Probabilistic Safety Assessment for the Davis-Besse Nuclear Power Station

Summary Report

prepared by

Probabilistic Risk Assessment Unit

Design Basis Engineering

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Contents

Contents (continued)

Tables

Illustrations

Executive Summary

A major update of the probabilistic safety assessment (PSA) for Davis-Besse has recently been completed. The update of the front-end portion of the PSA was described in a previous version of this report, produced in February of this year. The front-end analysis is that part of the PSA in which the sequences of events that could lead to core damage are identified and their frequencies are estimated. Since that time, updating of the back-end analysis has also been completed. The back-end analysis addresses the effects on containment of the severe accidents evaluated in the front-end analysis, and determines the potential for and severity of radionuclide releases that might result.

This report supercedes the February version. It summarizes the reasons for the update efforts and the technical activities required to accomplish them. The updated results from both the front-end and back-end analyses are described in some detail.

Objectives of the PSA Update

The current PSA is the result of a major update of the one performed to satisfy the US Nuclear Regulatory Commission's (NRC's) request for an individual plant examination (IPE) to identify severe accident vulnerabilities. The assessment for the IPE was completed in 1993. To complete the IPE, it was necessary to freeze the plant design early in the process to provide a well-defined baseline. Thus, the risk profile as it was developed for the IPE did not take into account changes made after the freeze date. Furthermore, the tools for PSA have continued to improve since the performance of the IPE. These improved tools simplify the construction and quantification of the PSA model and help to makes the PSA more useful for various applications. Therefore, the objectives of this update included the following:

- To ensure that the models comprising the PSA accurately reflected the current plant, including its physical configuration, operating procedures, maintenance practices, etc.;
- To take into account more recent operating experience with respect to the frequency of plant transients and the failure rates of potentially important components;
- To make the PSA more useful for risk-based applications, such as supporting the implementation of the Maintenance Rule, by modeling some systems in more detail than had been done for the IPE and by improving the ability to evaluate the models and to perform specific investigations;
- To eliminate errors uncovered subsequent to the IPE submittal;
- To provide a more complete set of documentation to aid in subsequent applications and to make later updates more tractable; and
- To train current PSA staff members who were not involved in the IPE effort to perform probabilistic evaluations and to apply them to support the operation of Davis-Besse.

Methodology for the Davis-Besse PSA

The PSA is comprised of two major areas of analysis:

(1) The identification of sequences of events that could lead to core damage and the estimation of their frequencies of occurrence (the front-end analysis); and

(2) The evaluation of the potential response of containment to these sequences, with emphasis on the possible modes of containment failure and the corresponding radionuclide source terms (the back-end analysis).

The overall methodology for the front-end analysis can be characterized as the "linked fault-tree" approach. Under this approach, set of event trees was developed for each general type of plant upset that could initiate an accident sequence (i.e., for various sizes of loss-of-coolant accidents and for transients that would not necessarily involve a breach in the RCS). These event trees allow the safety functions that must be achieved to keep the core cooled to be organized in a way that defines accident sequences that lead to core damage. The potential for failure of each of the safety functions is defined through the construction of a fault tree. The fault trees carry the modeling from the level of safety functions down to the basic hardware failures and human actions (or inactions) that can contribute to a core-damage sequence. Using reliability data bases assembled from a review of operating experience both at Davis-Besse and on an industry-wide basis, the integrated models can be evaluated to yield estimates of the frequencies of the core-damage accidents of concern.

The back-end analysis employs both deterministic and probabilistic analysis tools to follow the progression of the core-damage accidents. Computer codes were used to simulate the meltdown of the core, the failure of the reactor vessel due to contact with molten core materials, and the transport and interactions of core debris in the containment. Because of the large uncertainties associated with the progression of a core-damage accident, these deterministic calculations were supplemented with assessments that considered the potential for phenomena different from or more severe than those treated in the computer codes. The results of this part of the analysis include an assessment of the potential for a variety of containment failure modes for each type of core-damage sequence, and an estimate of the magnitude of the radionuclide release that would be associated with each.

Results and Conclusions

This section provides a summary of the results and insights gained during the performance of the PSA for Davis-Besse. The overall conclusion is consistent with that from the IPE: based on an assessment in which methods that reflect the state of the art in PSA technology were applied in a careful and conscientious manner, no vulnerabilities to severe accidents were identified for Davis-Besse. Although there is no widely accepted definition of a condition that would constitute a vulnerability, the following functional definition has been used for Davis-Besse[1]:

- Any feature of the plant design or operating practice that leads to an unacceptably high frequency of one or more core-damage sequences or that implies an unusually large conditional probability for a serious release from containment given core damage; or
- Any single feature that contributes a large fraction of the frequency of core damage or of serious release, even if the overall frequencies are judged to be generally acceptable.

Neither of these conditions was found to be present for Davis-Besse. As described in the sections that follow, neither the core-damage frequency nor the frequency of serious releases is high relative to risk estimates generally obtained for other plants. Although a relatively small number of sequences is responsible for most of the frequency of core damage, these sequences are comprised of many different contributing factors, none of which is disproportionately large. Moreover, as will be seen, the dominance of individual sequences is less marked than was the case at the time of the IPE.

In addition to a discussion of the results, a summary of the plant features that were found to be among those that contributed most to the results, as well as those that tended to limit the frequencies of certain accidents, is provided below. Because the current update only addressed the front-end portion of the PSA, the discussion of results provided here is limited to that portion as well. Results and insights from the back-end analyses can be found in Section 4.2 of Part 1 of the IPE Submittal.

Results from the Front-End Analyses

The core damage frequency is estimated to be 1.6×10^{-5} per year. Table E1 summarizes the contributions to this total from the general categories of events considered in the PSA. The table also provides a comparison to the results from the PSA performed for the IPE. Note that the overall core-damage frequency and the frequency for nearly every category was assessed to be lower in the current update than in the IPE. Although the potential for core-damage accidents initiated by a transient still contributes more than half the total, both that contribution and the frequency of core damage associated with transients reflect a significant decrease relative to the IPE results. The other event groups generally exhibit more modest reductions. Only the potential for core damage due to an internal flood was assessed to be higher than in the IPE. This is the result of a more careful evaluation of floods from the condenser circulating water system than was performed for the IPE.

[1]The term “severe-accident vulnerability” as it is used here originated in the NRC’s request of each plant that an IPE be conducted. The NRC chose not to define what constituted a vulnerability, leaving the definition instead to each individual utility. The definition provided here is the one that was developed by Toledo Edison and applied for the Davis-Besse IPE.

The frequency of core damage due to nearly fifty sequences was evaluated in the updated PSA. Most of these sequences were found to constitute relatively small contributors to core damage. The sequences whose contributions were more significant are listed in Table E2. As Table E2 indicates, two types of functional sequences were particularly important and together contribute almost half of the core-damage frequency.

The first of these was the potential for a loss of seal cooling for the reactor coolant pumps (RCPs), leading to a small LOCA due to failure of the seals, followed by failure to maintain adequate RCS inventory (i.e., failure of safety injection). A seal LOCA would only occur if the RCPs were permitted to continue operating after seal cooling was lost or substantially degraded. Seal cooling is normally supplied by injection from the makeup system and by the component cooling water (CCW) system. The latter removes heat from the thermal barrier coolers for the seals. Loss of both means of cooling, or failure to maintain adequate seal return flow, could lead to degradation of the three stages of the RCP seals.

The potential for failures of support systems played a dominant role for this type of sequence. Component cooling water is required for cooling of the pumps in both the makeup and high pressure injection (HPI) systems. Thus, if the CCW system were to fail, both sources of seal cooling (i.e., CCW and seal injection from makeup) would be lost, and there would be no means for safety injection at high pressure. Loss of cooling by the CCW system could also result from loss of the service water system, which serves as the heat sink for the CCW system. Both of these systems have significant redundancy, but they could be subject to common-cause failures. There are opportunities for the operators to restore cooling flow or to trip the RCPs to prevent the seal LOCA; failures of these actions are important elements of the cut sets as well.

The second type of transient-initiated sequence that was a significant contributor to the core-damage frequency involves loss of heat removal via the steam generators and failure of makeup/HPI cooling. This functional sequence would entail a loss of main feedwater, either as an initiating event or as a consequence of another initiating event. Both of the turbine-driven auxiliary feedwater (AFW) pumps, the motor-driven feed pump, and the startup feed pump would have to be unavailable to supply backup flow to the steam generators. Finally, makeup/HPI cooling would have to fail as well for core damage to result.

Although many different types of minimal cut sets contribute to this functional sequence, scenarios initiated by a loss of feedwater, a plant trip or a loss of offsite power account for a large portion of the total core damage frequency. The large reduction in plant trip frequency and, in particular, in the frequency of trips due to loss of main feedwater experienced since the IPE freeze date are responsible for much of the decrease in the contribution of this sequence as shown in Table E2, and in the overall core-damage frequency as well.

Table E1. Summary of IPE and Updated Sequence Core Damage Frequencies

Initiating Event Category	IPE Results		Update Results	
	Annual CDF	Percent of Total CDF	Annual CDF	Percent of Total CDF
Transients	5.7×10^{-5}	86.3	9.8×10^{-6}	60.3
LOCAs	5.5×10^{-6}	8.3	3.6×10^{-6}	22.0
Internal floods	2.0×10^{-6}	3.0	2.1×10^{-6}	13.2
Interfacing-system LOCAs	8.8×10^{-7}	1.3	4.8×10^{-7}	3.0
Steam generator tube rupture	4.6×10^{-7}	0.7	2.4×10^{-7}	1.5
Total core-damage frequency	**6.5×10^{-5}**		**1.6×10^{-5}**	

Table E2. Sequences Contributing Most to Core-Damage Frequency

Sequence	IPE Results		Update Results	
	Annual CDF	Percent of Total CDF	Annual CDF	Percent of Total CDF
Transient initiating event with small LOCA induced by failure of RCP seal cooling, and failure of safety injection	1.4×10^{-5}	21.2	3.7×10^{-6}	22.8
Transient initiating event with a total loss of feedwater and failure of makeup/HPI cooling	3.5×10^{-5}	53.0	3.3×10^{-6}	20.3
Transient with RCP seal LOCA and failure of long-term cooling	4.3×10^{-6}	6.5	1.9×10^{-6}	11.7
Internal flood with RCP seal LOCA and failure of safety injection	1.9×10^{-6}	2.9	1.7×10^{-6}	10.5
Medium LOCA initiating event with failure of low pressure recirculation	1.6×10^{-6}	2.4	1.3×10^{-6}	8.0
Large LOCA initiating event with failure of low pressure recirculation	8.7×10^{-7}	1.3	6.7×10^{-7}	4.1
Small LOCA initiating event with failure of long-term cooling via DHR or recirculation from sump	1.5×10^{-6}	2.9	5.4×10^{-7}	3.3
Total core-damage frequency for important sequences	**5.9×10^{-5}**	**90.2**	**1.3×10^{-5}**	**80.7**

Other types of small LOCAs have been found to be important at some PWRs. The frequency of core damage due to small LOCAs is relatively small for Davis-Besse for a variety of reasons, but partly because both the HPI and makeup systems can provide adequate control of RCS inventory, offering a degree of redundancy and diversity. In the long term, it would generally be possible to cool down to conditions at which core cooling could be provided by the decay heat removal (DHR) system, or high pressure recirculation could be established. For medium and large LOCAs, the dominant contributors were primarily common-cause failures or failures of the operating staff to establish recirculation. No individual failure modes were found to be particularly important.

Core-damage sequences initiated by steam generator tube ruptures (SGTRs) were also assessed to be relatively low in frequency. The primary reason for this was the very long time available for response in most cases. In general, the emergency procedure would lead to early cooldown to the point at which the steam generator containing the broken tube could be isolated, effectively terminating the leakage from the RCS. Even if this could not be accomplished for some reason, the borated water storage tank (BWST), which is the supply source for the injection systems, normally contains nearly 500,000 gallons of water. For most scenarios, the lowering of RCS pressure would cause the leak rate to be reduced to the point at which this volume would last for a period of days. This would afford ample time for response and recovery of affected equipment.

Another type of contributor is due to interfacing-systems LOCAs. These involve failures outside containment of systems that connect with the RCS. Although these sequences have relatively low frequencies, there are important because they entail a LOCA, the failure of emergency core cooling, and a breach of the containment. The frequencies of these LOCAs were assessed to be dominated by scenarios that would involve successful injection until the BWST contents were depleted. Since the break would be outside containment, no water would collect in the emergency sump to be available for recirculation when the BWST was depleted. For these cases, there would be significant time for operator action to isolate the breaks. The dominant scenario was one in which it was postulated that an operator error of commission could lead to premature entry into shutdown cooling while the RCS was still at high pressure. This scenario, originally postulated by the NRC in a generic study, was reevaluated for applicability to Davis-Besse. There remains significant uncertainty with respect to whether or not it is credible for such an error to be made while RCS pressure is high enough to threaten the integrity of the DHR system.

Internal flooding was also investigated in detail. Four areas were identified to be susceptible to flooding that could be important with respect to core damage. These areas are the CCW pump room, the service water pump or valve rooms, the rooms housing the pumps for the emergency core cooling systems (ECCS), and the turbine building. Flooding in the CCW or service water rooms is responsible for the largest fraction of the core damage frequency associated with internal flooding. These floods are important because they can cause a loss of all CCW and, therefore, contribute to sequences involving RCP seal LOCAs. Flooding in the turbine building could result from a large break in the condenser circulating water system. If the flood were not isolated it could lead to a total loss of feedwater and, potentially the partial loss of ac power. The engineered features provided to limit the effects of flooding were effective in reducing the contribution to core damage from this source.

In addition to examining the sequences that contribute to core-damage frequency, it can be useful to identify the systems that are most important. One measure of importance can be determined by evaluating the effect on core-damage frequency if the system is assumed to have perfect reliability. This allows the systems to be ranked according to their contributions to overall core-damage frequency (i.e., the larger the impact on core-damage frequency if the system were perfect, the larger the contribution to the base-case core damage frequency due to the failure of that system). This is a common importance measure, and is referred to as Fussell-Vesely importance. The most important systems are identified in Figure E1.

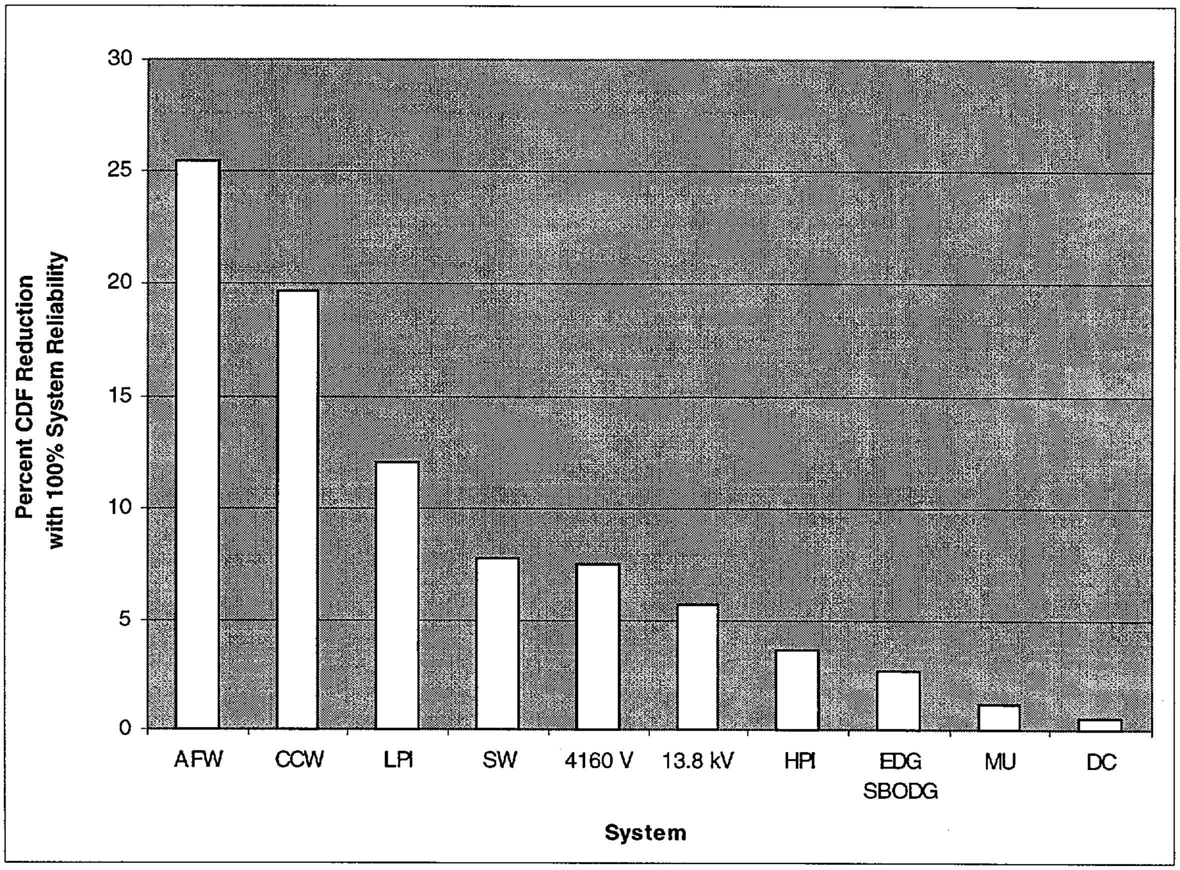

Figure E1. Systems Contributing Most to Core-Damage Frequency

The two most important systems, as indicated in Figure E1, correlate well with the two most important sequences, as described above. The AFW system has the highest Fussell-Vesely importance; if the system could be made perfect, the core-damage frequency could be reduced by more than 25%. The AFW system (including, in this assessment, the motor-driven feed pump) is clearly an important means of maintaining decay heat removal in the event of a loss of main feedwater. The redundancy in the system is offset somewhat by the fact that it relies on two turbine-driven pumps, which tend to be somewhat less reliable than motor-driven pumps.

The CCW system is important because of the role it plays in the potential for core damage due to a RCP seal LOCA. Although sequences involving a seal LOCA were somewhat more important than the sequences involving total loss of feedwater, the contributions to the seal LOCA sequences are split largely between the CCW and service water systems.

The important elements of the core-damage sequences can be broken down further by considering the Fussell-Vesely importances at the component level. The components that contribute most are illustrated in Figure E2.

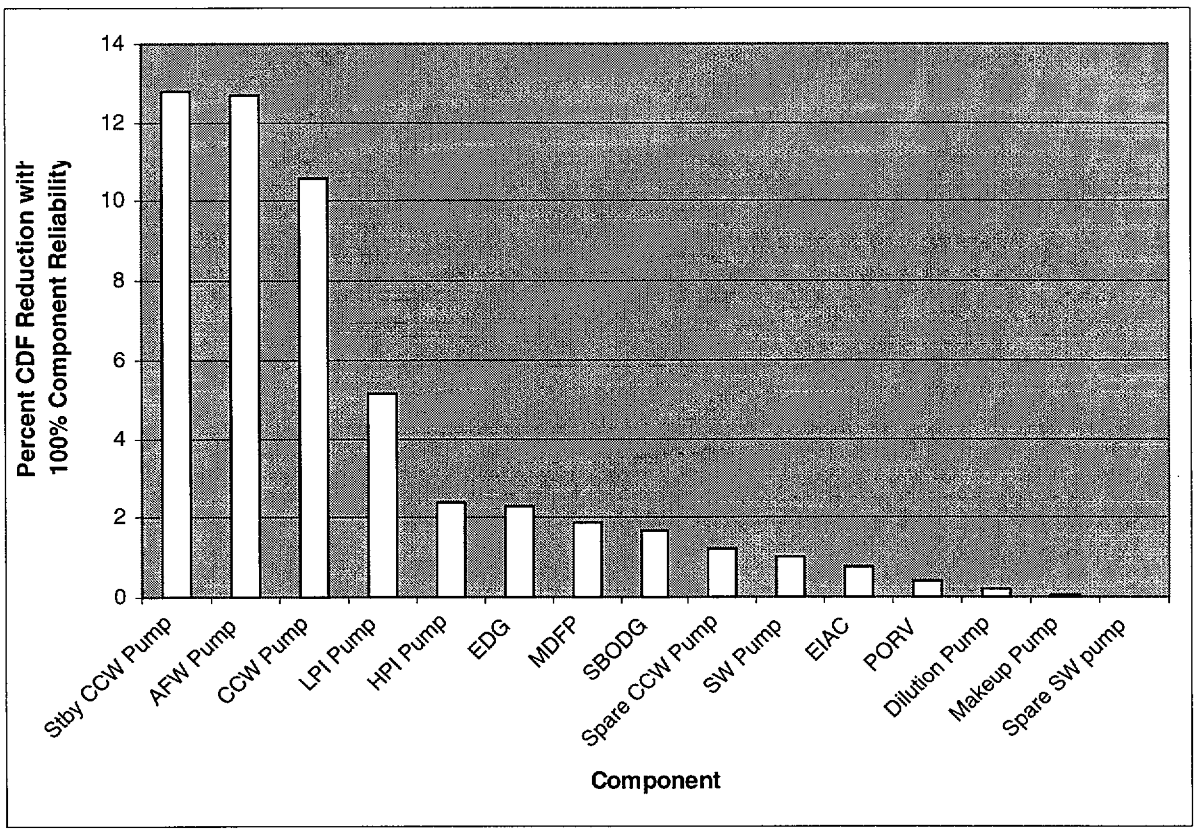

Figure E2. Components Whose Failures Contribute Most to Core-Damage Frequency

As Figure E2 indicates, the two most significant components according to the current PSA models would be the standby CCW pump and either of the turbine-driven EFW pumps. These components correlate directly with the important sequences discussed above. The standby CCW pump is an important means to restore CCW flow to maintain cooling for the RCP seals and/or to cool the HPI pumps if a seal LOCA should develop. The contribution is significant in large part because of the potential for common-cause failure of the pump given that the normally operating pump has failed.

The turbine-driven AFW pumps are clearly an important backup in the event of a loss of main feedwater. The contribution to core-damage frequency due to the potential for failure of these pumps arises because of the combination of the importance of the sequences involving total loss of feedwater and the relatively high failure rate for turbine-driven pumps (compared, for example, to motor-driven pumps).

