S_1D S_1C S_2D	AI	None	SI $T_{23}C$ $T_{23}PQI$	T_EUX T_FQUV T_FQUX T_1UX T_TQUX	None	TC TW

Table C.1. Important Accident Sequences from Past PRAs

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)	Surry RSS (11 DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs)	Peach Bottom RSS (2 DASs)
III. Initiating Event Issues a. Uncertainties associated with frequent initiators that occur during power operation	DASs initiated by reactor coolant pump seal LOCA, transient induced LOCAs and transients described in EPRI-2230	B(1.2)D B(1.2)DC T(FIA)K T(LOP)LD_1YC	All	All	T_1B_3MLU T_1MQU T_1MLUO T_2KMU T_2MLU T_2MLUO T_2MQD T_2MQFH T_3MLUO S_3D S_3H	$T_{23}ML$ S_2D S_2H S_2HF	TKQ TKQM TML TMLB' S_2D S_2C S_2H	S_2R $T_{MP}KL$ $T_{MP}KPR$ T_TKPR T_TKL $T_{SI}E$ $T_{MP}EL$ T_REL T_TEL T(LOSP)EL $T_{MP}LO$	All	T_1PQE T_1PQI T_1QUV T_1QW $T_{23}C$ $T_{23}PQE$ $T_{23}PQI$ $T_{23}PQI$	All	All	TC TW
b. Infrequent initiators that occur during power operation (larger LOCAs and support system caused initiating events	DASs initiated by pipe break and valve rupture LOCAs, severe bus failures, and multiple train failures within a support system.	B(1.66)H_1 B(4)H_1 T(A3)LD_1 T(A3)LD_1C T(A3)LQD_3 T(D01)LD_1 T(D01)LD_1C T(D01)LD_1YC T(D01)LQD_3 T(D01)LD_1YC	None	B_4H B_4FH B_4D B_4CD S_3FH S_3H V	S_1D S_2D S_2FH S_3D S_2FH S_2HF V	S_1D S_1H S_1HF S_2D S_2C V S_2H	AD AH S_1D S_1H S_1I	$T_{EQ}E$ AR S_1R AI	None	SI	None	None	None
c. Initiating events that occur during non-full power operation	None of the PRAs analyzed accidents initiated during shutdown	-	-	-	-	-	-	-	-	-	-	-	-

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Table C.1. Important Accident Sequences

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)
IV. Issues Related to System Response a. Uncertainties associated with system/function success criteria	DASs that involve controversial system/function success criteria assumptions. For example, feed & bleed core cooling at CE and $\underline{W}$ plants, ATWS criteria, conservative vendor or FSAR	$T(A3)LQD_3$ $T(DO1)LQD_3$ $T(FIA)KD_1$	T_1ML T_2KML T_2ML T_3ML	B_4D	S_1D S_2D T_2MQ-D T_2KMU	S_1H S_2H S_2HF $T_{23}ML$ S_2D
b. Uncertainties associated with system modeling depth	DASs that were derived with front line system models that did not include all relevant components and support systems	None	Calvert Cliffs used simplified support system models	None	Oconee used simplified support system models	All

rom Past PRAs

Surry RSS (11 DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs)	Peach Bottom RSS (2 DASs)
S_2H AH S_1H TKQM TML TMLB'	$T_{MF}KL$ $T_{MF}KPR$ T_TKPR T_TKL AI $T_{MF}LO$	T_UB T_ABM	$T_{23}C$ T_1QW $T_{23}QW$	None	T_2A	TC
None	None	None	Grand Gulf used simplified support system models	None	None	None

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Table C.1. Important Accident Sequences from Past PRAs

Uncertainty Issues That Impact Core Melt Frequency	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)	Surry RSS (11 DASs)	Zion PSS (17 DASs)	Browns Ferry IREP (8 DASs)	Grand Gulf RSSMAP (9 DASs)	Limerick (7 DASs)	Millstone IREP (12 DASs	Peach Bottom RSS (2 DASs)
c. Uncertainties associated with common cause modeling and data	DASs that were derived with system models that did not include system failure modes due to non "hard-wired" causes. For example, common design error, inadequate procedures, corrosion, etc.	All	All	All	All	All	All	All	All	All	All	All	All
d. Uncertainty associated with accuracy of information used in safety analyses	PRAs that were based primarily on PSAR information and did not have access to as built drawings, procedures, or plant personnel	None	All	None	All	All	None	None	All	All	None	None	None