The next highest contributor is the normally operating CCW pump. This pump is clearly important with respect to the sequence involving a RCP seal LOCA. It contributes somewhat less to the core-damage frequency because it is normally running, and therefore has a higher probability of continuing to function during a plant upset (i.e., the potential for failure of the pump to start is generally not relevant).

The low pressure injection (LPI) pumps are next, although there is a noticeable dropoff from the other contributors. The LPI pumps play an important role in nearly all types of sequences. Thus, they could have been even larger contributors to core-damage frequency had they not been relatively reliable.

It may be noted that the motor-driven feed pump was assessed to contribute a relatively moderate amount to the core-damage frequency. Although the pump is important to sequences involving the loss of main feedwater and failure of the turbine-driven AFW pumps, the Fussell-Vesely importance is not very high because the pump is relatively reliable. The importance of the motor-driven feed pump is highlighted for cases in which the pump is assumed to be unavailable (e.g., because it is in maintenance). Removing the pump from service would cause the core-damage frequency to increase significantly, as described later in this section.

Operator actions play a significant role in virtually all of the sequences in Table E2. The most important of these actions are listed in Table E3. Tripping the RCPs and lining up the spare CCW pump are both among the most risk significant operator actions; either action could avert a RCP seal LOCA, as described above.

Operator actions, including starting the motor driven feed pump and initiating makeup/HPI cooling, provide a link between the failure of feedwater for heat removal by the steam generators and the failure of makeup/HPI cooling for the second sequence described above. Although each of these actions was assessed to be reliable independently, a relatively high level of dependence was assessed between the failures related to the feedwater system and the failure to initiate makeup/HPI cooling. This was particularly the case for the failure to start the motor-driven feed pump, since both that action and the need for makeup/HPI cooling would be direct responses to the loss of feedwater from other sources. Therefore, the cut sets involving combinations of these interactions were among the important contributors to the sequence frequency.

Table E3. Important Operator Actions

Event or Sequence	Operator Action
Loss of seal cooling or seal return	Trip reactor coolant pumps
Total loss of feedwater	Establish makeup/HPI cooling
Loss of main and auxiliary feedwater	Initiate flow from the motor-driven feed pump
Loss of operating and standby CCW pumps	Line up spare CCW pump
Large or medium LOCA	Initiate low pressure recirculation from sump
Sustained station blackout	Manually control AFW pump speed

These results have been discussed in terms of the sequences and features that contribute most to the core-damage frequency. It is also instructive to examine the components that would have the greatest impact on core-damage frequency if they were unavailable. The most important components from this perspective are summarized in Table E4.

Table E4. Effect on CDF with Significant Components Out of Service

Effect on Nominal CDF	Component Out of Service
> 100x	4160v bus D1 480v bus F1
50 to 100x	4160v bus C1
10 to 50x	Motor-driven feed pump Standby component cooling water pump Station battery 480v bus E1
5 to 10x	Auxiliary feedwater pump Decay heat removal pump
2 to 5x	Emergency instrument air compressor Pressurizer PORV Dilution pump (backup service water pump) Station blackout diesel-generator High pressure injection pump Emergency ventilation for a low-voltage switchgear room
< 2x	Emergency diesel-generator Makeup pump Spare component cooing water pump Spare service water pump

As Table E4 indicates, the two components whose unavailability would have the greatest impact on overall core-damage frequency were assessed to be 4160v bus D1 and 480v bus F1. Neither of these was identified as important from the perspective of contribution to overall core-damage frequency as shown in Figure E2. This results from the relatively high reliability of the components (i.e., serious faults on such buses are quite rare). If either of these buses were not available, however, several systems would be affected.

The primary reason these buses rise higher on the list than equivalent buses in the opposite division is because of asymmetries in the arrangement of power supplies for the plant. One of the most important of these with respect to core-damage sequences is for the PORV. For example, the loss of one of these buses could lead to loss of dc power in the same division due to eventual depletion of that division's battery. This could cause loss of the turbine-driven feed pump (because of overfilling of the steam generators after loss of dc power); lack of control power to start one of the makeup pumps to support makeup/HPI cooling; and unavailability of the pressurizer pilot-operated relief valve (PORV), which could be needed to effect makeup/HPI cooling with only one makeup pump available.

The next set of contributors includes the motor-driven feed pump, the standby CCW pump, the station battery, and bus E1. With the exception of the CCW pump, none of these components is among the highest contributors to core-damage frequency, again because they are all highly reliable. The presence of the standby CCW pump is noteworthy, because it is important from both of the complementary perspectives; the potential for its failure plays a significant role in the basic core-damage frequency, and if the pump were unavailable, the frequency could be substantially higher.

Results from the Back-End Analyses

Several different types of results were obtained from the back-end analyses, including the following:

- The frequencies of various types of containment failure modes;
- The fraction of core-damage frequency that results in each of the failure modes;
- The frequencies for the release categories; and
- The frequency of large early releases (LERF).

The most likely outcome relative to the overall core-damage frequency was assessed to be a situation in which the containment maintained its integrity; approximately 93% of the core-damage frequency resulted in the no containment failure. The results for general categories of containment outcomes are summarized in Table E5.

The large early release frequency has been calculated to be about 7.3×10^{-8} per year. The chart in Figure E3 illustrates the relative contributions to LERF of the general categories of initiating events. As shown in this figure, transient initiators and steam generator tube ruptures each contribute about one third of the frequency of large early release. Interfacing-system LOCAs and internal floods are also significant contributors. LOCAs are less significant initiating events with respect to LERF.

Table E5. Probabilities and Frequencies of Containment Failure Categories

Containment Failure Category	Conditional Probability Given Core Damage	Annual Frequency
No failure	0.93	1.4×10^{-5}
Leak through small line penetrating containment that fails to be isolated	0.020	3.1×10^{-7}
Containment bypass, resulting in direct path from the RCS to outside the containment	0.020	3.0×10^{-7}
Core debris melting through the containment basemat	0.017	2.5×10^{-7}
Overpressurization of containment long after core debris melts through reactor vessel	0.0088	1.3×10^{-7}
Early containment failure (prior to or around the time of reactor vessel failure)	0.0034	5.1×10^{-8}
Containment sidewall failure due to direct contact with core debris	0.001	1.4×10^{-8}
Failure of a large path penetrating containment to be isolated	2.6×10^{-5}	3.9×10^{-10}

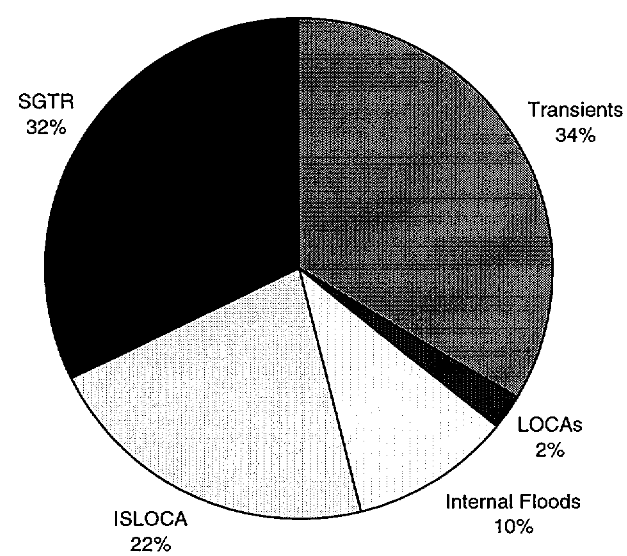

Figure E3. Contributions to Large Early Release Frequency from General Categories of Initiating Events

The importance measures for the containment outcomes and for LERF are generally similar to those described in some detail with respect to core-damage frequency. The primary differences include the following:

- Steam generator tube ruptures and interfacing-systems LOCAs were small contributors to the overall frequency of core damage. They were much more important with respect to the frequency of large, early release, since they result directly in a bypass of the containment boundary.
- LOCAs were a large contributor to core-damage frequency but are an insignificant contributor to LERF. Most LOCAs depressurize the RCS, which reduces some challenges to containment integrity and reduces the potential for induced SGTRs.
- Loss of component cooling water, and the possibility that it would lead to core-damage frequency due to a RCP seal LOCA, was very important with respect to core-damage frequency, but less so relative to the potential for serious releases. Failures of service water, however, were more important because loss of service water could contribute to both the potential for a seal LOCA and the failure of certain containment systems, including containment spray and containment air cooling.
- Sequences involving a total loss of feedwater and failure of makeup/HPI cooling were important to both the frequency of core damage and LERF. These sequences would be more likely to result in core damage at high pressure, which could lead to an induced steam generator tube rupture, or sidewall failure of containment due to dispersion of the core debris into the lower compartment of containment.

Conclusions

The overall core-damage frequency is lower than that assessed for the IPE for reasons that are largely reflective of improvements that have been achieved in the performance of Davis-Besse. The most important of these are the reductions in the frequency of plant trips, and in particular in the frequency of the loss of main feedwater. Improved component reliability, especially for motor-operated valves, also contributed to this reduction.

The objectives for the update seem to have been achieved. The PSA models and associated data bases more accurately reflect the current plant configuration and recent plant experience. They can be used much more effectively to perform a variety of applications to address plant safety and operational issues. The PSA staff is able to modify and exercise the PSA to accomplish these applications. The usefulness of these applications will be further enhanced when the update of the back-end analysis is completed later this year.

Section 1
Introduction

This report describes a recently completed update of the probabilistic safety assessment (PSA) for the Davis-Besse Nuclear Power Station. The PSA is a systematic evaluation of the potential for a core-damaging accident to occur. The PSA attempts to identify the combinations of plant upsets, hardware failures, and operator actions that are most likely to lead to core damage. It also evaluates the response of containment for these core-damage accidents and characterizes the frequencies and severities of radionuclide releases.

1.1 Background and Objectives

A PSA was completed for Davis-Besse early in 1993 in response to the USNRC's request for an individual plant examination (IPE) to search for severe-accident vulnerabilities [Refs. 1, 2]. This PSA focused on identifying the ways in which core damage could result from a wide variety of plant upsets and system failures and on quantifying the frequencies of these core-damage accidents. The PSA performed for the IPE also included an evaluation of the potential that these accidents might lead to some type of failure of the containment. This evaluation yielded an estimate of the likelihood that there might be a serious release following the severe accidents, and an initial estimate of the magnitude of any such releases. The PSA built upon a preliminary PSA of more limited scope that was performed in the late 1980's. The PSA for the IPE took more than three years to complete.

Following submittal of the IPE, a second set of evaluations was undertaken to expand the coverage to consideration of the potential for severe accidents to result from external events (e.g., earthquakes, fires, tornados, etc.). This, too, was done at the request of the USNRC, as an Individual Plant Examination for External Events (IPEEE) [Refs. 3, 4, 5].

Relatively early in the process of developing the PSA for the IPE, the plant design was "frozen". This was done to provide a clear baseline to evaluate for the IPE, since changes could not be incorporated into the analyses continuously without delaying the IPE submittal. During that period and after the submittal was made, changes to the plant and to procedures have continued to be made. Therefore, a major update of the PSA was undertaken in 1997 and has recently been completed. The objectives of this update included the following:

- To ensure that the models comprising the PSA accurately reflected the current plant, including its physical configuration, operating procedures, maintenance practices, etc.;
- To take into account more recent operating experience with respect to the frequency of plant transients and the failure rates of potentially important components;
- To make the PSA more useful for risk-based applications, such as supporting the implementation of the Maintenance Rule, by modeling some systems in more detail than had been done for the IPE and by improving the ability to evaluate the models and to perform specific investigations;

- To eliminate errors uncovered subsequent to the IPE submittal;
- To provide a more complete set of documentation to aid in subsequent applications and to make later updates more tractable; and
- To train the current PSA staff to perform probabilistic evaluations and to apply them to supporting the operation of Davis-Besse.

Each of these objectives has been met, as described in this report.

1.2 Conduct of the Update

The update was performed primarily by the PRA staff in the Design Basis Engineering section. Because of turnover, these engineers were not those who completed the IPE. An outside consultant provided some assistance, to help in directing some of the technical tasks, to provide hands-on training where necessary, to perform some of the technical tasks, and to review work products.

The Davis-Besse staff performed extensive reviews and updates of the system fault-tree models to ensure that they reflected the current plant configuration. They also conducted a significant assessment of plant-specific reliability data, involving an extensive review of plant maintenance and other records and the incorporation of that data into estimates of component failure rates, frequencies of initiating events, and other data needs for the PSA. The PSA staff also performed the bulk of the assessments of human interactions. They then integrated the models and data to quantify the frequencies of the core-damage accidents.

1.3 Major Areas Addressed in the Update

The updated PSA reflects changes that address four major considerations:

(1) The plant has changed since the PSA was completed to support the IPE;

(2) Significant additional operating experience has been gained since the time of the IPE;

(3) Some areas of analysis have been expanded to make the PSA a more useful tool for various risk-based applications; and

(4) Some of the technical methods used had been improved since the previous PSA was prepared.

With regard to the first of these, it should be noted that a PSA is composed of a diverse set of models that attempt to capture the plant configuration and operating characteristics as a snapshot in time. To complete the PSA for the IPE submittal, it was necessary to define a "freeze" date for the plant; that is, the PSA models were based on the plant as it existed at a specific point in time. For the IPE, the plant was frozen as it existed at the end of the seventh refueling outage (i.e., as of the end of 1991).

In the approximately seven years since the freeze date for the IPE, the plant has continued to undergo modifications (although not nearly to the extent that was the case in the years prior to the IPE). Procedures and operating and maintenance practices have also undergone revision and refinement. Among the more important changes incorporated into the updated PSA are the following:

- The station blackout diesel-generator has been fully implemented. At the time of the analysis for the IPE, this generator had been installed, but some procedures and other supporting details were not yet available.
- The startup feed pump had essentially been abandoned in place in the late 1980s. More recently, provisions have been made to use the pump as an additional line of defense in the event of failure of auxiliary feedwater and the motor-driven feed pump.
- Among the many changes that have been made to the operating and emergency procedures, one of the more significant relative to the PSA was the provision for somewhat different response to a steam generator tube rupture.
- New provisions have been installed to limit the effects of a flood that might originate in the turbine building due to a failure in the condenser circulating water system.

The update also involved an extensive review of more recent operating experience. As described in Section 2.1.3, various types of records developed over the course of plant operations are examined to help in developing data bases to support quantification of the PSA models. This is a very labor-intensive process, and for the IPE submittal, it was possible to evaluate experience at Davis-Besse only through the sixth refueling outage (June 1990). The current update has accounted for experience since 1990.

Changes in various practices in the late 1980s and into the 1990s have led to significantly increased attention to areas such as plant maintenance that have, in turn, yielded substantial improvements in plant performance. Among the areas in which these improvements have had a direct impact on the reliability data bases for the PSA are the following:

- The frequency of plant trips, and especially of trips due to or coincident with a loss of main feedwater, is much lower than it was for the period prior to 1990.
- Efforts to address problems with motor-operated valves have resulted in an improvement in their reliability by a factor of almost 5.
- Most other components for which operating experience was collected have experienced lower failure rates, although generally not as dramatic as that for motor-operated valves.

The modeling effort has been expanded in many areas to provide for more accurate reflection of actual operating practices, to allow for more effective applications to be made, and to correct errors uncovered since the IPE was submitted. Some of the modeling improvements include the following:

- The overall framework for the models for service water and component cooling water was revised to facilitate the consideration of restoring the systems by using the spare pumps.

- The event tree for sequences initiated by a steam generator tube rupture was modified both to reflect changes in the emergency procedure and to simplify the quantification process.
- The potential for flooding in the turbine-building basement due to failures in the condenser circulating water system was evaluated in detail. During the previous PSA, an assumption had been made that such floods could not pose a threat to core cooling, and no detailed analysis had been performed.
- The treatment of other internal floods was integrated more fully into the basic plant models than had been the case for the IPE.

Finally, the methods used for various parts of the PSA have improved since the IPE was prepared. The improvements employed during the PSA update included the following:

- The basic computer code package used to aid in the development and solution of the fault trees has been improved in many ways. The improved package supports easier development and modification of the fault trees, and has several tools to automate portions of the quantification process that previously involved tedious and time-consuming effort by the analysts.
- New tools have been used to perform analyses of the reliability data. These tools make it more efficient to document clearly the sources and treatment of the data, and to revise failure rates or other parameters as additional raw data is collected.
- The methods for human reliability analysis have been revised to address areas that were considered to be potential shortcomings in the PSA for the IPE. The tools for performing the analyses have also been improved somewhat, freeing the analysts to focus more on the potentially important elements of these evaluations.

All of these changes taken together have produced a PSA model for Davis-Besse that is more representative of the plant as it currently operates; that can be exercised more effectively and reliably for a variety of intended applications; and that is much more thoroughly documented.

1.4 Composition of the Summary Report

This report has been assembled to summarize the technical tasks that comprise the PSA update; to describe the results in a way that should prove useful to organizations that may find it necessary to apply the PSA in addressing a particular issue; and to serve as a guide to the detailed documentation for all of the technical tasks. Section 2 of this report provides an overview of each of the technical tasks that comprise the PSA. Section 3 summarizes the results and conclusions from the update of the front-end portion of the PSA. Section 4 likewise outlines the results from the back-end analyses. Section 5 provides an introduction to the potential benefits that may be gained from the PSA by discussing both applications that have been made in the past, as well as uses that are envisioned for the future.

Appendix A provides a glossary of terms used in the PSA and in some PSA applications. Appendix B is intended to provide direct reference to all of the detailed documentation that has been compiled as the PSA update was completed.

Section 2
Overview of the Technical Approach

The PSA consists of two major parts. The first of these is comprised of the models and data that provide the evaluation of the frequency of core-damage sequences. This part is variously referred to as the "level 1" portion or the "front-end" analysis. The second part encompasses the assessment of the physical response of the containment to the challenges presented by the core-damage accidents. It is referred to as the "level 2" portion or the "back-end" analysis. The discussion that follows focuses on the front-end analysis. The most recent update focused on the front-end analysis, although an update of the back-end analysis is also planned as an immediate follow-on. The back-end analysis from the IPE is also described briefly for ease of reference. The tasks for the PSA are summarized in Table 1.

2.1 Description of the Front-End Analysis

The overall methodology taken in the front-end analysis is characterized, in PSA terms, as the "linked fault-tree" approach. In such an approach, relatively small event trees are used to delineate the various sequences that could lead to core damage. The sequences are defined in terms of the success or failure of the events that make up the event tree. These events correspond to the top events of fault trees. To quantify the core-damage sequences, the top events are combined (or linked) according to the definition of the sequences, essentially forming a large fault tree for each sequence. This large, linked fault tree is then evaluated to find its "minimal cut sets". These are the combinations of initiating events, basic hardware failures, maintenance unavailabilities, and human interactions that would have to occur for the sequence to take place. It is at the level of those basic events that reliability data can be applied to provide for the estimation of sequence frequencies.

The approach as it was implemented incorporates the elements described in draft NUREG-1602 and meets the expectations for a quality PSA as stated in draft NUREG-1560 [Refs. 6, 7]. The update, and particularly the level of effort put into expanding documentation of the analyses, should also comprise a significant step toward meeting the upcoming standard for PSA [Ref. 8].

2.1.1 Delineation of Core-Damage Sequences

The starting point for the accident sequence analysis is the definition of initiating events. These reflect failures that would lead to a plant trip and that would have unique effects on the ability of the plant systems to maintain core cooling. For reference purposes, the complete set of initiating events considered in the PSA is provided in Table 2. Separate event trees were developed for each of the following general categories of initiating events:

- Large LOCA (equivalent flow area greater than 0.5 ft^2),
- Medium LOCA (0.02 to 0.5 ft^2—smaller than the large category, but still sufficient to provide for removal of decay heat),

Table 1. Technical Tasks Comprising the Davis-Besse PSA

Task	Description
Front-End Analysis	
Initiating event analysis	A careful search for system failures and other plant upsets that might initiate an accident that could ultimately threaten core cooling
Sequence analysis	Development of event trees to lay out the various types of core-damage accidents that might result from the initiating events, and definition of success criteria for the systems that would play a role in those accidents
Systems analysis	Construction of fault trees for each of the systems that could play a role in these core-damage accidents to define at a very detailed level the ways in which the systems could fail
Data analysis	Analysis of operating experience for Davis-Besse and for other operating nuclear power plants to develop reliability data bases that comprise hardware failure rates, common-cause failure rates, and maintenance unavailabilities, so that the probabilities for the basic events in the fault trees can be estimated
Human reliability analysis	Identification of actions required of plant personnel, such as correctly restoring equipment to operable status following tests or maintenance, backing up automatic systems, and performing necessary manual actions during an upset event; and estimating the probabilities for failure to accomplish these actions
Sequence quantification	Integration of the event-tree sequences, system fault trees, and reliability data to estimate the frequencies of core-damage accidents
Back-End Analysis	
Coordination with front-end analysis	Definition of conditions associated with different core-damage accidents that could present unique challenges to containment integrity
Containment modeling	Development of plant-specific computer models for thermal-hydraulic response of reactor coolant system (RCS) and containment
Accident analysis	Thermal-hydraulic analysis of the response of the RCS and containment for selected severe accidents
Containment event analysis	Development of an event tree to delineate the possible accident outcomes for each type of core-damage accident with respect to the effects on containment, and analysis of the probabilities for those outcomes
Release assessment	Characterization of severity of potential releases for each type of core-damage accident and containment response

Table 2. Initiating Events in the Davis-Besse PSA

Event	Annual Frequency
Loss-of-Coolant Accidents	
Large LOCA	1.0×10^{-4}
Medium LOCA	3.0×10^{-4}
Small LOCA	1.1×10^{-3}
Steam generator tube rupture	3.4×10^{-3}
Interfacing-systems LOCA via high pressure injection line	1.6×10^{-7}
Interfacing-systems LOCA via low pressure injection line	9.3×10^{-7}
Interfacing-systems LOCA via failure of isolation in decay heat removal letdown line	5.4×10^{-7}
Interfacing-systems LOCA due to premature opening of decay heat removal letdown line	1.8×10^{-5}
Reactor vessel rupture	5×10^{-7}
Transients	
Reactor/turbine trip	1.2
Loss of main feedwater	0.22
Loss of offsite power	3.3×10^{-2}
Spurious safety features actuation	3.5×10^{-3}
Steam generator 1 unavailable due to break in feedwater or steam line	5.7×10^{-4}
Steam generator 2 unavailable due to break in feedwater or steam line	5.7×10^{-4}
Loss of normally operating makeup pump	0.14
Loss of both makeup pumps	5.5×10^{-4}
Loss of makeup tank, letdown, or letdown cooler	0.15
Loss of power from bus YAU	4.1×10^{-2}
Loss of power from bus YBU	4.1×10^{-2}
Loss of dc power supply for NNI-X	1.7×10^{-2}
Loss of service water train 1	0.10
Loss of service water train 2	0.10
Failure of service water discharge header	3.0×10^{-4}
Loss of both trains of service water, with recovery by standby pump not possible	2.6×10^{-3}
Loss of both trains of service water, with potential for recovery by standby pump	1.9×10^{-3}
Loss of component cooling water train 1	0.19
Loss of component cooling water train 1, with recovery via spare train not possible	3.0×10^{-4}
Loss of component cooling water train 1, with potential for recovery by spare pump	7.1×10^{-4}

Table 2. Initiating Events in the Davis-Besse PSA (continued)

Event	Annual Frequency
Transients (continued)	
Loss of both trains of CCW, with no recovery possible	4.3×10^{-5}
Loss of power from 4160 vac bus C1	9.8×10^{-3}
Loss of power from 4160 vac bus D1	9.8×10^{-3}
Loss of dc power from bus D1P	1.1×10^{-2}
Loss of dc power from bus D2P	1.1×10^{-2}
Normal Service Air Compressor Fails to Run	4.1×10^{-1}
Loss of Station Air	2.1×10^{-2}
Loss of all Instrument Air	1.0×10^{-2}
Loss of Instrument Air Dryer	8.4×10^{-3}
Internal Floods	
Flood in room 52 from service water	1.0×10^{-3}
Flood in room 52 from cooling tower makeup*	1.5×10^{-4}
Flood in room 52 from fire suppression*	7.5×10^{-5}
Flood in room 51 from fire suppression system	6.0×10^{-5}
Flood in room 53 from service water supply*	5.8×10^{-5}
Flood in room 53 from failure of valve SW1395 or SW1399*	6.6×10^{-5}
Flood in room 53 from service water return*	1.6×10^{-4}
Flood in room 53 from cooling tower makeup*	4.3×10^{-5}
Flood in component cooling water pump room from service water*	7.2×10^{-4}
Flood in component cooling water pump room from fire suppression*	8.3×10^{-7}
Flood in ECCS pump room 1-1 from auxiliary building drainage	4.1×10^{-3}
Flood in ECCS pump room 1-1 from BWST suction piping*	2.5×10^{-4}
Flood in decay heat exchanger room from BWST suction piping*	2.5×10^{-4}
Flood in ECCS pump room 1-2 from BWST suction piping*	2.0×10^{-4}
Flood in ECCS pump rooms an decay heat exchanger room from auxiliary building drainage	1.3×10^{-3}
Circulating water flood in turbine building**	2.3×10^{-3}

*These initiating events were actually divided into large and maximum flood rates to account for the timing of equipment inundation during the analysis.

**This initiating event was divided into small, large, and maximum flood rates to account for the timing of equipment inundation during the analysis.

(continued from p. 7)

- Small LOCA (0.003 to 0.02 ft^2—too small to remove decay heat, but producing leakage that would exceed the normal makeup capability),
- Steam generator tube rupture,
- Transient (i.e., plant trip with no breach in the RCS larger than 0.003 ft^2), and
- Transient without scram (a special tree to treat cases in which the plant does not trip when required).

Each pathway through an event tree is defined by a set of successes and failures of safety functions that must be accomplished to prevent core damage (i.e., reactivity control, control of RCS pressure, control of RCS inventory, and decay heat removal). The failures for these safety functions are further developed through fault-tree logic to relate them to system-level failures. For example, depending on the initiating event and sequence, decay heat removal may fail because main feedwater is lost and auxiliary feedwater is unavailable. Each of the pathways for which the functional failures combine to lead to core damage, together with the initiating event, defines a core-damage sequence.

2.1.2 System Failure Analyses

A fault tree has been constructed for each of the systems that could play a role in those accident sequences. These fault trees lay out in a logical fashion all of the various ways in which the systems could fail, including failures of individual components, unavailability of components or trains of a system due to maintenance, and human interactions that could leave the components unavailable (e.g., due to failure to reopen an isolation valve following a maintenance activity). In addition to the fault trees for systems directly involved in the mission of maintaining core cooling, there are fault trees for the systems that must support these systems (such as electric power, component cooling water, service water, etc.). Thus, the effects a fault on an electrical power bus would have on the safety injection systems, and consequently on the possibility for an accident to occur, are accounted for explicitly. The systems for which failures were developed are summarized in Table 3.

Much of the effort during the recent update entailed thoroughly examining the fault trees developed for the IPE, and updating them to reflect current plant configuration and operating practices. A concerted effort was also made to upgrade the documentation associated with the fault trees. This documentation will make any future reviews, applications, and updates much more efficient.

2.1.3 Reliability Data Analysis

At the lowest level in the fault trees, reliability data is used to characterize the probabilities of the individual failures. The reliability data bases have been compiled from a variety of sources, including from an extensive review of records at Davis-Besse, as well as from experience at many other operating nuclear power plants. The data bases address a variety of data needs for the PSA, including the following:

Table 3. Systems Modeled in the Davis-Besse PSA

System	Type of Modeling
Decay heat removal	Detailed fault-tree models for low pressure injection, sump recirculation, and shutdown cooling.
High pressure injection	Detailed fault-tree models for injection and sump recirculation.
Core flood	Detailed fault-tree model was developed.
Makeup and purification	Detailed fault-tree models for seal injection, makeup/HPI cooling, and emergency boration.*
Reactor coolant	Detailed fault-tree models for pressurizer spray, the pilot-operated relief valve, the primary safety relief valves, and the reactor coolant pumps and seals.
Power conversion	Detailed fault-tree models for the atmospheric vent valves, the main steam isolation valves, and the turbine bypass valves. Simplified fault-tree models for the condensate system and the circulating water system. Simplified fault-tree model based on plant data for the main feedwater system.
Auxiliary feedwater	Detailed fault-tree model including both turbine-driven pumps and the motor-driven pump.*
Containment spray	Detailed model for injection and recirculation.
Containment air cooling	Detailed fault-tree model.
ECCS room coolers	Detailed fault-tree model.
Containment isolation	Detailed fault-tree model.
Reactor trip	System-level failure assessment.
Safety features actuation	Detailed fault-tree model for various actuation levels.
Electric power	Detailed fault-tree models for various busses, motor-control centers, all three diesel-generators, the batteries, and chargers.*
Service water	Detailed fault-tree models for various service water loads.*
Component cooling water	Detailed fault tree models for various cooling water loads.*
Instrument air	Detailed fault-tree model for various headers.

*Ventilation, heating, and cooling functions for these systems were modeled within the fault trees for the systems themselves (i.e., separate fault trees were not prepared for these functions).Maintenance unavailabilities. Operator logs were reviewed to determine the average fraction of the time trains of systems or individual components were out of service for preventive or corrective maintenance.

- Frequencies for initiating events, as summarized in Table 2. These were generally obtained from the occurrence rates for plant trips due to different causes from Davis-Besse experience and, for rarer events (such as LOCAs), from industry-wide experience.

- Hardware failure rates. Failure rates were estimated for a total of 65 different failure modes for 34 different types of components. These failure rates were developed through the review and analysis of maintenance work requests for Davis-Besse and the compilation of industry experience with similar components. In many cases, Bayesian reliability analysis techniques were employed to allow the broader industry experience to be combined with the smaller (but more directly relevant) experience specific to Davis-Besse to make the best possible use of both sources.

- Common-cause failure rates. Common-cause failures account for the possibility that multiple redundant components may fail at essentially the same time due to some shared cause, such as a design flaw that manifests itself only under unique circumstances, adverse environmental conditions, etc. Common-cause failure rates are usually expressed as a conditional probability that two (or more, depending on the system) components fail given that the first component fails. Because such failures occur too infrequently for their rates to be determined from plant-specific experience, the common-cause failure rates were estimated primarily from generic sources, based on state-of-the-art methods.

The reliability data bases therefore comprises an extensive accumulation of plant-specific and industry-wide experience. They have been compiled in manner that allows for the relevant reliability parameters to be easily traced back to their sources, and in a form that can be readily updated as additional experience is gained.

2.1.4 Human Reliability Analysis

The human reliability analysis accounts for the possibility that operators or other plant personnel may play a role in the accident sequence. The identification of possible human interactions was an integral part of the development of the sequence and system models. Once human interactions were identified, corresponding failure events were defined and included at the appropriate point in the models. Several types of failure events were included and evaluated:

- Failure to restore equipment to operable status following test or maintenance activities;

- Failure to calibrate some element of an instrumentation string properly;

- Failure to perform a manual action that is needed during the course of an accident (e.g., failure to switch to sump recirculation following a LOCA);

- Taking an inappropriate action based on a misunderstanding of plant conditions (considered in only a very limited number of cases); and

- Failure to restore a lost system or function, or failure to effect an alternative means of accomplishing the lost function.

Because it is generally not feasible to quantify the probabilities for such events by collecting data from operating experience, several different techniques were used to assess the events, based on the nature of the events. All of the methods used represent the state of the art in human reliability analysis.

2.1.5 Sequence Quantification

The frequencies of the accident sequences were calculated using the CAFTA computer program, CAFTA [Ref. 9], which was also used to aid in the development of the fault trees. CAFTA was used to find (through a process of Boolean reduction) the combinations of basic failures in the fault trees that could lead to the accidents of interest. It then applied the parameters from the appropriate data base to each of the basic failures to quantify the accident frequencies. The results were obtained in the form of lists of cut sets. Each cut set is one combination of an initiating event and additional individual failures that would cause the accident to occur. These cut sets were examined to ensure that they represented the accidents as realistically as possible, and to determine whether there were any further opportunities for the operators to act to prevent core damage in a way that was not explicitly modeled in the fault trees. Reviewing the cut sets also served as a check on the completeness and quality of the event-tree and fault-tree models. This was accomplished by verifying that the cut sets made logical sense (i.e., that the sets of failures they represent really would lead to the types of accidents being considered) and that no combinations of failures that would be expected to be produced in the quantification process were missing. Once the cut sets were verified to be appropriate and the opportunities for recovery from the accident are incorporated, the basic quantification process was complete.

For understanding both the basic PSA results and the results of many of the applications that may be made of the PSA, it is useful to have a clearer picture of this integration and quantification process. As an example, consider the sequence designated as "TBU" in the event tree for transient initiators. The first letter indicates the type of initiating event (T, in this case, refers to the general category of transient initiating events). The letter B is used to designate failure of the safety function of decay heat removal via the steam generators, and U represents failure to provide decay heat removal (and RCS inventory control) by means of makeup/HPI cooling. Each of these two functions corresponds to a top event in the event tree. Implicit in the sequence definition is that certain safety functions have not failed. In this case, successful reactor trip and maintenance of RCS integrity by successful closure of pressurizer relief valves are implied.

If the functions represented by both event B and event U were to fail, core damage would be expected. The sequence was quantified by combining the fault trees for the main feedwater system, the auxiliary feedwater (AFW) system, and the makeup system into a master fault tree and then evaluating it using CAFTA to obtain sequence-level cut sets. For various sequences, this solution process can produce dozens to thousands of sequence cut sets. An example of one of the cut sets for sequence TBU, presented in terms of the alphanumeric designators used to track the events through the fault trees and in the computer solution, is shown below:

T2	*	QMBAFP12	*	QMM0028A	*	QHAMDFPE	*	UHAMUHPE
(0.22 /yr)	x	(0.0028)	x	(0.011)	x	(0.0013)		

$$= 8.8 \times 10^{-9}/\text{yr}$$

where

T2	designates the initiating event loss of main feedwater,
QMBAFP12	indicates that turbine-driven AFW train 2 is unavailable because it is in maintenance,
QMM0028A	refers to failure of turbine-driven AFW pump 1 to start on demand,
QHAMDFPE	designates failure of the operators to start the motor-driven feed pump in time to preserve cooling via the steam generators, and
UHAMUHPE	refers to the failure of the operating staff to implement makeup/HPI cooling in time to avoid uncovering the core.

The asterisk between events (*) signifies the Boolean operator AND; that is, all of the failures represented by the events in the cut set must exist simultaneously for the sequence corresponding to the cut set to take place. Below the cut set itself are the corresponding initiating event frequency and event failure probabilities. Note that a single combined probability is included for the two events signifying human interactions (QHAMDFPE and UHAMUHPE). This single value reflects the fact that the human interactions appearing in a cut set were considered in combination, to ensure that any common aspects (e.g., the potential for misdiagnosis, leading to an overall incorrect response) were properly accounted for. The product of the values below the cut set is the frequency of the cut set, and the overall sequence frequency is estimated from the sum of the frequencies for all of the individual cut sets that comprise that sequence. Thus, this cut set has a frequency of 8.8×10^{-9} per year.

It is important to note that the quantitative aspects of the PSA are subject to uncertainties, which are sometimes quite large. These uncertainties stem from a variety of sources. For example, the failure rates for some components are uncertain because there is not much information available to use to estimate them. Some of the larger uncertainties arise because there are no precise methods to quantify the probability that humans will not successfully accomplish an important task. There are also uncertainties in the understanding of accident progression. For instance, although substantial information is available, there is no definitive picture of the degree of leakage that will occur through the seals for the reactor coolant pumps (RCPs) if they should lose all seal cooling. The definition of accident sequences that could include failure of the RCP seals reflects the best current understanding of their performance following loss of cooling, but changes in this understanding could affect the sequence definition and the corresponding estimates of core-damage frequency.

When discussing the results of the front-end analyses, it is most common to focus on the basic (or best-estimate) results. It can be important, however, to keep in mind the nature of uncertainties in those results, to avoid putting undue reliance on bottom-line numbers to support decision-making. The qualitative results, and especially the insights into the combinations of basic failures that could lead to an accident, are often the most useful. With the perspective of the quantitative results, tempered as appropriate by a recognition of the important uncertainties, the PSA can be used to aid in understanding a variety of issues.

2.2 Description of the Back-End Analysis

The back-end analysis focuses on the course of the accident beyond the point at which core damage begins. Much of the back-end analysis entailed the performance of thermal-hydraulic calculations to simulate the process of heatup and meltdown of the core, to follow the relocation of the core material after it leaves the reactor vessel, and to estimate the pressures and temperatures inside the containment. This information was used in a probabilistic framework to determine whether and how the containment might fail to retain its integrity due to the effects of the accidents, resulting in a large release of radioactivity. The tasks that comprise the back-end analysis are described below.

For Davis-Besse, most of the thermal-hydraulic calculations were performed with the MAAP code, which was designed specifically to be used in the analysis of core-damage accidents [Ref. 10]. The results of the MAAP calculations provide a nominal time history for a large number of parameters that characterize the core and containment response for each type of core-damage accident.

2.2.1 Coordination of the Front-End and Back-End Analyses

An important element of the IPE was the coordination of the front-end and back-end analyses. This was done to ensure that the core-damage sequences were developed to the level of detail necessary for meaningful assessment of containment response, with a minimum of iteration between the two major analysis areas. This coordination was accomplished through the definition of plant-damage states. The plant-damage states reflect binning of accident sequences at two major levels. First, the accident sequences up to the onset of core damage were grouped into core-damage bins according to similarities in their impact on subsequent containment response. These bins helped to ensure that the core-damage sequences were developed in sufficient detail to permit them to be tracked properly in the containment event tree. The second level encompassed the status of the containment systems (the containment air coolers, containment spray, etc.). The status of these systems defined in large measure the capability of the containment to prevent a serious release as a result of the core-damage accidents. The core-damage bins together with the states for the containment systems comprise the plant-damage states.

The most significant change in this area relative to the original IPE analyses was in the level of detail reflected in the plant-damage states, and the manner in which their frequencies were calculated. For the update, nearly 500 plant-damage states were defined to accommodate the core-damage bins and the various combinations of system states that could affect subsequent containment response. A framework was established to allow all of the plant-damage state frequencies to be calculated in a manner that could be readily repeated for sensitivity studies or follow-on applications.

2.2.2 Modeling of Containment Response and Accident Analysis

To characterize the containment response to a core-damage accident, the Modular Accident Analysis Program (MAAP), version 3.0B (Revision 18), was selected as the primary analytical tool [Ref. 10]. In addition to the analyses made using MAAP, specific issues were investigated through reviews of technical literature and other calculations. Models of the RCS, the emergency core cooling system, and the containment were developed based on information derived from drawings of major components, plant drawings, system descriptions, etc., so that the models were entirely plant-specific.

MAAP is intended to serve as a tool to perform realistic analyses of severe core-damage accidents. Evaluations using MAAP were made for a representative set of the plant-damage states. The results of these MAAP calculations were assumed to reflect the nominal or expected response of containment to the accidents. These results include a very large number of parameters defining the conditions in the RCS and containment, the location of core debris, and the transport of fission products as a function of time. Because many of the phenomena accounted for in the MAAP code are subject to potentially significant uncertainties, a series of sensitivity studies was conducted. These sensitivity studies were primarily derived from a set of standard studies recommended for PWRs. The results from the MAAP analyses performed for the IPE were used in the update as well; no new MAAP calculations were performed. As time and personnel resources permit, it would be desirable to perform new investigations using a new version of the code, MAAP 4.

In addition to the MAAP analyses, calculations were made using separate tools to investigate such issues as the possible pressures that could be generated by hydrogen burns, the potential for creep rupture of RCS components subjected to very high temperatures and pressures, and cooling of core debris in various configurations. For the update, a review was made of research and analytical studies completed since the IPE. Relatively little relevant information was identified during this update. The primary exception was the treatment of the potential for a rupture of a steam generator tube to be induced due to the transport of hot gases to the steam generators during meltdown of the core.

In addition to these analyses, an assessment was made of the capacity of the containment vessel to retain its integrity when exposed to internal pressurization. The analysis investigated various mechanisms for containment failure to identify those that might limit its capacity. The expected yield strength was calculated and, based on variability in the materials used in the containment vessel and uncertainty in the method used to calculate the yield strength, a probability distribution for containment failure as a function of internal pressure was developed. A second distribution was developed to apply for scenarios in which pressurization would occur over a long period of time, such that the heating of the containment might reduce the strength of the containment shell.

2.2.3 Containment Event Analysis

For most types of accidents, the specific response of the containment cannot be predicted with certainty. A containment event tree was therefore constructed to provide a framework for investigating possible outcomes given core damage. The top events in the containment event tree represent general types of containment failure modes and conditions that would affect the magnitude of release from containment. In a manner analogous to the event trees for the front-end analysis, each of the top events was decomposed further through fault-tree logic into the various combinations of phenomena, system operations, and human interactions that could bring them about. Thus, an integral model of potential containment responses was developed.

The probabilities of the occurrence or of the severity level for various phenomena were quantified by a number of means. Although only point estimates were developed for these events, they reflected an appropriate assessment of uncertainties. In many cases the results of MAAP calculations formed the primary inputs. The MAAP results were supplemented heavily by input from other technical efforts, and especially from the analyses performed in support of NUREG-1150 [Ref. 11]; by sensitivity studies; and by engineering judgment.

The probabilities of the end states for the containment event tree were quantified for each of the plant-damage states. This quantification produced an estimate of the conditional probability of each type of containment failure mode and permitted estimation of the frequencies for each type of release from containment. To aid in understanding the elements that contributed to these results, some of the key event probabilities were varied in a series of sensitivity studies.

2.2.4 Assessment of Fission-Product Releases

Associated with each combination of plant-damage state and containment event tree outcome is a particular type of release of fission products from containment. A characterization of these releases provides further indication of the level of severity of the accidents, and would be necessary in the event that offsite consequences were to be calculated. For convenience in the analysis and in the presentation of results, the releases have been grouped into nine release categories. Thus, each outcome from the containment event tree is assigned to one of the release categories.

In addition to providing a representation of the containment response, the MAAP code tracked the status of fission products in the RCS, the containment, and as they were released from the containment. The MAAP results for different cases were grouped according to the release fractions for important species of fission products, and were used to suggest representative release fractions for each release category. In some cases, adjustments were made to the release fractions available from the MAAP results to reflect containment outcomes that were not explicitly calculated using MAAP. This was done, for example, to adjust the release fractions from a case in which containment sprays were not available to apply for a release category in which scrubbing by the sprays would have been available.

Most of the applications that have been made and are envisioned for the future require obtaining results only from the front-end analyses. That is the primary reason for the priority placed on updating the front-end. It should be noted, however, that many applications of the PSA to address regulatory considerations require some consideration of the consequences of the core-damage accidents, and not just of there frequencies. The most common measure used to account, at least indirectly, for these consequences, is the frequency of large, early releases (commonly referred to as "LERF"). This measure is calculated by combining the results from the core-damage assessment (the front-end) with the conditional probabilities that the accidents would result in serious releases relatively soon after core materials melted through the reactor vessel.

2.3 Examination of Results

Once the technical review of the PSA results has been completed to ensure that they constitute a reasonable characterization of the risk of core damage and of failures of containment that could lead to various levels of radionuclide releases, the process of examining the results to gain additional insights can be accomplished. This examination can entail a variety of activities, including studying the basic results themselves, calculating various importance measures, and performing sensitivity studies.

Importance measures provide a quantitative means to rank elements of the models to understand the roles they play in the results from various perspectives. Two complementary importance measures are used most often:

- Fussell-Vesely importance, which indicates the fraction of the overall frequency for the result of interest that results from a particular event, system failure, etc., and
- Risk achievement worth (RAW), which reflects the increase in frequency that would result if an event probability were set to 1.0 (i.e., the impact if a component were actually unavailable).

Each of these measures has its usefulness. The Fussell-Vesely importance provides a measure of the contribution of any particular element of the model to the overall core-damage frequency or release frequency. For example, all of the components whose failures are reflected as basic events in the sequence cut sets can be ranked according to their Fussell-Vesely importance measures. It would be the components highest in this ranking for which the most gain would be realized in terms of reducing core-damage frequency if steps could be taken to increase their reliability or to provide for further redundancy. The components that place high in this ranking reflect some combination of playing an important role in the accident sequence and having a somewhat high probability of failure. For example, one of the most important components was assessed to be an AFW pump. Clearly, these pumps play an important part in preserving core cooling after a loss of main feedwater. Operating experience also indicates that these turbine-driven pumps do not have extremely low failure rates. A Fussell-Vesely ranking of operator actions could be used to help focus elements of training or improvements to procedures.