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Table C.1. Important Accident Sequences fr

Uncertainty Issues That Impact Core Melt Frequency DASs)	Attributes of Dominant Accident Sequences Affected by Issue(s)	ANO-1 IREP (14 DASs)	Calvert Cliffs RSSMAP (9 DASs)	Crystal River 3 IREP (8 DASs)	Oconee 3 RSSMAP (16 DASs)	Sequoyah RSSMAP (7 DASs)	S R (
V. Issues related to accident sequences analysis	DASs that were identified using event trees that do		T_1MQ-FH T_1MQH T_2MQH T_2KML	B_4H B_4FH	T_2KMU S_3FH S_3H	$T_{23}ML$ S_2H S_2HF	S T S T
a. Uncertainties associated with definition of event tree sequences	not reflect curent understanding of plant response. This is primarily a concern for older PRAs.	$T(FIA)KD_1$				TKQM	S T
b. Modeling of interactions between the initiating event and the event tree systems	PRAs that did not conduct a thorough special initiating event search	N/A	CC did not conduct special search.	CR3 did not report power bus search.	Oconee did not conduct special search.	Sequoyah did not conduct special search.	S a e d c a s
c. Uncertainty associated with modeling of interactions among event tree systems	DASs that involve multiple system failures and were derived using a methodology that did not rigorously model interactions due to support systems.	None	Calvert Cliffs used simplified support models.	None	Oconee used simplified support system models.	All	I u h s i a w t

ast PRAs

y	Zion PSS	Browns Ferry IREP	Grand Gulf RSSMAP	Limerick	Millstone IREP	Peach Bottom RSS
DASs)	(17 DASs)	(8 DASs)	(9 DASs)	(7 DASs)	(12 DASs)	(2 DASs)
'	$T_{MP}KL$ $T_{MP}KPR$ $T_{T}KPR$ $T_{T}KL$	All	SI T_1PQI T_1QW $T_{23}C$	None		
		$T_{23}PQI$	None $T_{23}QW$			TW
y r- y not uct pecial ch.	Problems have been found in search. This is docum- ented in NUREG/CR 3300.	N/A	Grand Gulf did not conduct special search.	It is unclear whether Limerick conduct ed a a special search.	N/A	Peach Bottom apparently did not conduct a special search.
s ear Surry ence r- ons e ated.	ZPSS only explic- consider- ed AC inter- actions.	Possibly all-- BF used a meth- odology that identi- fied the most important but not all system inter- actions.	Grand Gulf used simpli- fied support system models.	It is unclear how system inter- actions were treated.	None	It is unclear how system inter- actions were treated.

APPENDIX D

GENERIC SAFETY ISSUES AND RISK ISSUES

APPENDIX D

GENERIC SAFETY ISSUES AND RISK ISSUES

The tables presented below indicate relationships between the risk issues identified in this report and the issues identified in Tables II and III of NUREG-0933. For the purposes of this report, all of the issues in Tables II and III of NUREG-0933 are referred to as "Generic Safety Issues." The identifiers for the Generic Safety Issues are mostly those used in NUREG-0933; however, the Task Action Plan items are identified by "TAP" and the New Generic Issues are identified by "NGI". Table III of NUREG-0933 identifies 506 generic safety issues. Only 237 were used for the cross cut with the risk issues. They include the issues in the columns labeled I, USI, HIGH, MEDIUM, Note 4 and Note 5. The remaining 269 issues were not considered because they are in various stages of resolution, are covered in the other issues, or are ranked as low or drop. No details concerning the particular relationships are presented; we merely show that the issues are related and that the resolution of certain risk issues could impact the resolution of related generic safety issues.