The RAW measures complement such a ranking because it reflects the potential increase in core-damage or release frequency if, for example, a component fails, irrespective of the component's basic failure rate. For example, some of the ac power buses have relatively high RAW values, because if they were to fail significant sets of equipment could be unavailable. The buses are not as high on the Fussell-Vesely ranking, however, because their basic failure rates are relatively low. The RAW measure is particularly useful for such applications as the Maintenance Rule and risk monitoring activities, when equipment will be known to be unavailable. It is also useful for considering equipment for which prudent steps might need to be considered that could ensure that previous high levels of reliability are maintained.

Sensitivity studies can also help to provide further insights into the results of the PSA. The importance measures noted above are formalized types of sensitivity studies. The analyst may also vary any other parameters in the study to investigate the impact on core-damage frequency. This might be done, for example, if there were concern that the basic methods underlying the human reliability analysis produced results that were biased in one direction or the other. The probabilities for all of the human interactions could be varied to examine the possible impact of such a bias. Sensitivity studies can also be useful for a variety of applications, such as examining the effects of different allowable outage times, to support efforts to develop a sounder basis for technical specifications.

Section 3
Results from the Update of the Front-End Analyses

A PSA represents the integration of a diverse set of models and data that are developed based on a substantial volume and variety of information related to plant design and operation and that incorporate many elements of operating experience. The results of the PSA can, therefore, be broken back down in a variety of ways that may offer new insights into important plant features and a different perspective on safety issues, both new and historical.

3.1 Frequency of Core Damage and Important Contributors

The focus of the discussion in this section is on the results from the front-end analysis. These results encompass the frequency of core damage and the elements that contribute to it. For the update, the core-damage frequency has been calculated to be about 1.6×10^{-5} per year. This represents a reduction from about 6.6×10^{-6} per year as calculated for the IPE. As an absolute value, the frequency of core damage is not particularly meaningful. A discussion of the results, however, will help to illuminate the reasons that the current frequency is substantially lower than that calculated for the IPE. The results are broken down as follows:

- By the types of events that could be most important in initiating a core-damage sequence;
- By the types of sequences that contribute most to the core-damage frequency;
- By the types of human interactions that are most important in the sequence cut sets;
- By the components whose failures are most important with respect to core-damage frequency; and
- By the systems that are most important to the results.

3.1.1 Breakdown by Categories of Initiating Events

The chart shown in Figure 1 illustrates the relative contributions to core-damage frequency of the general categories of initiating events. As shown in this figure, transient initiators contribute most to the frequency of core damage. LOCAs and internal flooding are also significant contributors. Interfacing-system LOCAs and steam generator tube ruptures (SGTRs) are less significant initiating events with respect to core damage frequency, but can be more important to other risk measures since they may lead to releases that bypass the containment boundary.

Table 4 provides a more detailed breakdown of the contributions from specific types of initiating events. This table lists the name of the initiating event, the core damage frequency resulting from that initiating event, and the percent of the total core damage frequency resulting from the initiating event.

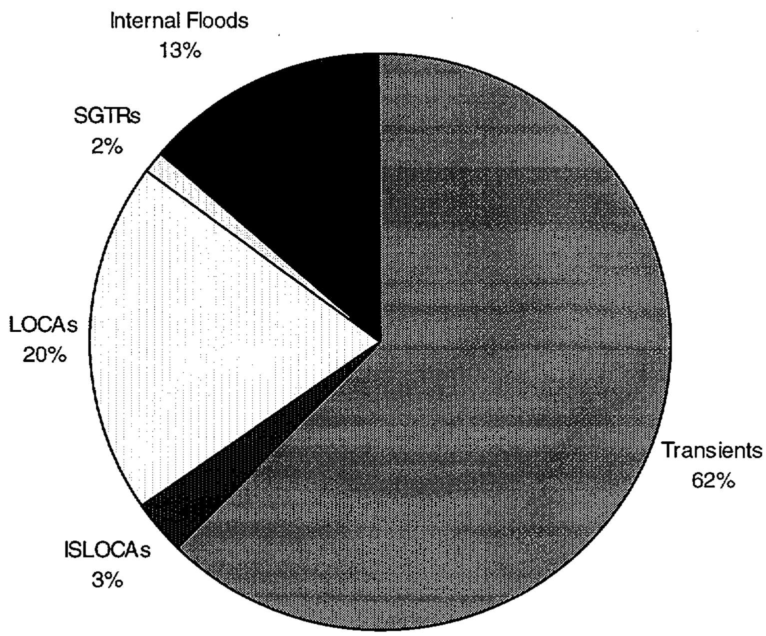

Figure 1. Contributions to Core-Damage Frequency of General Categories of Initiating Events

The results in Table 4 indicate that no single initiating event dominates the total core-damage frequency. Several initiators together, however, contribute a large fraction of the total. The discussion that follows describes some of the more significant initiators.

Malfunctions of the Component Cooling Water System

The initiator labeled "component cooling water malfunctions" in Table 4 represents a grouping of four initiating events involving failures in the CCW system.. Most of the contribution to core-damage frequency from this group of initiators can be attributed to a loss of the operating CCW pump with a failure of the standby pump to start. This initiator is significant because without CCW, seal cooling for the reactor coolant pumps (RCPs) would be lost. This could lead to a seal LOCA. Furthermore, without cooling from CCW, high pressure injection would not be available to mitigate the LOCA.

Loss of Instrument Air

Loss of instrument air is a significant initiator because it has a relatively high frequency and causes a loss of RCP seal return. Without operator action to trip the RCPs, the loss of seal return could lead to a seal LOCA as well. Although the frequency of this initiating event is larger than that for any of the events leading to a loss of CCW as outlined above, the loss of instrument air contributes less to the frequency of core damage because it does not contribute to the failure of high pressure injection.

Table 4. Breakdown of Core-Damage Frequency by Initiating Event Type

Event	Core Damage Frequency	Percent of Total CDF
Component cooling water malfunctions	2.4×10^{-6}	15.4%
Loss of instrument air	1.9×10^{-6}	12.2%
Medium LOCA	1.6×10^{-6}	10.4%
Flooding in CCW pump room	1.4×10^{-6}	8.9%
Reactor trip	1.1×10^{-6}	7.0%
Loss of 4160v bus	1.1×10^{-6}	6.8%
Loss of main feedwater	9.7×10^{-7}	6.1%
Large LOCA	7.8×10^{-7}	4.9%
Service water malfunctions	7.7×10^{-7}	4.9%
Small LOCA	6.7×10^{-7}	4.2%
Loss of dc bus	5.9×10^{-7}	3.7%
Loss of offsite power	4.8×10^{-7}	3.0%
Interfacing-system LOCA	4.8×10^{-7}	3.0%
Turbine building flood from circulating water	3.7×10^{-7}	2.3%
Loss of non-nuclear instrumentation bus YAU or YBU	3.1×10^{-7}	2.0%
Steam generator tube rupture	2.4×10^{-7}	1.5%
Flood in service water pump room	2.2×10^{-7}	1.4%
Makeup system malfunctions	1.4×10^{-7}	0.9%
Flood in service water valve room	9.5×10^{-8}	0.6%
Flood in ECCS room	6.9×10^{-8}	0.4%
Main steam-line or main feedwater line break	2.9×10^{-8}	0.2%
Spurious safety feature actuation	3.2×10^{-9}	< 0.1%

Medium LOCA

The initiating event frequency for the medium LOCA is much lower than that for the other significant initiating events. This event is significant because of the manual actions that are required to initiate recirculation after depletion of the inventory in the borated water storage tank (BWST). The same actions must be performed for a large LOCA in a shorter time interval.

For the large LOCA, however, these actions are less significant because the initiating event frequency for a large LOCA is lower than that for a medium LOCA.

Flooding in CCW Pump Room

Flooding in the CCW pump room is a special case of a loss of CCW. A large flood could cause the loss of all CCW with limited possibility for recovery in the short term. As described above for CCW malfunctions, a total loss of CCW is particularly significant because it would lead to the loss of RCP seal cooling, potentially causing a seal LOCA with high pressure injection unavailable to mitigate the LOCA.

Reactor Trip

The initiating event referred to as a reactor trip accounts for all plant trips in which none of the systems needed to maintain core cooling are directly affected. Although such an initiator has consequences that are less severe than those for most other initiating events, it remains somewhat significant with respect to the overall core-damage frequency because of its high frequency relative to the other events. Reactor trips are contributors to many sequences. The most significant sequence was assessed to be one in which main feedwater would fail to continue supplying the steam generators after the trip, contributing to a total loss of heat removal via the steam generators. Core damage could result in the event of coincident failure of makeup/HPI cooling. In the PSA performed for the IPE this loss of cooling sequence was much more significant than it was assessed to be in the updated study. The major reason for the reduction of its contribution was the significant drop in the frequency of plant trips over the intervening years.

Loss of 4160v Bus

The loss of a 4160v bus would result in the loss of the operating CCW pump, which can lead to a total loss of CCW. Once again, this could lead to the loss of cooling for the RCP seals and a consequential seal LOCA.

In addition to the potential for a RCP seal LCOA, the loss of 4160v bus D1 presents other challenges. Loss of the bus would lead to both unavailability of one of the makeup pumps and, eventually (as a consequence of depletion due to lack of charging), to the loss of one division of dc power and the unavailability of the pressurizer pilot-operated relief valve (PORV). This combination of failures would severely limit the potential success paths for makeup/HPI cooling. Thus, this initiating event is important due to its contributions to two different core-damage sequences.

Loss of Main Feedwater

Loss of main feedwater is a significant initiating event because it has a high frequency (relative to most of the other initiating events) and would contribute to the total loss of heat removal via the steam generators. Due to improved plant performance in recent years, the frequency of this initiator has dropped dramatically since the PSA for the IPE. Together with the reduced frequency of general plant trips, this reduction contributed to the overall decrease in core-damage frequency and to the large decrease in the frequency of sequences involving loss of heat removal via the steam generators.

3.1.2 Breakdown by Core-Damage Sequences

Each of the core-damage sequences is comprised of many cut sets that have the same general category of initiating events and the same failures of safety functions. Table 5 summarizes all of the core-damage sequences quantified in this assessment, their annual frequencies, and the fraction of the total core damage frequency each contributes. For each sequence, the core-damage bin to which it was assigned is also provided. This information is used to help in coordinating the front-end analyses with the assessment of the accident progression (i.e., the back-end analyses).

The core-damage frequency for Davis-Besse has contributions from 43 separate sequences listed in Table 5. A relatively small number of sequences, however, constitute a significant percentage of the total core-damage frequency. Five sequences contribute 75% of the total core-damage frequency, and ten sequences account for almost 90% of the total.

Each of the sequences that contributed at least 1% of the total core-damage frequency is described briefly below. In each case, a general description of the types of failures and initiating events that are found to be most significant is provided. The sequence cut-set files in the Level 1 results notebooks contain substantially more detail.

Sequence TQU

Sequence TQU reflects the potential for a transient-induced LOCA due to a failure of the RCP seals, followed by failure to provide adequate safety injection. A failure of the RCP seals is postulated to occur if component cooling water (CCW) to the seals' thermal barriers were lost coincident with loss of seal injection from the makeup system, or if seal return flow were lost and the operators did not trip the pumps. In either case, timely operator action to trip the RCPs would prevent serious leakage from the seals. This sequence was estimated to have a frequency of 3.6×10^{-6} per year, accounting for about 23% of the total core-damage frequency.

The CCW and service water systems play a significant role in this sequence. In addition to providing cooling for the RCP seals, CCW is required for operation of both the makeup and injection systems. Thus, an extended loss of CCW would lead to total loss of seal cooling and failure of both injection systems. In addition to faults within the CCW system, loss of service water could lead to heatup of the CCW system and eventual loss of cooling as well. Among the important contributors for this sequence are the following:

- The loss of one train of CCW, with the failure of the standby train, or failure of service water to supply cooling to the heat exchanger in the standby train. This was assessed to be the largest contributor for this sequence. Following the loss of CCW there would be about an hour to restore cooling before failure of the HPI pumps. Otherwise, without operator action to trip the RCPs, a seal LOCA would result, and the makeup and HPI pumps would not be available to provide injection.

- A plant trip due the total loss of CCW, with failure to trip the RCPs. As in the case above, about one hour would be available following the initiation of the seal LOCA to restore cooling before the failure of the HPI pumps.

Table 5. Results for All Core-Damage Sequences

Sequence	Description	Core-Damage Bin	Annual Frequency	Contribution to Total (%)
Large and Medium LOCAs				
MX	Medium LOCA initiating event with failure of low pressure recirculation	MRX	1.3×10^{-6}	8.0
AX	Large LOCA initiating event with failure of low pressure recirculation	ARX	6.7×10^{-7}	4.1
A_V	Reactor vessel rupture initiating event	ARX	4.6×10^{-7}	2.8
MU	Medium LOCA initiating event with failure of high or low pressure injection	MIX	3.8×10^{-7}	2.3
AU	Large LOCA initiating event with failure of low pressure injection	AIX	1.0×10^{-7}	0.6
Total for large and medium LOCAs			2.9×10^{-6}	17.9
Small LOCAs				
SX	Small LOCA initiating event with failure of long-term cooling via DHR or recirculation from sump	SRY	5.4×10^{-7}	3.3
SU	Small LOCA initiating event with failure of injection	SIY	8.4×10^{-8}	0.5
SCX	Small LOCA with failure to cool down and failure of high pressure recirculation	SRY	4.6×10^{-8}	0.3
SBU	Small LOCA initiating event with failure of feedwater and failure of makeup/HPI cooling	SIN	8.1×10^{-9}	< 0.1
SBX	Small LOCA initiating event with failure of feedwater and failure of high pressure recirculation	SRN	5.2×10^{-10}	< 0.1
Total core damage frequency for small LOCAs			6.7×10^{-7}	4.1
Steam Generator Tube Ruptures				
RCX	SGTR with failure of cooldown via intact steam generator and failure of long term cooling	RRY	1.3×10^{-7}	0.8
RLX	SGTR with failure to isolate ruptured steam generator and failure of long term cooling	RRY	6.1×10^{-8}	0.5
RCU	SGTR with failure of injection and failure to cooldown via the intact steam generator	RIY	2.5×10^{-8}	0.2
RB_IU	SGTR with loss of feedwater to intact steam generator and failure of makeup/HPI cooling	RIY	(Included above)	(Included above)

Table 5. Results for All Core-Damage Sequences (continued)

Sequence	Description	Core-Damage Bin	Annual Frequency	Contribution to Total (%)
Steam Generator Tube Ruptures (continued)				
RCLX	SGTR with failure of cooldown via intact steam generator, failure to isolate the ruptured steam generator and failure of long term cooling	RIY	1.7×10^{-8}	0.1
RUL	SGTR with failure to injection and failure to isolate the ruptured steam generator	RIY	6.3×10^{-9}	< 0.1
RB_IB_AU	SGTR with loss of feedwater to both steam generators and failure of makeup/HPI cooling	RIN	4.0×10^{-9}	< 0.1
RB_IB_AX	SGTR with a loss of feedwater to both steam generators and failure of long-term cooling	RRN	4.7×10^{-10}	< 0.1
Total for steam generator tube ruptures			2.4×10^{-7}	1.5
Interfacing-System LOCAs				
V_SI_S	Interfacing-systems LOCA due to premature opening of DHR suction valves and failure to isolate the break	V	3.2×10^{-7}	2.0
V_DI_D	Interfacing-systems LOCA due to hardware failure of DHR suction valves and failure to isolate break	V	1.1×10^{-7}	0.7
V_LI_L	Interfacing-systems LOCA due to a failure in LPI injection line and failure to isolate break	V	4.9×10^{-8}	0.3
V_HI_H	Interfacing-systems LOCA due to failure in HPI injection line and failure to isolate break	V	5.1×10^{-9}	< 0.1
Total for interfacing-systems LOCAs			4.8×10^{-7}	3.0
Transients				
TQU	Transient initiating event with RCP seal LOCA and failure of injection	SIY	3.7×10^{-6}	22.8
TBU	Transient initiating event with a total loss of feedwater and failure of makeup/HPI cooling	TIN	3.3×10^{-6}	20.3
TQX	Transient with RCP seal LOCA and failure of long term cooling	SRY	1.9×10^{-6}	11.7
TBLX	Transient with extended loss of feedwater and failure of high pressure recirculation	TRN	4.8×10^{-7}	3.0

Table 5. Results for All Core-Damage Sequences (continued)

Sequence	Description	Core-Damage Bin	Annual Frequency	Contribution to Total (%)
Transients (continued)				
TBQU	Transient with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of makeup/HPI cooling	SIN	2.7×10^{-6}	11.7
TKBP	Transient with failure to trip, loss of main feedwater and excessive RCS pressure	AIX	1.2×10^{-7}	0.7
TBQX	Transient with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of HPR	SRN	8.6×10^{-9}	0.1
TKBU	Transient with failure to trip, loss of main feedwater and failure to provide borated water	TIY	4.1×10^{-9}	< 0.1
TKBL	Transient with failure to trip and total loss of feedwater	AIX	7.0×10^{-9}	< 0.1
TBWX	Transient with initial loss of feedwater, successful makeup/HPI, stuck open relief when feedwater is restored and failure of HPR	SRY	8.1×10^{-9}	< 0.1
TBP	Transient with a total loss of feedwater and failure of pressurizer relief valves to open	TIN	1.7×10^{-10}	< 0.1
Total for transients			9.8×10^{-6}	60.3
Internal Floods				
FQU	Internal flood with RCP seal LOCA and failure of injection	SIY	1.7×10^{-6}	10.5
FBU	Internal flood with a total loss of feedwater and failure of makeup/HPI cooling	TIN	3.4×10^{-7}	2.1
FBLX	Internal flood with extended loss of feedwater and failure of high pressure recirculation	TRN	4.5×10^{-8}	0.3
FQX	Internal flood with RCP seal LOCA and failure of long term cooling	SRY	4.2×10^{-8}	0.3
FBQU	Internal flood with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of makeup/HPI cooling	SIN	2.0×10^{-8}	0.1
FBQX	Internal flood with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of HPR	SRN	9.4×10^{-10}	< 0.1

Table 5. Results for All Core-Damage Sequences (continued)

Sequence	Description	Core-Damage Bin	Annual Frequency	Contribution to Total (%)
Internal Flooding (continued)				
FKBP	Internal flood with failure to trip, loss of main feedwater and excessive RCS pressure	AIX	5.1×10^{-10}	< 0.1
FKBU	Internal flood with failure to trip, loss of main feedwater and failure to provide borated water	TIY	1.0×10^{-10}	< 0.1
FBWX	Internal flood with initial loss of feedwater, successful makeup/HPI, stuck open relief when feedwater is restored and failure of HPR	SRY	8.1×10^{-9}	< 0.1
FKBL	Internal flood with failure to trip and total loss of feedwater	AI X	7.0×10^{-9}	< 0.1
FBP	Internal flood with a total loss of feedwater and failure of pressurizer relief valves to open	TIN	1.7×10^{-10}	< 0.1
Total for internal floods			2.0×10^{-6}	13.2
Total for all sequences			1.6×10^{-5}	

(continued from p. 23)

- A plant trip required by total loss of service water, with failure of the operators to trip the RCPs. The potential for this sequence is due largely to the possibility of common-cause failure of the three service water pumps (two of which are normally operating) either because they fail to continue operating, or because of loss of ventilation (all three pumps are located in the same room). Cooling could be restored via use of a separate backup pump (the dilution pump). During normal operation, the CCW loads on the service water system are relatively low, so that it would take about an hour for loss of service water to lead to loss of CCW. This would allow ample time for recovery, if the third service water pump were not affected by the common cause, or if the dilution pump were available. Otherwise, without operator action to trip the RCPs, a seal LOCA would result, and the makeup and HPI pumps would not be available to provide injection.
- A plant trip due to loss of one of the operating trains of service water, with failure to provide cooling for the other train of CCW or failure of the other train of CCW.

The other mechanism for inducing a seal LOCA would be for the RCPs to continue to run after seal return had been isolated. The frequency of loss of seal return reflects dependence on several support systems, including instrument air, dc power, and ac power. The contribution to the frequency of core damage for sequences involving loss of seal return is lower than those cited above because there is substantially less dependence between the seal return and the injection systems than is the case for loss of either CCW or service water. Scenarios initiated by a loss of instrument air, however, do make a modest contribution to the frequency for sequence TQU.

Sequence TBU

Sequence TBU involves a transient initiating event with failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. With an estimated frequency of 3.3 x 10^{-6} per year, this sequence accounts for about 20% of the total core-damage frequency. Scenarios initiated by a loss of main feedwater, a plant trip, and loss of offsite power account for about 60% of the total frequency for this sequence. The remainder results from many different initiating events.

In the event of a loss of main feedwater, the two turbine-driven AFW pumps would be actuated automatically. If either or both of these pumps were to fail, emergency procedures call for starting the standby motor-driven feed pump. If no feedwater were available, procedures direct the operators to initiate makeup/HPI cooling to provide decay heat removal when RCS temperature exceeded 600F, irrespective of attempts to regain feedwater. If no feedwater were available and makeup/HPI cooling were not established, relatively early core damage at high RCS pressure could result.

The overall frequency for sequence TBU results from a large number of cut sets that reflect various scenarios that could all lead to a total loss of heat removal following a transient initiator. Among the types of scenarios that were found to contribute the most are the following:

- Loss of main feedwater (as the direct result of an initiating event or due to system faults following another trip initiator), followed by failure of the two turbine-driven AFW pumps (independently or due to a common-cause), with failure of the operators to start the motor-driven feed pump and to initiate makeup/HPI cooling.
- An extended loss of offsite power with failure of the emergency diesel-generators and the station blackout diesel-generator, leading to eventual depletion of the batteries and failure of control power for the turbine-driven AFW pumps. If the operators were unable to control the turbine-driven AFW pumps to preserve flow to the steam generators, a total loss of heat removal would result since, without ac power, use of the motor-driven feed pump or establishment of makeup/HPI cooling would not be viable options.
- Loss of a dc power bus, or loss of dc power following another initiating event, leading to loss of control power for one train of the turbine-driven AFW pumps. If the operators were unable to control flow for the affected train, the associated steam generator could be overfilled, possibly leading to carryover of water to both turbines (the steam supplies would usually be cross-connected automatically after system actuation). Depending on which train of dc power was lost, the ability to start the motor-driven feed pump could be affected directly, or the pump could be unavailable. The loss of one train of dc power would also reduce the number of configurations that would otherwise be available for makeup/HPI cooling.