It is particularly important to note that the lists of risk issues and generic safety issues were developed from different perspectives. The perspective for the development of risk issues was discussed in the first four chapters, while the generic safety issues were developed from a variety of perspectives, which deal with such things as operations, worker safety and licensing, as well as risk. Consequently, there are some risk issues which do not match up with generic safety issues and vice versa. The absence of relationships between the two lists of issues in certain areas does not, a priori, imply that these issues are not important. What these tables will allow is the identification of areas where there should be some interrelationships between the two sets of issues and none exist. We will not attempt to make any such judgments at this time, but will provide the information to facilitate such decision-making in the future.

Finally, the reader should recognize that both lists of issues contain issues of varying breadth. The interrelationships between a few broad issues, e.g., TAP A-45, can be just as important or more important than the interrelationships between large numbers of more narrowly defined issues.

Table D-1. Generic Safety Issues Related to Internal Event Issues

RISK ISSUES	ISSUES FROM NUREG-0933
I.A	I.F.1, II.C.1, II.C.2, II.C.4, II.D.1, II.E.6.1, II.F.5, II,K.3, TAP A-3, TAP A-4, TAP A-5, TAP A-30, TAP A-42, TAP B-55, TAP C-11, NGI-23, NGI-70
I.B	I.F.1, II.C.1, II.C.2, II.C.4, II.F.5, TAP A-1, TAP A-2, TAP A-3, TAP A-4, TAP A-5, TAP A-10, TAP A-11, TAP A-12, TAP A-26, TAP A-49, NGI-29, NGI-68, NGI-79
I.C	I.F.1, II.C.1, II.C.2, II.F.5, II,K.3, TAP A-24, NGI-61, NGI-68
I.D	II.C.1, II.C.2, II.D.1, TAP A-24, TAP A-26
II.A	I.A.1.1, I.A.1.2, I.A.1.3, I.A.2.1(1), I.A.2.1(2), I.A.2.1(3), I.A.2.2, I.A.2.3, I.A.2.6(1), I.A.2.6(4), I.A.2.7, I.A.3.1, I.A.3.3, I.A.3.4, I.A.4.2(1), I.A.4.2(4), I.B.1.1(1), I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(5), I.B.1.1(6), I.B.1.1(7), I.C.1(1), I.C.1(2), I.C.1(3), I.C.2, I.C.3, I.C.4, I.C.5, I.C.7, I.C.9, I.D.1, I.D.2, I.D.3, I.D.4, I.D.5(5), I.E.2.2, I.E.3.1, I.G.1, I.G.2, II.C.1, II.C.2, II.C.4, II.K.1(4), II.K.1(6), II.K.1(7), II.K.3, III.A.1.1(1), III.A.1.1(2), III.A.1.2(1), III.A.1.2(2), III.A.1.2(3), TAP B-17
II.B	I.A.4.2(1), I.A.4.2(4), I.C.1(1), I.C.1(2), I.C.1(3), I.D.1, I.D.2, I.D.3, I.D.4, I.D.5(5), I.G.1, I.G.2, II.C.1, II.C.2, II.D.3, II.E.1.2, II.E.2.2, II.F.1, II.F.2, II.H.2, II.K.3
II.C	I.A.2.2, I.A.3.4, I.B.1.1(1), I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(6), I.B.1.1(7), I.C.2, I.C.3, I.C.5, I.C.6, I.C.7, I.C.8, I.C.9, I.D.3, II.C.1, II.C.2, II.C.4, II.K.1(1), II.K.1(2), TAP B-61
II.D	I.A.1.1, I.A.1.2, I.A.1.3, I.A.2.1(1), I.A.2.1(2), I.A.2.1(3), I.A.2.2, I.A.2.3, I.A.2.7, I.A.3.1, I.A.3.3, I.A.3.4, I.A.4.2(1), I.A.4.2(4), I.A.2.6(1), I.A.2.6(4), I.B.1.1(1), I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(6), I.B.1.1(7), I.C.1(1), I.C.1(2), I.C.1(3), I.C.1(4), I.C.2, I.C.3, I.C.4, I.C.5, I.C.6, I.C.7, I.C.8, I.C.9, I.D.1, I.D.2, I.D.3, I.D.4, I.D.5(5), I.G.1, I.G.2, II.B.2, II.B.4, II.C.1, II.C.2, II.D.3, III.A.1.2(1), III.A.1.2(2), III.A.1.2(3), III.A.3.4, III.D.3.3(1), III.3.3(2), III.D.3.3(3), III.D.3.3(4), III.D.3.4, TAP B-17
III.A	I.F.1, II.C.1, II.C.2, II.C.4, II.F.5, TAP A-42, TAP A-44, TAP B-55, NGI-23