Sequence TQX

Sequence TQX involves a transient-induced LOCA due to failure of the RCP seals, with failure of long-term cooling after successful injection. The sequence was assessed to have a frequency of approximately 1.9 x 10^{-6} per year, and contributes about 12% to the total core-damage frequency.

The RCP seal failure in this case results primarily from loss of seal return, since the total loss of seal cooling typically results in conjunction with failure of injection (i.e., sequence TQU due to the loss of all CCW). For this scenario, long-term cooling could be accomplished if the operators were able to cool down and establish flow from the DHR system (either in normal shutdown cooling mode or via recirculation from the containment sump), or if they initiated high pressure recirculation before the BWST was depleted. The scenarios that contribute most to the frequency of this sequence include the following:

- Loss of instrument air, which would both cause one of the valves in the portion of the seal return line common to all four RCPs to fail closed, and hamper the ability to cool down the RCS, followed by failure of the operators to initiate high pressure recirculation.
- Loss of dc bus D2P, which would also cause the seal return valve to fail closed, and would cause one train of the systems needed for long-term cooling to be unavailable. In this case, any fault in the opposite train (e.g., of a DHR pump to start) would lead to failure of long-term cooling.
- Loss of seal return by any mode, with failure of the operators to establish some means of long-term cooling, or common-cause failure of the pumps or valves needed for successful long-term cooling.

Sequence FQU

Sequence FQU involves an internal flood that results in the loss of all service water or CCW, and in turn causes a LOCA due to failure of the RCP seals, followed by failure to provide adequate safety injection. This sequence is functionally equivalent to sequence TQU, but is distinguished by the nature of the initiating event. It has an estimated frequency of 1.7 x 10^{-6} per year, and contributes about 10% of the total core damage frequency.

As previously discussed for sequence TQU, the CCW system plays a significant role since, in addition to providing cooling for the RCP seals, it is required to support operation of the injection systems. In addition to CCW system faults, a loss of service water could lead to heatup of the CCW system and an eventual loss of cooling as well. The floods of interest include the following:

- A service water or fire-suppression flood in the CCW pump room (room 328);
- A service water, fire suppression or cooling tower makeup system flood in the service water pump area (room 52);
- A fire-suppression system flood in the room housing the diesel-driven fire-suppression pump (room 51, adjacent to the service water pump room); or
- A service water supply or return line rupture or a cooling tower makeup system line rupture in the service water valve room (room 53).

Sequence MX

This medium LOCA sequence involves successful high pressure and low pressure injection, but failure of low pressure recirculation. This sequence has an estimated frequency of 1.3 x 10^{-6} per year, and contributes approximately 8% of the total core-damage frequency.

Sequence MX is dominated by failure of the operators to initiate recirculation following a medium LOCA. The combined operation of the injection systems and the containment spray pumps would deplete the BWST in less than about 2 hr, and the operators would have on the order of 20 minutes to initiate recirculation once the BWST lo-lo level was reached. Switching to recirculation would entail locally closing the breakers for the suction valves from the sump (DH9A and DH9B) and then opening the valves from the control room.

Sequence AX

This large LOCA sequence involves failure of long-term core cooling following successful low pressure injection by the DHR system. This sequence was estimated to have a frequency of 6.7 x 10^{-7} per year, and contributes approximately 4% of the total core-damage frequency.

As in the medium LOCA sequence described above, sequence AX is dominated by the failure of the operators to initiate recirculation following the LOCA. In the event of a large LOCA, the BWST would be depleted in approximately 44 minutes. The operators would be instructed to initiate switchover to draw suction from the containment sump when the BWST level reached 8 feet. In addition to monitoring tank and sump level, there is an annunciator on lo-lo BWST level at the 8-ft level. This annunciator would be actuated at the same time that an interlock with the sump valves would clear. In addition, there is a separate section within the emergency procedure governing operator response to a large LOCA.

Sequence SX

This small LOCA sequence involves failure of long-term cooling via DHR or recirculation from the containment sump after successful injection. With an estimated frequency of 5.4 x 10^{-7} per year, it contributes about 3% of the total core-damage frequency.

Sequence SX is dominated by a small LOCA followed by failure of both the DHR or HPI pumps (note that high pressure injection could have been accomplished via the makeup pumps in the event of failure of the HPI pumps, but the makeup pumps would not be used for recirculation from the sump following depletion of the BWST). Common-cause failures of the DHR pumps and failures of room cooling contribute most to the unavailability of long-term cooling. A failure associated with the common room cooler return line (containing manual valve SW82) or common-cause failures of all room coolers would result in failure of the DHR pumps.

Sequence TBLX

Sequence TBLX involves a transient initiating event with loss of heat removal via the steam generators. Makeup/HPI cooling succeeds, but feedwater cannot be restored in the long term, and high pressure recirculation fails. Sequence TBLX was assessed to have an annual frequency of 4.8 x 10^{-7}, and contributes about 3.0% of the total core damage frequency.

In the event of a sustained loss of RCS heat removal, makeup/HPI cooling could provide core cooling as long as the inventory of water in the BWST was available. Prior to depletion of the BWST inventory, either feedwater would have to be restored, or high pressure recirculation would have to be established.

It has been estimated that the BWST inventory would reach the setpoint at which the operators would be instructed to initiate high pressure recirculation at about 20 hours after makeup/HPI cooling was started. Depending on the reasons for the early failure of feedwater this provides a substantial amount of time for cooling to be made available to the steam generators.

If makeup/HPI cooling had been initiated as a consequence of a total loss of feedwater, and feedwater could not be restored before the BWST inventory was depleted, it would be necessary to establish high pressure recirculation.

Since TBLX is a long-term sequence involving the failure to restore feedwater, human actions both to restore feedwater or to compensate for other failures (such as the failure to recover power to deenergized components) play a significant role.

Sequence MU

Sequence MU is initiated by a medium LOCA and involves failure of injection by either HPI or DHR. The sequence has an estimated frequency of 3.8 x 10^{-7} per year, and it contributes about 2% of the total core-damage frequency.

Sequence MU is dominated by a medium LOCA followed by failure of both DHR or both HPI pumps. Common-cause failures of the DHR pumps and failures of room cooling for the pumps contribute most to the unavailability of long-term cooling. A failure associated with the common room cooling return line (containing manual valve SW82) or common-cause failures of all room coolers would result in failure of the DHR and HPI pumps.

Sequence FBU

Sequence FBU involves an internal flood that results in the failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. This sequence is functionally equivalent to sequence TBU, but is distinguished by the nature of the initiating event. It has an estimated frequency of 3.4 x 10^{-7} per year, and contributes about 2% of the total core damage frequency.

Sequence FBU is dominated by circulating water floods that result in the loss of some or all of the feedwater pumps. Large floods with failures of automatic detection of the flooding in the turbine building or failure of the circulating water pumps to trip automatically trip contribute most to the frequency of this sequence due to the very significant consequences of these floods.

Sequence V_S

Sequence V_S is postulated to result from a cognitive error that would lead the operators to initiate shutdown cooling via the DHR system during cooldown, but while the RCS pressure was still high. This sequence has an estimated frequency of 3.2×10^{-7} per year, and contributes less than 2% to the total core-damage frequency.

Once plant shutdown has been initiated, the operators would monitor primary system pressure and temperature in order to ensure plant cooldown rates were adhered to. When RCS temperature and pressure were reduced to approximately 280F and 266 psig, respectively, shutdown cooling would be initiated. It has been postulated that the operators might attempt to open the DHR suction valves prematurely (an error of commission that would also require installing jumpers to bypass pressure interlocks), and that high RCS pressure could result in a rupture of the DHR system. If the operators were unable to isolate the break by reclosing one of the DHR suction valves before the BWST inventory was depleted by low pressure injection, core damage could result.

This sequence was first put forth by NRC contractors in NUREG/CR-5604. That document reports the analyses of interfacing-systems LOCAs for a "generic" Babcock & Wilcox plant, which happened to be modeled after Davis-Besse. The NRC obtained a core-damage frequency of 1.1×10^{-6} for this sequence. There remains considerable skepticism that such a scenario is credible, but it has been retained in the current PSA. The difference in the frequency for this sequence relative to that calculated for the NRC stems from a different assumption regarding the RCS pressure regime in which it is even remotely conceivable that the operators might jumper interlocks to allow them to open the DHR suction valves early.

Sequence TBQU

Sequence TBQU involves failure of decay heat removal via the steam generators, a transient-induced LOCA, and failure of high pressure injection. Sequence TBQU was assessed to have an annual frequency of 2.7×10^{-7} which contributes about 2% of the total core damage frequency.

TBQU is initiated by a transient, which is followed by the interruption of heat removal via the steam generators. After the loss of feedwater, loss of RCS integrity results due to either a stuck open relief valve or failure of the RCP seals. Adequate injection to make up for inventory loss and to provide decay heat removal is not available.

Sequence TBQU is dominated by scenarios involving a loss of RCS integrity due to failure of the RCP seals. RCP seal failures account for over 80% of the transient-induced LOCAs in TBQU. A loss of instrument air or a loss of bus D2P initiate most of the seal failure scenarios in sequence TBQU. Both of these initiators result in the loss of seal return, and operator action would be required to trip the RCPs to prevent serious leakage past the seals.

3.1.3 Important Operator Actions

The assessment of human events is one of the most important tasks in performing a PSA. Operating experience has repeatedly demonstrated that human interactions can have a strong influence on the potential for an accident to occur or for one to be avoided. Approximately 150 human actions and 250 combinations of human actions were modeled for the Davis-Besse PSA.

Of the 150 human actions, about 20 were determined to be risk-significant. Those events and the reasons they were assessed to be significant are summarized below.

Tripping Reactor Coolant Pumps Following Loss of Seal Cooling

In the event that all RCP seal cooling is lost, the operators must trip the RCPs to prevent failure of their seals and a consequential small LOCA. Loss of seal cooling involves loss of both cooling of the thermal barriers by the CCW system and seal injection from the makeup system. It is estimated that if the operators do not respond to trip the RCPs, a seal LOCA would result within approximately 30 minutes.

This human action is important in scenarios involving a total loss of CCW or service water. If both trains of CCW or service water were lost either due to equipment failures or due to flooding, a loss of seal cooling to the RCPs would result. The loss of CCW would also cause the loss of cooling to the pumps that might provide injection to the RCS, so that they would be unavailable to respond to a seal LOCA.

Tripping RCPs Following a Loss of Seal Return

In the event of a loss of seal return flow, the operators must trip the RCPs to prevent failure of their seals and a consequential small LOCA. Loss of seal return was assessed to be most likely to occur due to failures associated with the common seal return valve (valve MU38), due to the loss of instrument air, or due to the loss of bus D1 or D2P. If the RCPs were not tripped within approximately 30 minutes following the loss of seal return, a seal LOCA would be expected.

Establishing Makeup/HPI Cooling

In the event that heat removal is not available via the steam generators, it would be necessary to establish a path for the removal of decay heat and to provide adequate injection flow from the makeup system to keep the core cooled. The operators are directed to initiate makeup/HPI cooling whenever the hot leg temperature exceeds 600°F. For the most limiting case of a total loss of feedwater at the time of the plant trip, it is estimated that the hot leg temperature would exceed 600°F about 1 minute after the reactor trip. Makeup/HPI cooling would have to be initiated within about an additional 10 minutes. For cases in which a loss of feedwater occurred at least two hours after the trip, the time available would be much longer. In the PSA, it was estimated that over an hour would be available to initiate makeup/HPI cooling for scenarios involving this delayed loss of feedwater.

Initiating Flow from the Motor-Driven Feed Pump

Following a loss of main feedwater, the turbine-driven AFW pumps would be actuated automatically. It they should fail to start or deliver flow, the motor-driven feed pump could be manually started to serve as a backup. The emergency procedure provides directions for starting the motor-driven feed pump in the event one or both auxiliary feedwater pumps are not available.

This action was assessed to be quite reliable. A more important consideration was the potential for dependence between the failure to start the motor-driven feed pump and failure to establish makeup/HPI cooling after loss of all feedwater. Although these are much different actions, they share some elements of diagnosis and decision-making. If the operators failed to start the motor-driven feed pump, it was assessed to be relatively likely that they would also fail to initiate makeup/HPI cooling.

Recovering Component Cooling Water

The potential for a scenario involving a RCP seal LOCA due to the loss of CCW has been addressed at several points in this report. In the event that a CCW pump failed, the affected train could be recovered using the spare CCW pump. To restore cooling the operators would line up service water to the spare CCW heat exchanger and start the spare pump.

This event is significant in the PSA because the frequency of an initiating event that involves the loss of the running CCW pump was assessed to be relatively high. If the standby pump failed to start, a total loss of CCW would result.

Initiating Low Pressure Recirculation Following a Large or Medium LOCA

In the event of a large LOCA, the combined action of the injection systems and the containment spray pump would deplete the BWST inventory in as little as 45 minutes. The operators would be instructed to initiate the switchover to draw suction from the containment sump when the BWST level reached 8 feet. This would involve locally closing the breakers for the suction valves from the BWST and then opening the valves. For a medium LOCA the actions are similar, but the time for BWST depletion to occur would be longer.

Large and medium LOCAs constitute only moderate contributors to the total core-damage frequency. This action is important, however, because the successful switchover to low pressure recirculation is required for all large and medium LOCAs.

Manually Controlling AFW Pump Speed

If dc power were lost to an AFW flow control valve (AF6451 or AF6452), the valve would fail fully open. Without operator intervention, the affected steam generator could overfill, and carryover of water into the steam lines could affect continued operation of both turbine-driven AFW pumps (since their steam supplies would be cross-connected). Procedural guidance is provided for the operators to take local manual control of the turbine speed for the AFW pump in the affected train to limit flow to the steam generator.

This event could occur in scenarios involving a dc bus fault or a loss of power from 4160v bus C1 or D1 bus followed by battery depletion. The scenario is significant because it could lead to the loss of both turbine-driven AFW pumps.

Recovering Service Water

If all three of the service water pumps were lost, either due to common cause failure or internal flooding, the dilution pump might provide an option for restoring service-water flow.

Failure to restore service water flow would eventually result in the loss of CCW, which could lead to a seal LOCA with no high pressure injection capability.

Using the dilution pump to restore service-water flow was evaluated in the PSA to be more risk significant than restoration using the spare service water pump. This is because the dilution pump is in a different location and is less susceptible to common-cause failures that might be shared with the other service water pumps. This action is less risk significant than the recovery of CCW using the spare CCW pump, however, for several reasons. First, the CCW system does not have an equivalent to the dilution pump so this puts more significance on using the spare CCW pump. Additionally, two pumps are normally running in the service water system, whereas only one CCW pump is usually operating, with a second pump in standby (and the third pump isolated as a spare). The probability of failure of the standby CCW train given the loss of the running train is higher than the probability of failure of a running service water pump given the loss of the other train.

Initiating High Pressure Recirculation Following a Transient-Induced LOCA

In the event of a small LOCA induced by a transient (i.e., due to a RCP seal failure or a stuck-open pressurizer relief valve), it may not be possible to cool down and establish shutdown cooling via the DHR system prior to depleting the BWST. Therefore, the operators would be required to initiate high pressure recirculation from the emergency sump.

This action s similar to initiating low pressure recirculation after a large or medium LOCA, as described above, except that much more time would be available before recirculation would be required for a small LOCA. Therefore, this event was evaluated to be less risk significant than establishing recirculation for a large or medium LOCA despite the greater frequency of transient-induced LOCAs.

Adding Fuel for the Station Blackout Diesel-Generator

In the event of an extended loss of offsite power it could be necessary to refuel the station blackout diesel-generator if one or both of the emergency diesel-generators were to fail. The fuel tank normally contains sufficient fuel to allow for 8 to 10 hours of operation at full load.

This is a risk-significant human action because the station blackout diesel-generator is important in scenarios involving a loss of offsite power. Although this generator must be started and refueled manually, it is an important option for maintaining electric power because it is not susceptible to some of the sources of common-cause failure shared by the two emergency diesel-generators.

The action to add fuel to the fuel oil tank for the station blackout diesel-generator is subject to very limited procedural guidance, and would depend on the specific conditions at the time of the emergency demand. These considerations entered into an assessment of a relatively high probability of failure for this action.

Isolating Flooding in the CCW Pump Room

A large rupture in the service water piping or fire protection piping in the CCW pump room could cause flooding and a loss of all three CCW pumps. Although the frequency of such an event is low, the effects could be severe. In some cases, it would be possible for the operators to isolate the source of the flooding before the CCW pumps were affected.

Establishing Feedwater Flow from the Startup Feed Pump

In the event of a total loss of feedwater, including the failure of both turbine-driven pumps and the motor-driven feed pump, the operators could use the startup feedwater pump as an additional backup means of supplying flow to the steam generators.

Establishing flow from the startup feed pump to a steam generator would require a series of actions, since the pump is isolated both electrically and mechanically. Therefore, this event has the highest failure probability of any of the proceduralized events analyzed in the PSA.

Aligning Alternate Makeup Room Ventilation

As part of implementing makeup/HPI cooling, the operators are instructed to establish flow from both makeup pumps. The heat load produced in the makeup pump room when both pumps are operating would be sufficient to require room cooling. The fans that serve the room, however, are non-essential and would not be powered in the event of a loss of offsite power. If room cooling were not available, the emergency procedure would instruct the operators to prop open the makeup pump room door to provide adequate ventilation. Analysis has shown that opening the door within 30 minutes of starting both pumps without room cooling would prevent the room from reaching unacceptably high temperatures.

Tripping Circulating Water Pumps Following a Flood

In the event of a flood in the turbine building from the circulating water system and the failure of the automatic pump trip system, the operators could take action to stop the flooding before it reached the 585 foot elevation. At that point, equipment that would play a significant role in core cooling could be affected (including the AFW pumps and high-voltage switchgear room A).

The frequency of a circulating water flood with the failure of the automatic trip system is very low, but the consequences of a circulating water flood that caused flooding to the 585 foot level could be severe. In the worst case, if the flood inundated the AFW pump rooms and reached the high-voltage switchgear room, the ability to maintain core cooling could be reduced to using a single makeup pump and the PORV for makeup/HPI cooling.

Opening Valve SW82 Following Maintenance

Valve SW82 is the manual isolation valve in the service water return line for the ECCS room coolers. It is in a portion of the return line that is common to all five ECCS room coolers. If it were left closed after some maintenance activity, the lack of flow through the coolers would constitute a potential source of common-cause failure.

This action differs from the others discussed in this section in that it relates to a human interaction prior to a plant trip (i.e., the failure would leave equipment unavailable when it was needed, rather than involving a lack of proper response).

Closing DH11 and DH12 to Isolate an Interfacing Systems LOCA

This action is in response to a postulated human error of commission that involves prematurely opening the DHR suction valves while cooling down the RCS to cold shutdown. This action was postulated to lead to overpressurizing the DHR system piping outside containment, leading to an interfacing-systems LOCA.

The potential for this error was identified in a study the NRC performed for a "generic" B&W design that used Davis-Besse as its basis [Ref. 12]. There is substantial uncertainty regarding the actual potential for such an error (it would, for example, require a conscious decision to violate significant administrative procedures and install a jumper to permit opening one of the suction valves). Nevertheless, the error was retained in the current study for completeness.

If the LOCA were to occur, it would still be possible to isolate the break and to preserve sufficient inventory in the RCS to prevent core damage. The scenario in question would not result in core damage until after the BWST inventory was depleted. Because the reactor coolant lost through the break would not be collected in the containment emergency sump, there would be no means to support recirculation in the long term. Isolating the break by closing valves DH11 or DH12 would eliminate the LOCA, and permit core cooling to be sustained. A period of some hours would be available to take this action. Unlike most other responses considered in the PSA, however, there is no procedural guidance related directly to isolating such a break. The amount of time available would generally give the operators ample opportunity to identify the need to isolate the break and to act.

3.1.4 Importance of Individual Components

Examining the contributions to core-damage frequency can provide further insight into the importance of various plant features. Using the PSA results to highlight risk-significant components is also one of the tasks required to develop the technical information needed to implement the Maintenance Rule.

One of the most useful measures of component importance is referred to as the risk achievement worth (RAW). This measure of importance is calculated by determining the ratio of the core-damage frequency with the probability of failure for the component set to 1.0 to the nominal core-damage frequency. This measure is most useful for determining the effects on risk associated with removing equipment from service for maintenance. It is also valuable from the perspective of equipment for which maintaining a high degree of reliability is important. Table 6 summarizes the RAW values for the more important components.

Table 6. Effect on Core-Damage Frequency with Significant Components Out of Service

Risk Achievement Worth	Component
> 100x	4160v bus D1 480v bus F1
50 to 100x	4160v bus C1
10 to 50x	Motor-driven feed pump Standby component cooling water pump Station battery 480v bus E1
5 to 10x	Auxiliary feedwater pump Decay heat removal pump
2 to 5x	Emergency instrument air compressor Pressurizer PORV Dilution pump (backup service water pump) Station blackout diesel-generator High pressure injection pump Emergency ventilation for a low-voltage switchgear room
< 2x	Emergency diesel-generator Makeup pump Spare component cooing water pump Spare service water pump

Another way to characterize component importance is to calculate the actual contribution to core-damage frequency due to a particular failure. This is equivalent to the percentage by which the core-damage frequency could be reduced if the component could be made 100 percent reliable. This measure is referred to as the Fussell-Vesely importance. An equivalent measure, referred to as the risk reduction worth (RRW), can be calculated as the ratio of the nominal core-damage frequency to the frequency that would be obtained if the probability of failure for a component were set to zero. Figure 2 illustrates the components with the highest values for Fussell-Vesely importance (the ranking would be the same for risk reduction worth). This measure is essentially complementary to the risk achievement worth. Such a ranking is valuable because it can be used as a guide in determining where resources could be applied most effectively in reducing the frequency of core damage.

A comparison of the listing in Table 6 to the ranking of components in Figure 2 reveals that a component that has a high RAW may not necessarily contribute significantly to the frequency of core damage. For example, the motor-driven feed pump has a RAW value greater than 10 (i.e., if the pump were out of service, core-damage frequency would increase by more than 10). According to the Fussell-Vesely ranking, however, the failure of the pump contributes only about 2% of the core-damage frequency. In contrast, the RAW value for the turbine-driven AFW pumps is lower than that for the motor-driven feed pump, but the contribution to core-damage frequency is about 12%. The higher reliability of the motor-driven pump makes it a lower contributor to the frequency of core damage than the turbine-driven pumps. Because it is diverse from the turbine-driven pumps, if the motor-driven feed pump were actually unavailable the effect would be a more significant increase in core-damage frequency.

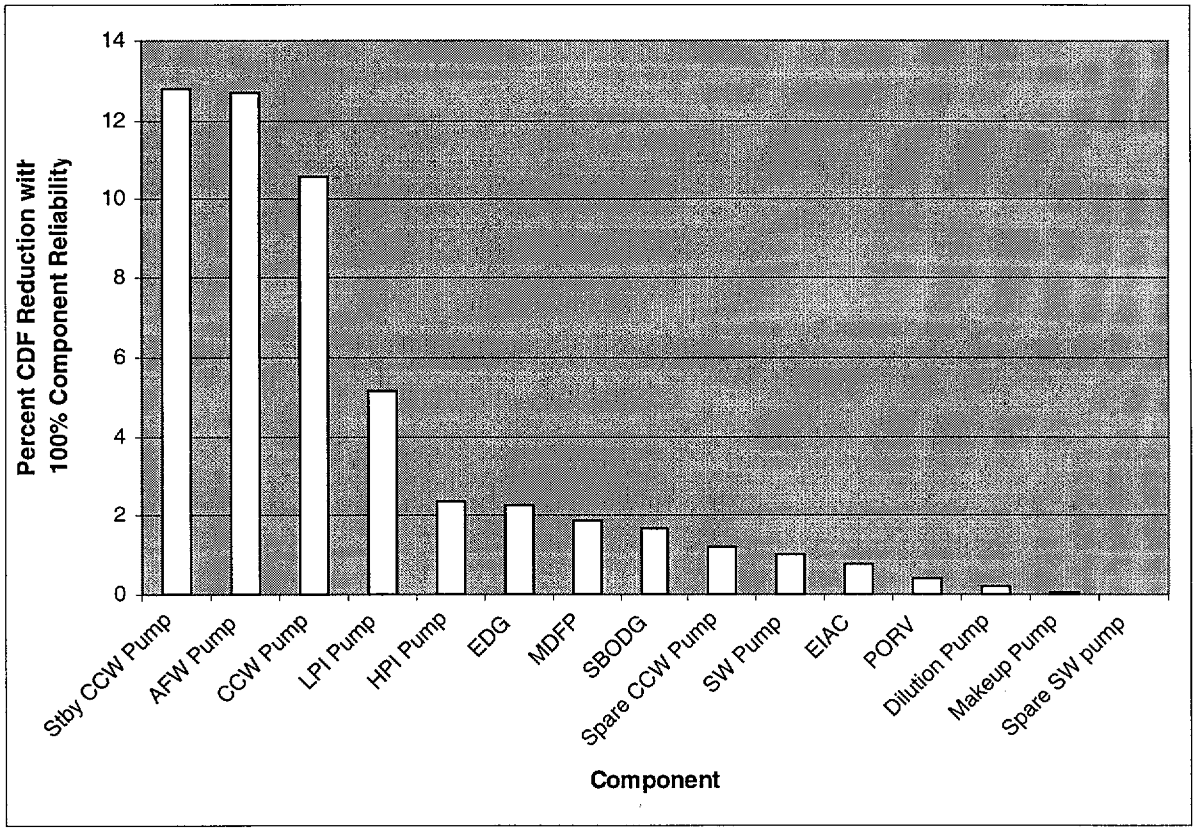

Figure 2. Components Whose Failures Contribute Most to Core-Damage Frequency

Some of the insights gained from considering these importance measures are addressed in the discussion that follows.

Electrical Power Buses

The three components with the highest RAW values are all electrical buses (4160v buses C1 and D1 and 480v bus F1). These components rank high according to RAW, but are very small contributors to the nominal core-damage frequency (they do not even appear in Figure 2).

The reason for this apparent contradiction is that these buses are quite reliable. Therefore, the frequencies of core-damage accidents in which their failures would play a role tend to be low. If any of the buses were unavailable, however, a significant number of systems could be affected, leading to a higher potential increase in core-damage frequency.

Component Cooling Water Pumps

The standby CCW pump has a relatively high level of importance for both measures. This is because the pump is the most important option for restoring CCW flow if the normally operating pump should fail. If the standby CCW pump were unavailable, restoration of CCW flow would rely on use of the spare pump.

It should be noted that this characterization is, to a certain extent, artificial. If a CCW pump were known to be unavailable, it would become the spare pump, and another pump would be configured as the standby. Again, however, this ranking provides an indication of the importance of keeping the reliability of the standby pump as high as possible. In fact, with all other factors being equal (which, of course, they never are), Figure 2 would indicate that efforts to reduce core-damage frequency further would be most effective if they were to begin with consideration of improving the reliability of the standby CCW pump.

Diesel-Generators

The emergency diesel-generators are not especially high in either importance ranking. This may seem contrary to expectations, but there are clear reasons for this. First, the addition of the station blackout diesel-generator has improved the overall reliability of emergency power for the station. With respect to the potential for accidents involving station blackout, Davis-Besse also benefits somewhat by having two turbine-driven pumps to supply auxiliary feedwater. Either of these may be capable of operating with no ac power. Most PWRs have only one such pump.

Non-Safety Equipment

The results in Table 6 and Figure 2 illustrate that equipment that is not necessarily safety-related can play an important role. Among the important equipment are the motor-driven feed pump, the emergency instrument air compressor, the dilution pump, and the station blackout diesel-generator.

In a PSA, no attempt is made to distinguish between safety-related equipment and equipment that is not safety-related. All of the options for achieving a particular safety function are investigated, and an attempt is made to characterize the reliability of each option in the most realistic and appropriate manner possible. Thus, the motor-driven feed pump can be important, for the reasons cited above. The emergency instrument air compressor is important because it can support functions such as seal return for the RCPs. The dilution pump provides a somewhat diverse means to restore service water if the three normal service water pumps are unavailable (due to flooding, failure of room cooling, or other common causes). The station blackout diesel-generator is important because it provides further redundancy for emergency power. This is valuable because diesel-generators tend to be among the least reliable of important types of safety equipment. This generator can also be used as a backup to either of the emergency generators.

3.1.5 System Importance

System importance can be calculated using methods similar to those used for individual components. Since entire systems are not normally removed from service, the system importance for this report was evaluated by calculating the percent reduction in core-damage frequency that would be achieved by 100% reliable system performance. As with individual components, this method is useful because it can be used as a guide in determining where resources can be applied in improving plant safety. The Fussell-Vesely rankings by system are illustrated in Figure 3.

The results of this importance assessment are very consistent with the discussions of the breakdowns of the results in the preceding sections. Auxiliary feedwater (in this instance, including the motor-driven feed pump) is very important because of the fraction of core-damage sequences due to total losses of feedwater. The potential for failure of the CCW system is also important, although its contribution to the frequency of core damage due to RCP seal LOCAs is shared with the service water system.

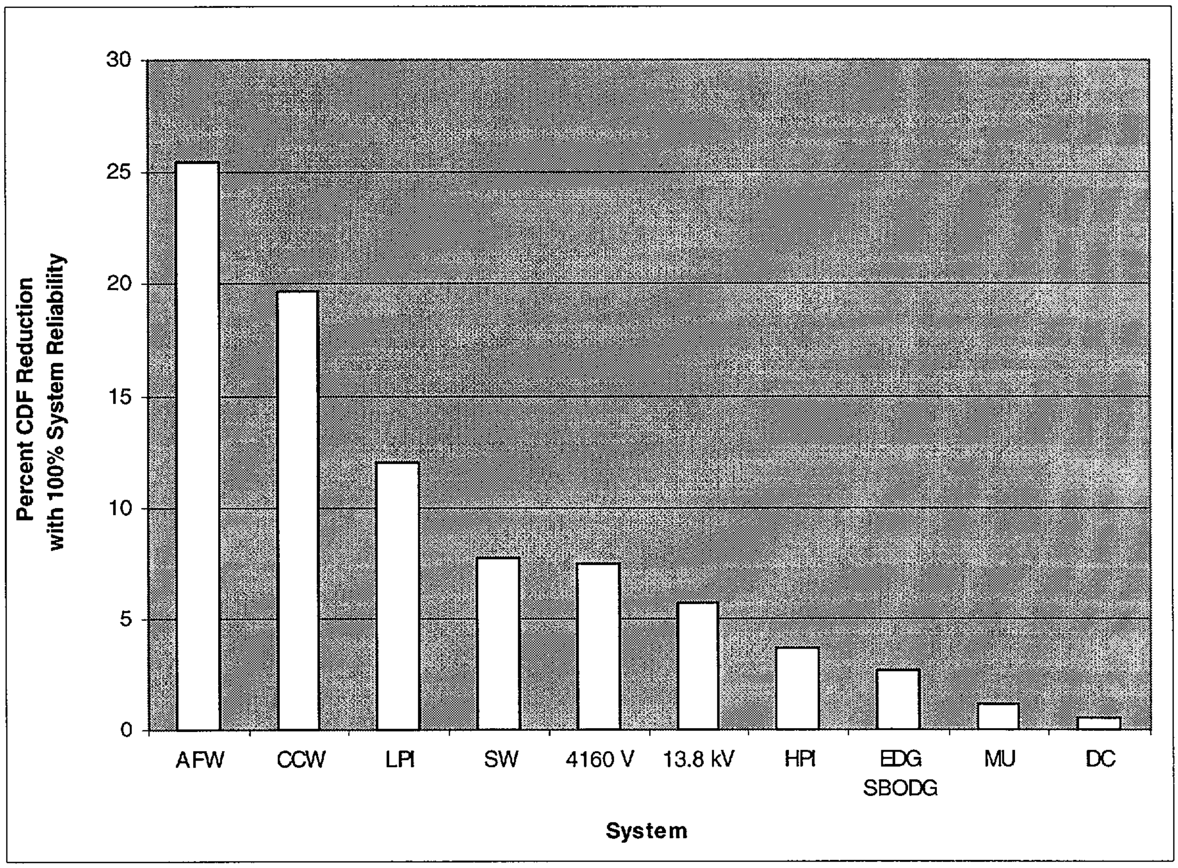

Figure 3. Systems Contributing Most to Core-Damage Frequency

3.2 Comparison to Results for the IPE

The PSA for the IPE was essentially completed in late 1992, and reflected the plant as it existed even earlier than that. Changes to the plant, and especially to particular operating and maintenance practices, have contributed to a reduced estimate of the core-damage frequency and, to some extent, a shift in the relative contributors to that frequency. Some of the more important of these differences are addressed in the discussion that follows. For reference purposes, Table 7 provides a comparison of the frequencies for the core-damage sequences as they were assessed in the update to those from the PSA for the IPE.

Table 7. Comparison of Sequence Frequencies

Sequence	Description	IPE Results		Update Results	
		Annual CDF	Contrib. (%)	Annual CDF	Contrib. (%)
Large and Medium LOCAs					
MX	Medium LOCA initiating event with failure of low pressure recirculation	1.6×10^{-6}	2.4	1.3×10^{-6}	8.0
AX	Large LOCA initiating event with failure of low pressure recirculation	8.7×10^{-7}	1.3	6.7×10^{-7}	4.1
Av	Reactor vessel rupture initiating event	4.6×10^{-7}	0.7	4.6×10^{-7}	2.8
MU	Medium LOCA initiating event with failure of high or low pressure injection	4.6×10^{-7}	0.7	3.8×10^{-7}	2.3
AU	Large LOCA initiating event with failure of low pressure injection	2.1×10^{-7}	0.3	1.0×10^{-7}	0.6
Totals for large and medium LOCAs		3.6×10^{-6}	5.4	2.9×10^{-6}	17.9
Small LOCAs					
SX	Small LOCA initiating event with failure of long term cooling via DHR or recirculation from sump	1.5×10^{-6}	2.9	5.4×10^{-7}	3.3
SU	Small LOCA initiating event with failure of injection	5.9×10^{-7}	0.9	8.4×10^{-8}	0.5
SCX	Small LOCA with failure to cool down and failure of high pressure recirculation	Not Modeled	-	4.6×10^{-8}	0.3
SBU	Small LOCA initiating event with failure of feedwater and failure of makeup/HPI cooling	3.8×10^{-8}	< 0.1	8.1×10^{-9}	< 0.1
SBX	Small LOCA initiating event with failure of feedwater and failure of high pressure recirculation	$< 10^{-8}$	< 0.1	5.2×10^{-10}	< 0.1
Totals for small LOCAs		2.1×10^{-6}	3.2	6.7×10^{-7}	4.1

Table 7. Comparison of Sequence Frequencies (continued)

Sequence	Description	IPE Results		Update Results	
		Annual CDF	Contrib. (%)	Annual CDF	Contrib. (%)
Steam Generator Tube Ruptures*					
RCX	SGTR with failure of cooldown via intact steam generator and failure of long term cooling	1.1×10^{-7}	0.2	1.3×10^{-7}	0.8
RLX	SGTR with failure to isolate ruptured steam generator and failure of long term cooling	1.9×10^{-7}	0.3	6.1×10^{-8}	0.5
RCU	SGTR with failure of injection and failure to cooldown via the intact steam generator	(included with RUL)		2.5×10^{-8}	0.2
RB_IU	SGTR with loss of feedwater to intact steam generator and failure of makeup/HPI cooling	(included with RCU)		(included with RCU)	
RCLX	SGTR with failure of cooldown via intact steam generator, failure to isolate the ruptured steam generator and failure of long term cooling	(included with RUL)		1.7×10^{-8}	0.1
RUL	SGTR with failure of injection and failure to isolate the ruptured steam generator	$< 10^{-8}$	< 0.1	6.3×10^{-9}	< 0.1
RB_IB_AU	SGTR with loss of feedwater to both steam generators and failure of makeup/HPI cooling	4.2×10^{-8}	< 0.1	4.0×10^{-9}	< 0.1
RB_IB_AX	SGTR with a loss of feedwater to both steam generators and failure of long term cooling	1.1×10^{-7}	0.2	4.7×10^{-10}	< 0.1
Totals for steam generator tube ruptures		4.6×10^{-7}	0.7	2.4×10^{-7}	1.5
*Note that, for steam generator tube ruptures the sequence breakdown for the update was substantially different from that for the IPE. Therefore, direct comparison of the result is somewhat difficult.					
Interfacing-Systems LOCAs					
V_SI_S	Interfacing-systems LOCA due to premature opening of DHR suction valves and failure to isolate the break	9.1×10^{-8}	0.1	3.2×10^{-7}	2.0
V_DI_D	Interfacing-systems LOCA due to hardware failure of DHR suction valves and failure to isolate break	1.7×10^{-7}	0.3	1.1×10^{-7}	0.7
V_LI_L	Interfacing-systems LOCA due to a failure in LPI injection line and failure to isolate break	5.6×10^{-7}	0.8	4.9×10^{-8}	0.3

Table 7. Comparison of Sequence Frequencies (continued)

Sequence	Description	IPE Results		Update Results	
		Annual CDF	Contrib. (%)	Annual CDF	Contrib. (%)
Interfacing-Systems LOCAs (continued)					
$V_H I_H$	Interfacing-systems LOCA due to failure in HPI injection line and failure to isolate break	6.4 x 10^{-8}	01	5.1 x 10^{-9}	< 0.1
Total for interfacing system LOCAs		8.8 x 10^{-7}	1.3	4.8 x 10^{-7}	3.0
Transients					
TQU	Transient initiating event with RCP seal LOCA and failure of injection	1.4 x 10^{-5}	21.2	3.7 x 10^{-6}	22.8
TBU	Transient initiating event with a total loss of feedwater and failure of makeup/HPI cooling	3.5 x 10^{-5}	53.0	3.3 x 10^{-6}	20.3
TQX	Transient with RCP seal LOCA and failure of long term cooling	4.3 x 10^{-6}	6.5	1.9 x 10^{-6}	11.7
TBLX	Transient with extended loss of feedwater and failure of high pressure recirculation	3.2 x 10^{-7}	0.5	4.8 x 10^{-7}	3.0
TBQU	Transient with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of makeup/HPI cooling	2.9 x 10^{-6}	4.4	2.7 x 10^{-6}	11.7
TKBP	Transient with failure to trip, loss of main feedwater and excessive RCS pressure	1.7 x 10^{-7}	0.3	1.2 x 10^{-7}	0.7
TBQX	Transient with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of HPR	2.9 x 10^{-7}	0.4	8.6 x 10^{-9}	0.1
TKBU	Transient with failure to trip, loss of main feedwater and failure to provide borated water	1.6 x 10^{-7}	0.3	4.1 x 10^{-9}	< 0.1
TKBL	Transient with failure to trip and total loss of feedwater	2.4 x 10^{-8}	< 0.1	7.0 x 10^{-9}	< 0.1
TBWX	Transient with initial loss of feedwater, successful makeup/HPI, stuck open relief when feedwater is restored and failure of HPR	< 10^{-8}	< 0.1	8.1 x 10^{-9}	< 0.1
TBP	Transient with a total loss of feedwater and failure of pressurizer relief valves to open	< 10^{-8}	< 0.1	1.7 x 10^{-10}	< 0.1
Total for transients		5.7 x 10^{-5}	86.3	9.8 x 10^{-6}	60.3

Table 7. Comparison of Sequence Frequencies (continued)

Sequence	Description	IPE Results		Update Results	
		Annual CDF	Contrib. (%)	Annual CDF	Contrib. (%)
Internal Floods					
FQU	Internal flood with RCP seal LOCA and failure of injection	1.9×10^{-6}	2.9	1.7×10^{-6}	10.5
FBU	Internal flood with a total loss of feedwater and failure of makeup/HPI cooling	3.9×10^{-8}	0.1	3.4×10^{-7}	2.1
FBLX	Internal flood with extended loss of feedwater and failure of high pressure recirculation	$< 10^{-8}$	< 0.1	4.5×10^{-8}	0.3
FQX	Internal flood with RCP seal LOCA and failure of long term cooling	1.2×10^{-8}	< 0.1	4.2×10^{-8}	0.3
FBQU	Internal flood with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of makeup/HPI cooling	$< 10^{-8}$	< 0.1	2.0×10^{-8}	0.1
FBQX	Internal flood with total loss of feedwater, RCP seal LOCA or stuck open relief, and failure of HPR	$< 10^{-8}$	< 0.1	9.4×10^{-10}	< 0.1
FKBP	Internal flood with failure to trip, loss of main feedwater and excessive RCS pressure	$< 10^{-8}$	< 0.1	5.1×10^{-10}	< 0.1
FKBU	Internal flood with failure to trip, loss of main feedwater and failure to inject borated water to achieve shutdown	$< 10^{-8}$	< 0.1	1.0×10^{-10}	< 0.1
FBWX	Internal flood with initial loss of feedwater, successful makeup/HPI, stuck open relief when feedwater is restored and failure of HPR	$< 10^{-8}$	< 0.1	8.1×10^{-9}	< 0.1
FKBL	Internal flood with failure to trip and total loss of feedwater	$< 10^{-8}$	< 0.1	7.0×10^{-9}	< 0.1
FBP	Internal flood with a total loss of feedwater and failure of pressurizer relief valves to open	$< 10^{-8}$	< 0.1	1.7×10^{-10}	< 0.1
Total for internal floods		2.0×10^{-6}	3.0	2.0×10^{-6}	13.2
Total core-damage frequency		6.6×10^{-5}		1.6×10^{-5}	

Initiating Event Frequencies

Several of the frequencies for initiating events were based primarily on operating experience at Davis-Besse. The experience upon which these frequencies were based for the previous PSA largely reflected performance prior to the 1985 loss of feedwater event and the subsequent extended shutdown. Since that time, frequencies for the more common types of plant trips have decreased dramatically. For the PSA update, this was very important relative to the frequency of general plant transients, and to the frequency of loss of main feedwater in particular. The reductions in these two frequencies (from 6.0 per year to 1.2 per year for plant trips and 1.7 per year to 0.22 per year for loss of feedwater events) were among the most important reasons for the overall decrease in the core-damage frequency.

The effects of the lower initiating event frequencies are most clearly seen with respect to sequence TBU (a transient initiating event with a total loss of feedwater and failure of makeup/HPI cooling). The frequency of this sequence, which accounted for 53% of the core-damage frequency calculated for the IPE, has dropped in frequency by over a factor of ten in the update.

Modeling of Electric Power Provisions

The contribution to core-damage frequency associated with the loss of offsite power as assessed for the IPE was an order of magnitude larger for Davis-Besse than was evaluated for the other Babcock & Wilcox plants. Review during the update effort indicated that this contribution was due in large measure to simplifying assumptions made in the previous PSA that had the effect of overstating the core-damage frequency. Several changes were made in the modeling effort for the PSA update to address these simplifications. Among the most important were the following:

- At the time of the PSA for the IPE, not all procedures for the use of the station blackout diesel-generator has been put into place. In particular, no provisions had been made for replenishing the fuel tank for the generator during an extended emergency demand. Therefore, in the previous PSA, it was assumed that the station blackout diesel-generator would be available for only 6 hours if it were called upon (based on information available at the time, that duration would correspond to the average amount of fuel in the tank). For the PSA update, refueling the generator was explicitly taken into account, and the contribution to core-damage frequency of sequences involving an extended loss of offsite power decreased significantly.
- The potential for a loss of offsite power subsequent to plant trip due to another cause (e.g., failure to switch sources to the startup transformers, or inducing a transient in the grid due to the loss of the Davis-Besse generating capacity) was somewhat important for the previous PSA. In the update, this potential was assessed in substantially more detail, and the overall contribution to total loss of offsite power decreased markedly.
- The update incorporated further detail relating to the ability to supply power to either division from an available bus, from the station blackout diesel-generator, or from one of the emergency diesel-generators. Some of these options were neglected in the previous PSA.

Loss and Recovery of Service Water and Component Cooling Water

The fault trees for both the service water and CCW systems have been revised extensively. These revisions have made it much easier to account for opportunities for the operators to recover from various system failures by using the standby or spare pumps in these systems or (in the case of service water) by lining up the dilution pump. The overall effect on core-damage frequency as a result of these changes is relatively modest. The changes do, however, make it much easier to apply the models to investigate a wide variety of issues.

Modeling of Circulating Water Floods

The PSA for the IPE rejected the possibility that flooding in the turbine building due to gross failures in the circulating water system could be important. This was based on the judgment that floods would cause the circulating water pumps to fail before levels could rise above the basement of the turbine building. Once the pumps failed the driving head for further flooding would be lost.