Table D-1. Generic Safety Issues Related to Internal Event Issues (Continued)

RISK ISSUES	ISSUES FROM NUREG-0933
III.B	I.F.1, II.C.1, II.C.2, II.C.4, II.F.5, TAP A-3, TAP A-4, TAP A-5, TAP A-9, TAP A-10, TAP A-11, NGI-51, NGI-61, NGI-65, NGI-68
III.C	I.F.1, II.C.4, II.F.5, NGI-51
IV.A	II.k.1(6), II.k.1(7), II.k.2, II.k.3, II.B.1, II.C.1, II.C.2, II.E.1.1, II.E.1.2, II.E.2.2, II.E.3.1, II.G.1, II.k.1(5), TAP A-9, NGI-23
IV.B	II.C.1, II.C.2, II.E.1.1, TAP A-9, TAP A-30, TAP A-47
IV.C	II.C.1, II.C.2, II.C.4, II.E.1.1, TAP A-9
IV.D	II.C.1, II.C.2, II.E.1.1, TAP A-9, TAP A-30, TAP A-44
IV.E	II.C.1, II.C.2, II.E.1.1, TAP A-9, TAP A-30, TAP A-31, TAP A-44, TAP A-47, TAP B-55, NGI-12, NGI-68
V.A	II.C.1, II.C.2, TAP A-3, TAP A-4, TAP A-5
V.B	II.C.1, II.C.2, II.C.4, II.E.4.2, TAP A-3, TAP A-4, TAP A-5, TAP A-30, TAP A-31, TAP A-44, TAP A-47, NGI-12, NGI-23, NGI-61, NGI-65, NGI-68, NGI-70
V.C	II.C.1, II.C.2, II.C.4, II.E.4.2, II.k.3, TAP A-1, TAP A-2, TAP A-3, TAP A-30, TAP A-31, TAP A-43, TAP A-44, TAP A-47, TAP B-55, NGI-12, NGI-70

Table D-2. Generic Safety Issues Related to Fire Issues

FIRE ISSUES	ISSUES FROM NUREG-0933
F.I.i	I.D.1, I.D.4, II.E.3.2, II.E.3.3, TAP A-29, TAP A-45, NGI-57, NGI-81, NGI-83
F.I.ii	II.E.3.2, II.E.3.3, TAP A-45, NGI-81
F.I.iii	II.E.3.2, II.E.3.3, TAP A-45, NGI-81
F.II.i	TAP A-40, TAP A-41, TAP A-45
F.II.ii	TAP A-29, TAP A-45
F.II.iii	TAP A-45
F.III.i	TAP A-45
F.III.ii	TAP A-40, NGI-77
F.IV.i	TAP A-45, NGI-57, NGI-81
F.IV.ii	TAP A-40, TAP A-41, TAP A-45, TAP A-46, NGI-57, NGI-77
F.IV.iii	TAP A-40, TAP A-41, NGI-57
F.V.i	TAP A-29, NGI-1, NGI-83
F.V.ii	I.D.4, TAP A-24, NGI-57
F.VI.i	I.D.4, TAP A-24, TAP A-29, TAP A-45, NGI-57
F.VI.ii	I.D.4, TAP A-24, TAP A-29, TAP A-45, NGI-57
F.VII.i	I.D.1, I.D.4, II.E.3.2, II.E.3.3, II.F.5, TAP A-17, TAP A-45, TAP A-47, NGI-83
F.VII.ii	I.D.1, I.D.4, II.E.3.2, II.E.3.3, II.F.5, TAP A-17, TAP A-29, TAP A-30, TAP A-45, TAP A-47, NGI-83
F.VII.iii	I.D.1, I.D.4, II.E.3.2, II.E.3.3, II.F.5, TAP A-17, TAP A-29, TAP A-30, TAP A-45, TAP A-47
F.VII.iv	TAP A-40, TAP A-41, TAP A-46, NGI-57, NGI-77