More careful analysis since that time determined that it would, indeed, be possible to flood higher elevations before the circulating water pumps stopped. As a result of these analyses, a new system for detecting the presence of flooding and for tripping the circulating water pumps has been installed. The updated PSA includes both the potential for core damage to result from a circulating water flood, and credit for terminating the flood based on the new protective features. The contribution from this source is therefore modest.

Hardware Failure Rates

The reliability of some classes of equipment has improved, as indicated by the assessment of failure rates since the IPE was submitted. The most dramatic improvement is associated with motor-operated valves. The failure rate for these valves, based on plant-specific evidence, has decreased from about 0.008 to about 0.0017 per demand. The overall effect of this decrease on core-damage frequency is modest (about 10%) but noticeable.

Section 4
Results from the Update of the Back-End Analyses

This section presents the results obtained from updating the back-end or level 2 analyses. These results include the frequencies of core-damage accidents that result in various failure modes for the containment failure modes and discussion of the large early release frequency (LERF). For this update the large early release frequency has been calculated to be about 7.3 x 10^{-8} per year. The large early release frequency was not calculated for the IPE, so the updated value can not be easily compared to an equivalent IPE result. The updated LERF, however, appears to represent a substantial reduction from the value that would have been estimated using the IPE results. As absolute values, the LERF and the frequencies of the various containment failure modes are not particularly meaningful. A breakdown of the contributors to these results does, however, offer useful insights. The sequences and features that contribute to the important containment failure scenarios are discussed in Section 4.1. Section 4.2 focuses on the contributors to LERF.

4.1 Frequencies of Containment Failure Modes

All of the core-damage sequences have some potential to result in a release from containment. The containment may, as a result of the challenges presented by the severe accidents, fail in various ways and at various times. For many sequences, the containment would be able to retain its integrity, despite loadings that may exceed its design basis. The remainder of the outcomes have been collected into the following categories of containment failure:

- Early containment failure, including failure of containment isolation and failure due to phenomena such as hydrogen burns before or around the time of reactor vessel failure;
- Containment sidewall failure, which could occur due to ablation of the concrete curb at the basement level of containment, leading to containment failure in the intermediate to late time frame;
- Containment bypass, encompassing interfacing-systems LOCAs and SGTRs (including those that initiated an accident sequence and those that resulted from creep rupture during core degradation);
- Late containment failure, due to slow overpressurization or burning of combustible gases long after vessel failure;
- Small isolation failures, including containment leakage paths that do not cause appreciable containment depressurization.
- Large isolation failure, encompassing containment leakage paths that do cause containment depressurization.

Table 8 summarizes the results for each of the containment failure outcomes identified above. The conditional probability represents the fraction of the overall core-damage frequency that results in each of the outcomes. The frequency is the total frequency of core-damage sequences that lead to each outcome.

Table 8. Probabilities and Frequencies of Containment Failure Categories

Containment Failure Category	Conditional Probability Given Core Damage	Annual Frequency
No failure	0.93	1.4×10^{-5}
Small isolation failure	0.020	3.1×10^{-7}
Containment bypass	0.020	3.0×10^{-7}
Basemat meltthrough	0.017	2.5×10^{-7}
Late containment failure	0.0088	1.3×10^{-7}
Early containment failure	0.0034	5.1×10^{-8}
Containment sidewall failure	0.001	1.4×10^{-8}
Large isolation failure	2.6×10^{-5}	3.9×10^{-10}

As the results in Table 8 indicate, the containment would be expected to retain its integrity for approximately 93% of the sequences that comprise core damage frequency. For those cases that comprise no containment failure, the core debris would be in a cooled state and containment heat removal would be functioning to limit the pressure rise inside containment.

The sections that follow provide more detailed discussion concerning the containment failure modes and the plant-damage states that are the primary contributors to them.

4.1.1 Plant-Damage States Contributing to Early Containment Failure

Early containment failure represents a small fraction of outcomes that correspond to loss of containment integrity prior to or near the time at which core debris penetrates the reactor vessel. The potential for early failure arises due to several types of low probability challenges. The most significant of these include high pressure melt ejection, hydrogen burn after vessel breach, and steam explosion at vessel breach.

The frequency of early containment failure is distributed among a number of plant-damage states. There are, however, a relatively small number of states that are responsible for a significant portion of the frequency. Table 9 summarizes these states; the manner in which each contributes to the potential for early containment failure is described briefly below.

Table 9. Plant-Damage States Contributing to Early Containment Failure

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
SIY_36Y	1.5×10^{-8}	30%	• Seal LOCA with failure of injection but feedwater available • Containment air cooling and containment spray are available • Low pressure recirculation not available to restore core cooling late
TIN_18Y	7.8×10^{-9}	15%	• Transient with total loss of feedwater and early failure of core cooling • Reactor coolant pumps, containment air cooling, containment spray, PORV and low pressure recirculation are available
MRX_02Y	5.2×10^{-9}	10%	• Medium LOCA with failure of cooling in recirculation phase • Containment air cooling and containment spray are available
SRY_39Y	2.6×10^{-9}	5.0%	• Seal LOCA with failure of cooling in recirculation phase • Containment air cooling is available • Containment spray and low pressure recirculation fail
ARX_02Y	2.6×10^{-9}	5.0%	• Large LOCA with failure of core cooling in the recirculation phase • Containment air cooling and containment spray available
TIN_35Y	2.5×10^{-9}	5.0%	• Transient with total loss of feedwater and early failure of core cooling • Containment air cooling, containment spray, PORV and low pressure recirculation are available • Reactor coolant pumps not available
SRY_36Y	2.4×10^{-9}	4.8%	• RCP seal LOCA with failure of core cooling in the recirculation phase. • Ac power and feedwater are available • Containment air cooling, containment spray, and BWST injection fail
SIY_36Y	1.9×10^{-9}	3.7%	• RCP seal LOCA with failure of injection • Ac power and feedwater are available • Containment air cooling, containment spray, and BWST injection fail
Other	1.1×10^{-8}	21%	
Total	5.1×10^{-8}	100%	

Plant-Damage State SIY-36Y

Plant-damage state SIY_36Y includes sequences initiated by transients that result in a RCP seal LOCA. Core damage occurs as a result of loss of reactor coolant due to the failure of high pressure injection, but feedwater remains available. Containment air cooling and containment spray are available, but low pressure recirculation (which might otherwise be able to allow core damage to be arrested before the reactor vessel failed) is not available. The frequency of containment failure associated with this plant-damage state was calculated to be 1.5 x 10^{-8}; this accounts for about 30% of the total frequency of early containment failure.

For this plant-damage state, the probability of early failure is distributed among several phenomena, including high pressure melt ejection, hydrogen burn after vessel breach and steam explosion at vessel breach.

The frequency of this plant-damage state is dominated by scenarios initiated by a loss of component cooling water (CCW). Component cooling water is required for operation of the makeup and injection systems as well as for providing cooling for the RCP seals. Containment spray and containment air cooling are not, however, dependent on CCW, so they are available to mitigate the effects on containment from this plant-damage state. This plant-damage state does not contribute to LERF because the containment spray pumps are available to provide fission produce scrubbing.

Plant-Damage State TIN-18Y

Plant-damage state TIN_18Y includes sequences initiated by a transient that entail a total loss of feedwater and failure of makeup/HPI cooling. Systems that could affect containment response, including containment air cooling, containment spray, low pressure recirculation, and the PORV, are available. The annual frequency of early containment failure associated with this plant-damage state was estimated to be 7.8 x 10^{-9}, which accounts for about 15% of the total.

Accidents initiated by a loss of main feedwater or a plant trip account for a large part of the contribution to the frequency of early failure for this plant-damage state. A typical scenario found to contribute to the frequency of this plant-damage state would be the loss of main feedwater, followed by the failure of the two turbine-driven AFW pumps, with failure of the operators to start the motor-driven feed pump and to initiate makeup/HPI cooling.

Plant-Damage State MRX-02Y

Plant-damage state MRX_02Y involves a medium LOCA with failure of core cooling in the recirculation phase. Containment systems, including containment air cooling, and containment spray, are available. The early containment failure frequency for this plant-damage state was estimated to be 5.2 x 10^{-9}, which accounts for about 10% of the total.

4.1.2 Containment Sidewall Failure

Sidewall failure may result if there is substantial dispersal of core debris to the lower compartment of the containment (also referred to as the basement level). Plant-damage states for which the pressure in the RCS would be relatively high just prior to failure of the reactor failure are most likely to contribute to this failure mode. Because core debris generally must remain uncooled for the attack on the sidewall to progress, plant-damage states with failed containment systems contribute more to sidewall failure.

Table 10 summarizes the plant-damage states that contribute most to containment sidewall failure. Each of these is described briefly below.

Table 10. Plant-Damage States Contributing to Containment Sidewall Failure

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
TIN_49S	4.3×10^{-9}	30%	• Transient with feedwater failure and early failure of core cooling • Depressurization via PORV is potentially available • Containment air cooling, containment spray, and injection of BWST contents fail
TIN_18Y	3.2×10^{-9}	22%	• Transient with feedwater failure and early failure of core cooling • RCPs, containment air cooling, containment spray, PORV, and low pressure recirculation are available
SIY_49Y	1.4×10^{-9}	9.7%	• RCP seal LOCA with failure of injection but feedwater available • Ac power available • Containment air cooling, containment spray, and injection of BWST contents fail
SIY_51S	9.7×10^{-10}	6.7%	• RCP seal LOCA with failure of injection but feedwater available • Ac power available • Containment air cooling, containment spray, depressurization, and injection of BWST contents fail
SIY_43Y	9.0×10^{-10}	6.2%	• RCP seal LOCA with failure of injection but feedwater available • Ac power and containment air cooling available • Containment spray, depressurization, and injection of BWST contents fail
TIN_53Y	7.3×10^{-10}	5.0%	• Transient with failure of feedwater and early core cooling failure • Ac power not available
Other	2.9×10^{-9}	20%	
Total	1.4×10^{-8}	100%	

Plant-Damage State TIN-49S

Plant-damage state TIN_49S involves a transient initiating event with a failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. Containment-related systems including containment air cooling, containment spray, low pressure recirculation are not available, but the PORV is available for reactor coolant system depressurization. The frequency of accidents resulting in containment sidewall failure due to this plant-damage state is estimated to be 4.3 x 10^{-9}, which accounts for about 30% of the total frequency for this mode of containment failure.

The potential for sidewall failure is significant for this plant-damage state because it is possible for the reactor vessel to fail when the RCS pressure is relatively high, resulting in substantial dispersal of core debris to the lower compartment. For this plant-damage state containment systems that could provide cooling of the core debris are not available. Therefore, the core debris dispersed to the lower compartment might fail to form a coolable bed, and the containment vessel would be exposed to direct attack from molten debris.

Most of the plant-damage state TIN_49S frequency results from turbine building flooding caused by a large rupture in the circulating water system. Typical scenarios involve circulating water flooding that is not detected or isolated by automatic features. The resulting flooding causes a loss of all feedwater and a loss of one train of essential electrical power when the high voltage switchgear room is flooded. A failure in the other electrical train results in the loss of containment protective systems including containment spray, containment air cooling and low pressure injection.

Plant-Damage State TIN-18Y

Plant-damage state TIN_18Y involves a transient initiating event with a failure of decay heat removal via the steam generators and failure of makeup/HPI cooling (i.e., the same core-damage bin as that described above). Containment systems including containment air cooling, containment spray, low pressure recirculation, and the PORV are available. The containment sidewall failure frequency from this plant-damage state is estimated to be 3.2 x 10^{-9}, which accounts for about 22% of the total.

Sidewall failure frequency is significant for this plant-damage state because it is possible for the reactor vessel to fail when the reactor coolant pressure is high or intermediate resulting in substantial dispersal of core debris to the lower compartment. Although containment cooling is available, it may be possible for a coolable debris bed to fail to form, resulting in direct attack on the containment vessel by molten debris.

Plant-Damage State SIY-49Y

Plant-damage state SIY_49Y involves a transient initiating event that results in a RCP seal LOCA. Core damage occurs due to the failure of injection, but feedwater is available. Systems important to containment response, including containment air cooling, containment spray, and the injection of the BWST contents, fail. Containment sidewall failure due to this plant-damage state is estimated to have a frequency of 1.4 x 10^{-9}, which is about 10% of the total.

Initiating events that lead to a total loss of the service water system, including loss of service water or flooding in the intake structure, play an important role in this plant-damage state. Loss of service water leads to eventual loss of CCW, which is required for operation of the makeup and injection systems as well as for providing cooling for the RCP seals. Additionally, the loss of service water results in the failure of containment air cooling, containment spray, and low pressure recirculation.

A typical scenario would be a plant trip required by a total loss of service water, with failure of the operators to trip the RCPs. The potential for this sequence is due largely to the possibility of common-cause failure of the three service water pumps with the failure of recovery by the backup service water pump. Without operator action to trip the RCPs, a seal LOCA could result and the makeup and HPI pumps would not be available to provide injection.

4.1.3 Containment Bypass

Containment bypass was estimated to occur for about 2% of the total core-damage frequency. Containment bypass can occur due to interfacing-systems LOCAs, sequences initiated by a SGTR, or creep rupture of steam generator tubes for other accidents in which core damage was expected to progress at high pressure.

The potential for containment bypass is dominated by two plant-damage states that contribute about 85% of the total frequency. These plant-damage states include interfacing-system LOCAs and SGTR initiating events. Induced steam generator tube ruptures represent only about 5% of the containment bypass frequency, but represent a large fraction of the large early release frequency.

Table 11 summarizes the plant-damage states contributing significantly to containment bypass. Each is described briefly below.

Plant-Damage State V-07Y

Plant-damage state V_07Y involves a LOCA initiating event outside containment (i.e., an interfacing-systems LOCA). Failure to isolate the LOCA results in core damage after the contents of the BWST are injected. The containment bypass contribution from this plant-damage state is estimated to be 1.6×10^{-7} which accounts for about 52% of the total frequency.

Plant-Damage State RRY-07Y

Plant-damage state RRY_07Y involves a SGTR initiating event with successful injection but failure of long term cooling. The containment bypass contribution from this plant-damage state is estimated to be 1.1×10^{-7} which accounts for about 35% of the total frequency.

Plant-Damage State RIN-02Y

Plant-damage state RIN_02Y involves a SGTR initiating event with failure of injection and failure of feedwater. The containment bypass contribution from this plant-damage state is estimated to be 1.1×10^{-8} which accounts for about 4% of the total frequency.

Table 11. Plant-Damage States Contributing to Containment Bypass

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
V_07Y	1.6×10^{-7}	52%	• Interfacing-systems LOCA
RRY_07Y	1.1×10^{-7}	35%	• Steam generator tube rupture • Failure of long term cooling
RIN_02Y	1.1×10^{-8}	3.8%	• Steam generator tube rupture • Failure of injection • Feedwater not available
TIN_18Y	9.9×10^{-9}	3.5%	• Transient with feedwater failure and early failure of core cooling • RCPs, containment air cooling, containment spray, PORV, and low pressure recirculation are available
RRN_07Y	9.5×10^{-9}	3.1%	• Steam generator tube rupture • Failure of long term cooling • Feedwater not available
Other	8.0×10^{-9}	2.6%	
Total	3.0×10^{-7}	100%	

Plant-Damage State TIN-18Y

Plant-damage state TIN_18Y involves a transient initiating event with a failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. Containment systems including containment air cooling, containment spray, low pressure recirculation, and the PORV are available. The frequency of containment bypass due to this plant-damage state was estimated to be 9.9×10^{-9}, which is for about 3.5% of the total bypass frequency.

The containment bypass frequency for this plant-damage state results from the possibility of induced steam generator tube ruptures that could occur following a total loss of feedwater that could cause a loss of secondary side steam generator pressure. Additionally, the lack of feedwater to the steam generators results in an unscrubbed release.

A typical scenario found to contribute to the frequency of this plant-damage state would be the loss of main feedwater, followed by the failure of the two turbine-driven AFW pumps, with failure of the operators to start the motor-driven feed pump and to initiate makeup/HPI cooling.

4.1.4 Late Containment Failure

Late containment failure refers to the potential for overpressurization of the containment long (potentially tens of hours) after the failure of the reactor vessel. This could occur as a result of slow pressurization in the absence of containment heat removal, or the potential for late burning of hydrogen and other combustible gases. Late failure was estimated to occur for about 1% of the total core damage frequency.

Table 12 identifies the plant-damage state that contribute significantly to late containment failure. As the table indicates, two plant-damage states contribute more than 99% of the total frequency for this failure mode. These damage states are described in the discussion that follows.

Table 12. Plant-Damage States Contributing to Late Containment Failure

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
SIY_49Y	9.4×10^{-8}	71%	• RCP seal LOCA with failure of injection but feedwater available • Ac power is available • Containment air cooling, containment spray, and injection of BWST contents fail
TIN_53Y	3.7×10^{-8}	28%	• Transient with failure of feedwater and early failure of core cooling • Ac power is not available
Other	4.2×10^{-10}	0.3%	
Total	1.3×10^{-7}	100%	

Plant-Damage State SIY-49Y

As discussed above, plant-damage state SIY_49Y encompasses core-damage sequences initiated by a transient in which a RCP seal LOCA results. Plant damage occurs due to the failure of high pressure injection, but feedwater remains available. Containment systems including the containment air coolers, containment spray, and injection of the BWST contents fail. The frequency of this contribution to late containment failure was estimated to be 9.4×10^{-8}, which is about 71% of the total for this failure mode. Initiating events that lead to a total loss of the service water system, including failures within the service water system or flooding in the intake structure, play an important role in this plant-damage state. As noted earlier, this would lead to loss of CCW and consequential loss of RCP seal cooling and failure of HPI. The loss of service water also makes containment air cooling and containment spray unavailable, and prevents injection of the BWST contents by the low pressure injection system.

For this plant-damage state late containment is assured if containment cooling cannot be restored before the pressure in the containment vessel exceeds its capacity (unless the containment fails earlier for some other reason).

Plant-Damage State TIN-53Y

Plant-damage state TIN_53Y involves a transient initiating event that results in a loss of all ac power. Core damage occurs due to a failure of decay heat removal via the steam generators and a failure of makeup/HPI cooling. The frequency of late containment failure associated with this plant-damage state was estimated to be 3.7×10^{-8}, which accounts for about 28% of the total.

Eventual overpressurization of containment is effectively assured if power cannot be restored to allow for containment heat removal.

4.1.5 Basemat Meltthrough

Outcomes in which there was no other failure but the core debris was not cooled were assigned by default to the category of basemat meltthrough. This assignment may be somewhat conservative, since, for many accidents, it would be expected that the debris would freeze long before the basemat was penetrated. Depending on where the debris was located, ablation through the full depth of the concrete could take on the order of 100 hours.

Overall, about 2% of the core damage frequency was estimated to result in basemat meltthrough. As Table 13 indicates, the plant-damage states that contribute are primarily those in which the BWST contents would not be injected, since successful debris cooling is much more likely if the debris is deeply flooded. The damage states that contribute most to this category are described below.

Plant-Damage State SIY-49Y

Plant-damage state SIY_49Y encompasses sequences initiated by a transient, followed by failure of the RCP seals due to loss of seal cooling. Core damage occurs due to the failure of injection, but feedwater is available. Containment systems including the containment air coolers, containment spray and injection of the BWST fail. The frequency of basemat failure for this plant-damage state was estimated to be 8.4×10^{-8}, which accounts for about 33% of the total. As described for previous containment failure modes, this plant-damage state is dominated by scenarios that involve loss of all service water. This leads both to failure of seal cooling and of high pressure injection, as well as loss of containment heat removal and containment spray.

Plant-Damage State SIY-36Y

Plant-damage state SIY_36Y also involves transient initiating events that result in a RCP seal LOCA. Containment air cooling and containment spray remain available but low pressure recirculation fails. The frequency of basemat meltthrough due to this plant-damage state was estimated to be 3.7×10^{-8}, which accounts for about 15% of the total.

This plant-damage state is dominated by scenarios initiated by a loss of CCW. Component cooling water is required for operation of the makeup and injection systems as well as for providing cooling for the RCP seals.

Plant-Damage State TIN-53Y

As discussed above, plant-damage state TIN_53Y includes a transients with loss of all ac power. Core damage results because of loss of all feedwater and failure of makeup/HPI cooling. The frequency of basemat meltthrough from this plant-damage state was estimated to be 3.3×10^{-8}, which accounts for about 13% of the total.

Table 13. Plant-Damage States Contributing to Containment Basemat Failure

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
SIY_49Y	8.4×10^{-8}	33%	• RCP seal LOCA with failure of injection but feedwater available • Ac power is available • Containment air cooling, containment spray, and injection of BWST contents fail
SIY_36Y	3.7×10^{-8}	15%	• RCP seal LOCA with failure of injection but feedwater available • Ac power , containment air cooling, and containment spray are available • Injection of BWST contents fail
TIN_53Y	3.3×10^{-8}	13%	• Transient with failure of feedwater and early failure of core cooling • Ac power is not available
SIY_41Y	2.0×10^{-8}	8.1%	• RCP seal LOCA with failure of injection but feedwater available • Ac power and containment air cooling are available • Containment spray and injection of BWST contents fail
TIN_18Y	1.3×10^{-8}	5.1%	• Transient with failure of feedwater and early failure of core cooling • RCPs, containment air cooling, containment spray, PORV, and low pressure recirculation are available
MRX_02Y	1.3×10^{-8}	5.0%	• Medium LOCA with failure of core cooling in the recirculation phase • Containment air cooling and containment spray are available
Other	4.2×10^{-10}	0.3%	
Total	1.3×10^{-7}	100%	

4.1.6 Small Isolation Failure

Small containment isolation failures include containment leakage paths that do not cause appreciable containment depressurization. About 2% of the total core damage frequency was estimated to result in a small isolation failure. The plant-damage states that contribute the most to small isolation failure are primarily those that involve a failure of ac power. Loss of ac power could prevent motor-operated containment isolation valves from repositioning to their containment isolation positions.

4.1.7 Large Isolation Failure

Large containment isolation failures include containment leakage paths that are large enough to prevent long-term pressurization of the containment vessel. The fraction of core damage frequency with a large isolation failure was determined to be negligible. Less than 0.003% of the total core-damage frequency was estimated to result in large isolation failure.