Table D-3. Generic Safety Issues Related to Flooding Risk Issues

FLOOD ISSUE	ISSUES FROM NUREG-0933
L.I.A	II.C.1, II.C.2, II.C.4, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-21, TAP A-24, TAP A-30, TAP A-31, TAP A-45, TAP B-4
L.I.B	II.C.1, II.C.2, II.C.4, II.F.5, TAP A-17, TAP A-24, TAP A-45, TAP B-4
L.III.A	II.C.1, II.C.2, TAP A-18, TAP A-45, TAP B-4, TAP B-6
L.IV.A	I.A.2.1, I.C.1, II.C.1, II.C.2, II.C.3, II.E.1.1, TAP A-17, TAP A-21, TAP A-45,

Table D-4. Generic Safety Issues Related to Seismic Risk Issues

SEISMIC ISSUE	ISSUES FROM NUREG-0933
S.I.A	II.C.1, II.C.2, II.C.3, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-56, TAP B-57, NGI-55
S.I.B	II.C.1, II.C.2, II.C.3, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, TAP A-17, TAP A-18, TAP A-40, TAP A-41, TAP A-45, TAP B-5, TAP B-50
S.I.C	II.C.1, II.C.2, II.C.3, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, TAP A-18, TAP A-40, TAP A-41, TAP A-45, TAP B-6, TAP B-50
S.I.D	II.C.1, II.C.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, TAP A-12, TAP A-18, TAP A-22, TAP A-40, TAP A-41, TAP A-45, TAP A-46, TAP B-4, TAP B-5, TAP B-6, TAP B-50, TAP B-51
S.I.E	II.C.1, II.C.2, II.C.3, II.D.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-21, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-50, TAP B-51, TAP B-52, TAP B-55, TAP B-56, TAP B-57, NGI-29, NGI-55, NGI-70
S.I.F	II.C.1, II.C.2, II.C.3, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, TAP A-31, TAP A-40, TAP A-41, TAP A-45, TAP A-46, TAP B-4, TAP B-50, TAP B-51
S.I.G	II.C.1, II.C.2, II.C.3, II.D.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-52, TAP B-56, TAP B-57, NGI-29
S.I.H	II.C.1, II.C.2, II.C.3, II.D.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, II.G.1, II.K.1(5), TAP A-17, TAP A-21, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP A-46, TAP A-47, TAP B-4, TAP B-24, TAP B-55, TAP B-56, TAP B-57, NGI-29, NGI-55, NGI-70
S.I.I	II.C.1, II.C.2, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, TAP A-12, TAP A-22, TAP A-40, TAP A-41, TAP A-45, TAP A-46, TAP B-4, TAP B-5, TAP B-6, TAP B-51, TAP B-56
S.II.A	I.A.2.1, II.C.1, II.C.2, II.E.1.1, II.E.3.3, II.3.4, TAP A-40, TAP A-41, TAP A-45, TAP B-4

Table D-4. Generic Safety Issues Related to Seismic Risk Issues (Continued)

SEISMIC ISSUE	ISSUES FROM NUREG-0933
S.III.A	I.A.4.4, II.C.1, II.C.2, II.E.1.1, II.E.3.3, II.E.3.4, TAP A-24, TAP A-30, TAP A-31, TAP A-40, TAP A-41, TAP A-44, TAP A-45, TAP B-4, TAP B-24, TAP B-56, TAP B-57
S.III.B	I.C.1, II.C.2, II.C.4, II.E.1.1, II.E.2.1, II.E.3.3, II.E.3.4, II.F.5, TAP A-12, TAP A-18, TAP A-40, TAP A-41, TAP A-45, TAP A-46, TAP B-4, TAP B-6, TAP B-24, TAP B-51, NGI-86, NGI-89
S.III.C	I.C.1, II.C.2, II.C.3, II.E.1.1, II.E.3.3, II.E.3.4, TAP A-17, TAP A-40, TAP A-41, TAP A-45, TAP B-4
S.IV.A	I.C.1, II.C.2, II.E.1.1, II.E.3.3, II.E.3.4, TAP A-12, TAP A-31, TAP A-40, TAP A-41, TAP A-45, TAP B-4, TAP B-52
S.V.A	II.C.1, II.C.2, II.E.2.3, II.E.3.3, II.E.3.4, TAP A-40, TAP A-41, TAP A-45

Table D-5. Generic Safety Issues Related to Containment/Consequence Risk Issues

RISK ISSUES	ISSUES FROM NUREG-0933
R.1.A	II.B.1
R.1.B	II.B.1, II.B.5(1), II.B.5(2), II.B.7, II.B.8
R.1.C	II.B.5(2), II.B.7, II.B.8
R.1.D	II.B.5(2)
R.1.E	II.B.5(1), II.B.5(2), II.B.8
R.1.F	II.B.1, II.B.5(2), TAP A-1
R.1.G	II.B.5(1), II.B.5(2)
R.1.H	II.B.5(2), TAP A-11
R.2.A	II.B.5(2), II.B.7
R.2.B	
R.2.C	II.B.5(2)
R.2.D	II.B.1, II.B.5(3), II.B.7, II.B.8, TAP A-48, TAP B-14
R.2.E	II.B.5(3), II.B.7, II.B.8, TAP A-48
R.2.F	II.B.5(3), II.B.7, II.B.8, TAP A-48
R.2.G	II.B.1, II.B.5(3), II.B.7, II.B.8, TAP A-48
R.2.H	II.B.5(2)
R.2.I	II.B.5(2), II.B.5(3), II.B.7, TAP A-48
R.2.J	
R.2.K	II.B.5(2)
R.2.L	II.B.5(2)
R.2.M	II.B.1, II.B.5(2)
R.2.N	II.B.5(3), II.B.7, II.B.8, TAP A-8, TAP A-39, TAP A-43, TAP A-48, TAP B-10, TAP B-18, TAP B-54
R.2.O	
R.2.P	II.A.2.1, I.A.2.6, I.A.4.4, II.C.1, II.C.2, II.E.3.3, II.E.3.4, TAP A-22, TAP A-40, TAP A-41, TAP A-45
R.3.A	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.B	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.C	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.D	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.E	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.F	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.G	II.A.1, II.B.5(1), II.B.8, II.H.3
R.3.H	II.A.1, II.B.5(1), II.B.8
R.4.A	II.A.1, II.B.5(2), II.B.8
R.4.B	II.A.1, II.B.5(2), II.B.8
R.4.C	II.A.1, II.B.5(2), II.B.8
R.4.D	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.E	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.F	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.G	II.A.1, II.B.5(2), II.B.8
R.4.H	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.I	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.J	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.K	II.A.1, II.B.5(2), II.B.8, II.H.3
R.4.L	II.A.1, II.B.5(2), II.B.8, II.H.3

Table D-5. Generic Safety Issues Related to Containment/ Consequence Risk Issues (Continued)

RISK ISSUES	ISSUES FROM NUREG-0933
R.4.M	II.A.1 II.B.5(2), II.B.8, II.H.3
R.5.A	II.A, IV.E.5
R.5.B	II.A, IV.E.5
R.5.C	
R.5.D	
R.5.E	II.A, III.A, IV.E.5, NGI-88
R.5.F	II.A, IV.E.5
R.5.G	II.A, IV.E.5
R.5.H	II.A, IV.E.5
R.6.A	TAP A-2, TAP A-8, TAP A-24, TAP A-30, TAP A-34, TAP A-48, TAP B-50, TAP B-76, TAP B-85, TAP B-87, TAP B-91, TAP B-93, II.B.2, II.D.3, II.E.1.2, II.F.1, II.6.1
R.6.B	TAP A-2, TAP A-8, TAP A-24, TAP A-30, TAP A-47, TAP A-48, TAP B-50, TAP B-55, TAP B-76, II.B.2, II.E.1.2, II.G.1, II.K.1(5), II.K.1(6), II.K.1(7)
R.6.C	TAP A-2, TAP A-8, TAP A-24, TAP A-30, TAP A-48, TAP B-32, TAP B-50, TAP B-56, TAP B-58, TAP B-21, TAP B-41, TAP B-49, TAP B-55, TAP B-70, TAP B-71, II.B.2, II.E.3.1
R.7.A	TAP A-32, TAP A-38
R.7.B	TAP A-32
R.7.C	
R.7.D	
R.7.E	
R.7.F	TAP A-23, TAP B-5, TAP B-9, TAP B-10, TAP B-26, TAP B-54
R.7.G	II.B.5(3)
R.7.H	TAP A-41
R.7.I	
R.7.J	II.E.4.3
R.8A.A	TAP A-34, TAP A-88, HF.01.5.1
R.8A.B	TAP B-17, TAP B-83, HF.01.1.1, HF.01.1.3, HF.01.4.1, HF.01.5.1, HF.01.5.2
R.8A.C	I.A.21(1), I.A.2.1(2), I.A.2.1(3), I.A.2.2, I.A.2.3, I.A.2.6(1), I.A.2.6(3), I.A.2.6(4), I.A.2.7, I.A.3.1, I.A.3.3, I.A.3.4, I.A.4.2(1), I.A.4.2(4), I.B.1.1, I.B.1.1(2), I.B.1.1(3), I.B.1.1(4), I.B.1.1(6), I.B.1.1(7), I.B.1.2(1), I.B.1.2(3), I.B.1.2(3), I.E.8, II.B.4, IV.E.5, HF.01.1.2, HF.01.1.3, HF.01.1.5, HF.01.2.1, HF.01.2.2, HF.01.3.1, HF.01.3.2, HF.01.4.2, HF.01.5.1
R.8B.A	II.F.1, III.A.1.1(1), III.A.1.1(2), III.A.1.2(1), TAP A-34
R.8B.B	I.C.1(1), I.C.1(2), I.C.1(3), I.C.3, I.C.4, I.C.5 I.C.6, I.C.7, I.C.8, I.C.9, I.D.1, I.D.2, I.D.3, I.D.4 II.B.8, II.F.1, II.K.1(4), II.K.1(5), II.K.1(6), II.K.1(7), TAP B-17, TAP B-24, TAP B-83

Table D-5. Generic Safety Issues Related to Containment/ Consequence Risk Issues (Continued)

RISK ISSUES	ISSUES FROM NUREG-0933
R.8B.C	II.K.1(4)
R.9.A	II.B.1, II.B.5(1), II.B.5(2), II.B.8, TAP A-9
R.9.B	II.B.1, II.B.5(2), II.B.5(3), II.B.7, II.B.8, TAP A-8, TAP A-39, TAP A-43, TAP A-48, TAP B-10, TAP B-11, TAP B-14, TAP B-20, TAP B-54
R.9.C	II.B.5(3), II.B.8, TAP B-26
R.9.D	II.B.1, II.B.5(1) II.B.5(2)
R.9.E	II.B.1
R.9.F	II.A.1, II.A.2, II.B.6, TAP B-72
R.9.G	II.A.1, II.A.2, II.B.1, II.B.5(1), II.B.5(2), II.B.5(3), II.B.6, II.B.7, II.B.8, TAP A-9, TAP A-39, TAP A-43, TAP A-48, TAP B-10, TAP B-11, TAP B-14, TAP B-20, TAP B-54
R.10.A	II.B.1, II.B.5(1), II.B.5(2), II.B.5(3), II.B.7, II.B.8, TAP A-9, TAP A-48, TAP B-10, TAP B-14, TAP B-54
R.10.B	II.A.1, II.A.2, II.B.5(2), II.B.5(3), II.B.6, II.B.7, TAP A-48, TAP B-10, TAP B-54
R.10.C	II.A.1, II.A.2, II.B.1, II.B.5(2), II.B.5(3), II.B.6, II.B.7, II.B.8, TAP A-9, TAP A-48, TAP B-10

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This report presents the results of an effort to identify and rank reactor safety and risk issues identified from past Probabilistic Risk Assessments (PRAs) and other safety analyses. Because of the varied scope of these analyses, the list of issues may be incomplete. Nevertheless, those studies comprised ordered anlayses to whatever their respective depths; hence, they warranted scrutiny for whatever insights they could reveal with respect to issue importance. The top ranked issues in terms of their contribution to the uncertainty in risk are described in some detail. All of these risk issues are compared to the "generic safety issues" for completeness and omissions.

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