4.2 Large Early Release Frequency (LERF)

A large early release is defined a severe accident that results in the rapid, unscrubbed release of airborne fission products to the environment before the effective implementation of offsite emergency response and protective actions. The frequency of these releases (LERF) provides a measure of both the potential for and severity of certain accidents that can be used in considering the acceptability of plant changes in risk-informed regulatory interactions.

It was assumed that the frequencies of all core-damage sequences that result in early containment failures, bypass failures, and containment sidewall failures would be included in LERF. It can be argued that sidewall failures evolve over a sufficiently long period of time that they should not be considered to be early releases. The frequency of accidents involving sidewall failure was, however, included in LERF to avoid arguments regarding whether the LERF was underestimated. The LERF results are discussed in the sections that follow, broken down in the following ways:

- By the types of events that are most important in initiating a large early release;
- By the types of plant-damage states that contribute most to LERF;
- By the types of human interactions that are the most important in LERF sequences;
- By the components whose failures are the most important with respect to LERF; and
- By the systems that are most important with respect to LERF.

4.2.1 Initiating Events Contributing to LERF

The chart in Figure 4 illustrates the relative contributions to LERF of the general categories of initiating events. As shown in this figure, transient initiators and steam generator tube ruptures each contribute about one third of the frequency of large early release. Interfacing-system LOCAs and internal floods are also significant contributors. LOCAs are less significant initiating events with respect to LERF.

Figure 5 provides a more detailed breakdown of contributions from specific types of initiating events. This figure indicates that steam generator tube ruptures are the largest contributor, but no single event dominates the total frequency.

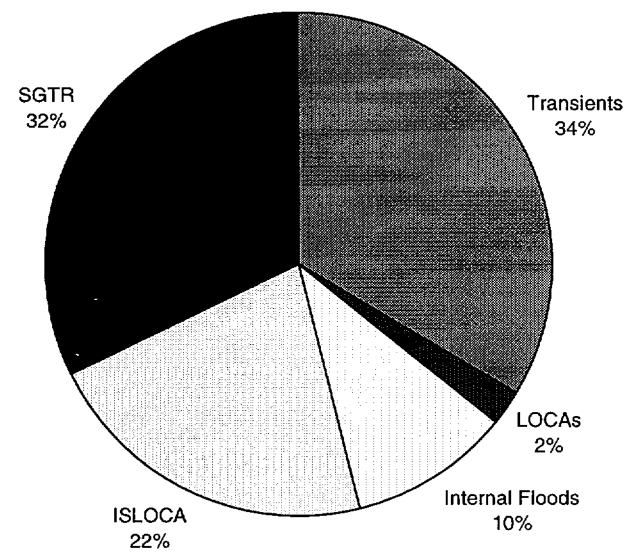

Figure 4. Contributions to Large Early Release Frequency from General Categories of Initiating Events

The contributions to large early release frequency from the various initiating events differ in several respects from the contributions to core-damage frequency:

- Steam generator tube ruptures and interfacing-systems LOCAs are relatively insignificant contributors to core-damage frequency but are the largest contributors to LERF. They are more important with respect to LERF because they lead to releases that bypass the containment.
- Component cooling water malfunctions are the largest contributors to core-damage frequency but do not contribute appreciably to LERF. Component cooling water is a support system for several systems that are important with respect to preventing core damage, but is not required for containment cooling systems. Therefore, additional failures of containment systems must generally occur for core damage to progress to an early, unscrubbed release.
- LOCAs were a large contributor to core-damage frequency but are an insignificant contributor to LERF. Most LOCAs depressurize the RCS, which reduces some challenges to containment integrity and reduces the potential for induced SGTRs.
- Turbine building flooding is a minor contributor to core-damage frequency but is one of the larger contributors to LERF. Turbine building flooding has a very low frequency but the consequences of major flooding can affect both systems needed to preserve core cooling, and features that protect against unscrubbed releases.

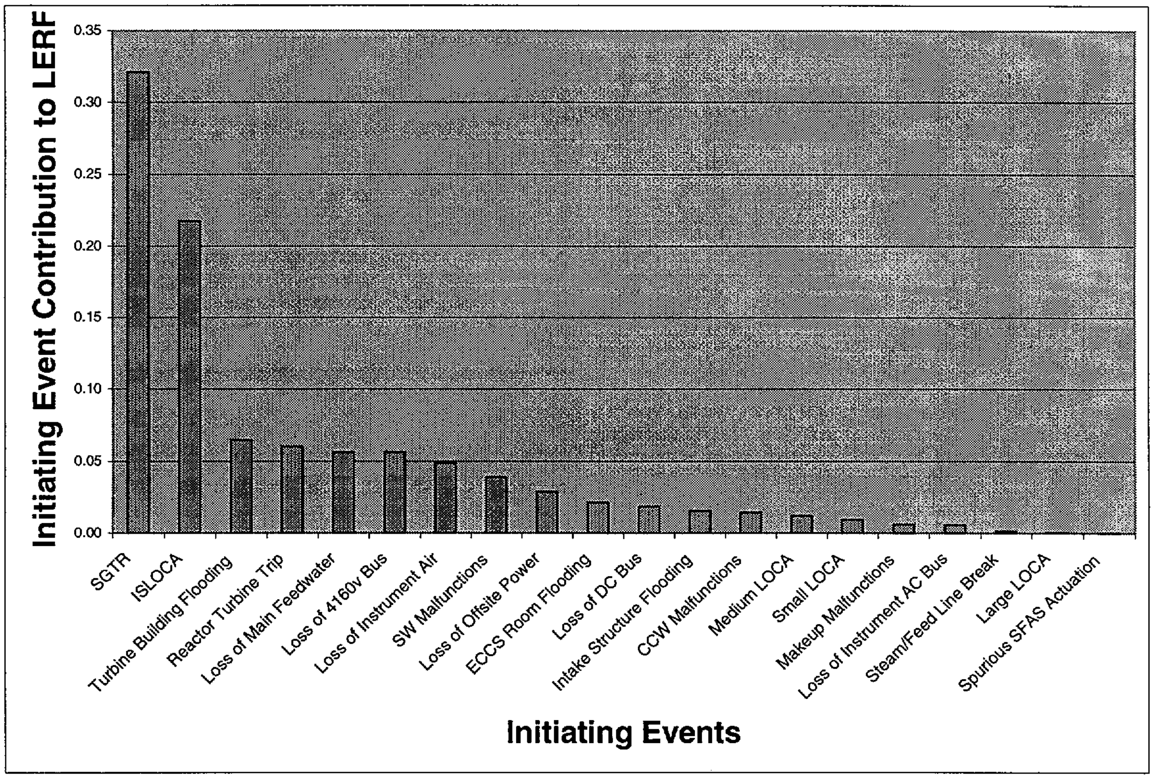

Figure 5. Contribution to Large Early Release Frequency by Initiating Event

- Service water malfunctions and flooding of service water areas contribute more to LERF than to core-damage frequency. Service water provides cooling for several systems that are important with respect to core-damage frequency and is required for containment cooling systems.

The following discussion describes some of the more significant initiators with respect to large early release frequency

Steam Generator Tube Ruptures

Accidents initiated by a steam generator tube rupture comprise less that two percent of the total core damage frequency but constitute almost one third of the large early release frequency. The disproportionate contribution of steam generator tube ruptures to LERF occurs because they result in a containment bypass. Scrubbing would be provided by feedwater to the steam generator but in the event of a feedwater failure an unscrubbed release would occur.

Interfacing-Systems LOCA

Interfacing-systems LOCAs are significant initiators with respect to LERF because they result in a containment bypass. Releases from an interfacing system LOCA would be expected to take place through the break directly into the auxiliary building. Scrubbing would be provided by submergence of the break location or by holdup and removal processes within the auxiliary building. It was assumed that a small fraction of the interfacing-systems LOCAs would not be submerged and no scrubbing was assumed for this fraction. Therefore, the large early release contributions result from the fraction of the interfacing-systems LOCA frequency that was assumed not to be scrubbed.

Turbine Building Flooding

Turbine building flooding constitutes a small contributor to core-damage frequency due to the low frequency of the initiating event and the automatic features that are designed to stop flooding after it is detected. For the cases in which flooding fails to be isolated before systems needed for core cooling are affected, however, the ability to maintain containment spray and other containment safety features may be compromised.

Reactor Trip

The reactor trip initiating event accounts for all plant trips in which none of the systems needed to maintain core cooling are directly affected. Although reactor trip is not a severe initiator, it is significant to LERF because of its high frequency relative to other events and because it can lead to a loss of feedwater. Failure of feedwater contributes to the probability that an induced SGTR will result in containment bypass.

Loss of Feedwater

Loss of feedwater is a significant event with respect to LERF due to the contribution of this event to induced steam generator tube ruptures. Failure of feedwater contributes to the probability that an induced steam generator tube rupture will result in containment bypass.

4.2.2 Plant-Damage States Contributing to LERF

The core-damage bins from the front-end analysis were split into more than 200 plant-damage states to ensure that possible effects relating to containment response were adequately addressed. These plant-damage states account for the failures of functions that would lead to core damage, some aspects of system response that could affect subsequent containment response, and the status of containment systems. Despite the large total number of plant-damage states, a relatively small number contribute a large percentage of the total LERF. The following are the general types of accidents that contribute most significantly to LERF.

- A steam generator tube rupture initiating event that leads to core damage due to failure of early or long term core cooling. If feedwater has failed the resulting release will be unscrubbed and contribute to LERF.

- Containment bypass due to an interfacing-systems LOCA. This containment bypass scenario may lead to an unscrubbed release, depending on the break location.

- A transient with total loss of feedwater and failure of makeup/HPI cooling. Because core damage would progress at high pressure, and because the steam generators would be dry, the potential exists for an induced, unscrubbed SGTR.

- A transient resulting in a loss of service water. Loss of service water could cause a RCP seal LOCA and the failure of injection systems, leading to core damage. Containment systems, including containment air cooling and containment spray, would also fail due to loss of service water, and an unscrubbed release could occur due to early or sidewall containment failure.
- A transient event resulting in a loss of all ac power. Core damage occurs due to a failure of decay heat removal via the steam generators and a failure of makeup/HPI cooling. Containment systems fail due to the loss of ac power and an unscrubbed release occurs due to an induced SGTR, early containment failure, or containment sidewall failure.

One category of accident that was a significant contributor to CDF was a determined to be a small contributor to LERF:

- Transients resulting in a loss of CCW constitute the largest contributor to core-damage frequency but a small contribution to LERF. This is primarily because containment air cooling and containment spray are not dependent on CCW.

Each of the plant-damage states that contribute significantly to LERF is described briefly below. Table 14 summarizes the significant plant-damage state contributors to LERF.

Plant-Damage State V-07Y

Plant-damage state V_07Y involves an interfacing system LOCA initiating event that results in containment bypass. Failure to isolate the LOCA results in core damage after the contents of the BWST are injected. The LERF contribution from this plant-damage state is estimated to be 1.6E-8 which accounts for about 22% of the total LERF.

Releases from an interfacing system LOCA would be expected to take place through the RCS break directly to the auxiliary building. Scrubbing would be provided by submergence of the break location or by holdup and removal processes within the auxiliary building. It was assumed that a fraction of the interfacing systems LOCA would not be submerged and no scrubbing was assumed for this fraction. Therefore, the large early release for this plant-damage state results from the fraction of interfacing system LOCA that are assumed to not to be submerged.

Plant-Damage State RIN-02Y

Plant damage state RIN_02Y involves a steam generator tube rupture initiating event with failure of early core cooling. For this plant damage state, feedwater to the steam generator has failed, resulting in an unscrubbed release. The LERF contribution from this plant-damage state is estimated to be 1.1×10^{-8} which accounts for about 14% of the total LERF.

Table 14. Plant-Damage States Contributing to LERF

Plant-Damage State	Annual Frequency	Contribution for Category	Plant-Damage State Description
V_07Y	1.6×10^{-8}	22%	• Interfacing-systems LOCA (containment bypass)
RIN_02Y	1.1×10^{-8}	14%	• Steam generator tube rupture • Failure of early core cooling • Failure of feedwater
TIN_18Y	9.9×10^{-9}	14%	• Transient with failure of feedwater and early failure of core cooling • RCPs, containment air cooling, PORV, and low pressure recirculation are available
RRN_07Y	9.5×10^{-9}	13%	• Steam generator tube rupture • Failure of long term cooling • Failure of feedwater
TIN_49S	4.4×10^{-9}	6.1%	• Transient with failure of feedwater and early failure of core cooling • Depressurization succeeds • Containment air cooling, containment spray, and injection of BWST contents fail
TIN_53Y	3.4×10^{-9}	4.7%	• Transient with failure of feedwater and early failure of core cooling • Ac power not available
SIY_49Y	3.3×10^{-9}	4.5%	• RCP seal LOCA with failure of injection but feedwater available • Ac power available • Containment air cooling, containment spray, and injection of BWST contents fail
SRY_39Y	2.6×10^{-9}	3.5%	• RCP seal LOCA with failure of injection but feedwater available • Containment air cooling available • Containment spray, and injection of BWST contents fail
Other	1.3×10^{-8}	18.2%	
Total	7.3×10^{-8}	100%	

Plant-Damage State TIN-18Y

Plant-damage state TIN_18Y involves a transient initiating event with a failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. Containment systems including containment air cooling, containment spray, low pressure recirculation and the PORV are available. The LERF contribution from this plant-damage state is estimated to be 9.9×10^{-9}

which accounts for about 14% of the total LERF. The LERF for this plant-damage state results from the possibility of induced SGTRs following a total loss of feedwater.

Plant-Damage State RRN-02Y

Plant damage state RIN_02Y involves a steam generator tube rupture initiating event with failure of long term core cooling. For this plant damage state, feedwater to the steam generator has failed, resulting in an unscrubbed release. The LERF contribution from this plant-damage state is estimated to be 9.5×10^{-9} which accounts for about 13% of the total LERF

Plant-Damage State TIN-49S

Plant-damage state TIN_49S involves a transient initiating event with a failure of decay heat removal via the steam generators and failure of makeup/HPI cooling. Containment systems including containment air cooling, containment spray, and low pressure recirculation are not available, but the PORV is available for RCS depressurization. The LERF associated with this plant-damage state was estimated to be 4.4×10^{-9}, which accounts for about 6% of the total LERF.

This contribution to LERF results primarily from the potential for containment sidewall failure. Sidewall failure would occur if the reactor vessel were to fail when reactor coolant pressure was relatively high. This could lead to substantial dispersal of core debris to the lower compartment. For this plant-damage state containment systems that could provide cooling of the core debris would not be available; therefore, the containment vessel would be exposed to direct attack from molten debris.

As indicated earlier, most of the frequency for plant-damage state TIN_49S results from turbine building flooding caused by a large rupture in the circulating water system. Typical scenarios involve circulating water flooding that is not detected or isolated by automatic features. The resulting flooding causes a loss of all feedwater and a loss of one train of essential electrical power when the high voltage switchgear room is flooded. A failure in the other electrical train results in the failure of containment protective systems including containment spray, containment air cooling, and low pressure injection.

Plant-Damage State SIY-49Y

Plant-damage state SIY_49Y involves a transient initiating event that results in a RCP seal LOCA. Plant damage occurs due to the failure of injection but feedwater is available. Containment systems including containment air cooling, containment spray, and injection of the BWST contents fail. The LERF contribution from this plant-damage state was estimated to be 3.3×10^{-9}, which is about 4.5% of the total LERF.

For this plant-damage state large early release frequency is divided between sidewall failure and early containment failure. These failures occur due to the high pressure associated with this plant-damage state and the loss of containment systems. Induced tube ruptures do not occur due to the availability of feedwater.

Initiating events that lead to a total loss of the service water system, including failures of service water equipment or flooding in the intake structure, play an important role for this plant-damage state. The loss of service water leads to loss of CCW, contributing to RCP seal failure. Additionally, the loss of service water results in the failure of containment systems including containment air cooling, containment spray, and low pressure recirculation.

Plant-Damage State TIN-53Y

Plant-damage state TIN_53Y involves a transient initiating event that results in a loss of all ac power. Core damage occurs due to a failure of decay heat removal via the steam generators and a failure of makeup/HPI cooling. The LERF contribution from this plant-damage state is estimated to be 3.4×10^{-8}, which accounts for about 4.7% of the total LERF.

The potential for a large early release for this plant-damage state is divided between induced tube ruptures, containment sidewall failure, and early containment failure. Induced tube ruptures contribute to LERF since feedwater is not available to limit heating in the tubes and for fission produce scrubbing. Sidewall and early failure also contribute to LERF because the loss of ac power means that fission-product scrubbing by containment spray is not available.

4.2.3 Operator Actions of Significance to LERF

Approximately 150 human actions and 250 combinations of human actions were modeled for the Davis-Besse PSA. Of these human actions, about a dozen were determined to be risk-significant with respect to LERF. There was a general correlation between the actions determined to be risk significant in the front-end and back-end portions of the PSA analysis. Some differences were, however, noted.

- Actions associated with the isolation of interfacing-system LOCAs were not significant in the front-end (core-damage frequency) part of the analysis, but are significant with respect to LERF because they lead directly to a bypass of containment.
- Actions associated with recovering CCW or isolating flooding in the CCW pump room were less significant in the LERF analysis because containment cooling systems are not dependent on CCW.
- Actions associated with recovering service water or isolating flooding in the service water pump room were more significant in the LERF analysis because the containment cooling systems are dependent on service water.
- Actions associated with LOCA response, including initiating low pressure recirculation, are important in the front-end analysis but are not risk significant with respect to LERF. This is a reflection of the relatively small potential for LOCA-induced accidents to lead to a large early release.

The events that were assessed to be risk significant and the reasons they were assessed to be risk significant are summarized below. Since most of these actions were discussed in section 3.1.3, only a brief description is provided here.

Establishing Makeup/HPI Cooling

This event represents failure of the operators actions to establish makeup/HPI cooling in the event that heat removal is not available via the steam generators. This action is significant in the back-end analysis because the resulting core-damage sequences would be likely to progress at high pressure prior to failure of the reactor vessel. This would present the challenges associated with a high pressure melt ejection, as well as the possibility of induced SGTRs.

Tripping Reactor Coolant Pumps Following Loss of Seal Cooling

In the event that all RCP seal cooling is lost, the operators must trip the RCPs to prevent failure of their seals and a consequential small LOCA. Loss of seal cooling involves loss of both cooling of the thermal barriers by the CCW system and seal injection from the makeup system. It is estimated that if the operators do not respond to trip the RCPs, a seal LOCA would result within approximately 30 minutes.

This human action is important in scenarios involving a total loss of service water. If both trains of service water were lost either due to equipment failures or due to flooding, a loss of seal cooling to the RCPs would result. The loss of service water would also cause the loss of cooling to the pumps that might provide injection to the RCS, so that they would be unavailable to respond to a seal LOCA. Additionally, the service water failure would cause the loss of cooling to containment air cooling and ECCS room cooling, which is required for operation of the containment spray pumps.

Initiating Flow from the Motor Driven Feed Pump

Following a loss of main feedwater, the turbine-driven AFW pumps would be actuated automatically. It they should fail to start or deliver flow, the motor-driven feed pump could be manually started to serve as a backup. This action is significant in the backend analysis because loss of feedwater to the steam generators could lead to core damage at high pressure. This leads to greater potential for failure of the containment due to high pressure melt ejection, and has he potential to result in an induced SGTR.

Isolating an Interfacing Systems LOCA

Failure to find and isolate an interfacing-systems LOCA is important in the backend analysis because it leads to core damage following a break outside containment. Such breaks are more likely than most other sequences to contribute to LERF.

Manually Controlling AFW Pump Speed

If dc power were lost to an AFW flow control valve (AF6451 or AF6452), the valve would fail fully open. Without operator intervention, the affected steam generator could overfill, and carryover of water into the steam lines could affect continued operation of both turbine-driven AFW pumps (since their steam supplies would be cross-connected).

This event could occur in scenarios involving a dc bus fault or a loss of power from 4160v bus C1 or D1 bus followed by battery depletion. The scenario is significant because it could contribute significantly to loss of all feedwater, when power would be unavailable to containment safety features.

Adding Fuel for the Station Blackout Diesel-Generator

In the event of an extended loss of offsite power it could be necessary to refuel the station blackout diesel-generator if one or both of the emergency diesel-generators were to fail. Failure to do so could contribute to station blackout scenarios that could both contribute to core damage and loss of containment cooling.

Recovering Service Water

If all three of the service water pumps were lost, either due to common-cause failure or internal flooding, the dilution pump might provide an option for restoring service-water flow. Failure to restore service water flow would eventually result in the loss of CCW, which could lead to a seal LOCA with no high pressure injection capability.

Using the dilution pump to restore service-water flow was evaluated in the PSA to be more risk significant than restoration using the spare service water pump. This is because the dilution pump is in a different location and is less susceptible to common-cause failures that might be shared with the other service water pumps.

Establishing Feedwater Flow from the Startup Feed Pump

In the event of a total loss of feedwater, including the failure of both turbine-driven pumps and the motor-driven feed pump, the operators could use the startup feedwater pump as an additional backup means of supplying flow to the steam generators.

Opening Valve SW82 Following Maintenance

Valve SW82 is the manual isolation valve in the service water return line for the ECCS room coolers. It is in a portion of the return line that is common to all five ECCS room coolers. If it were left closed after some maintenance activity, the lack of flow through the coolers would constitute a potential source of common-cause failure.

This action differs from the others discussed in this section in that it relates to a human interaction prior to a plant trip (i.e., the failure would leave equipment unavailable when it was needed, rather than involving a lack of proper response).

4.2.4 Importance of Individual Components to LERF

Section 3.1.4 presented the risk achievement worth (RAW) of individual components. The RAW values were calculated by determining the ratio of the core-damage frequency with the component failure probability set to 1 to the nominal core-damage frequency. Risk achievement worth can also be calculated for LERF. In this case the RAW for a component is the ratio of the LERF with the component failure probability set to 1 to the nominal LERF. The RAW calculated based on LERF is a useful measure for determining the effects on release frequency associated with removing equipment from service for maintenance. Table 15 summarizes the RAW values based on LERF for a variety of components.

Table 15. Effect on LERF with Significant Components Out of Service

Effect on Nominal LERF	Component Out of Service
>25 x	• 480v bus F1 • 4160v bus D1 • Station battery
10 to 25 x	• 4160v bus C1 • 480v bus E1
5 to 10 x	• Motor-driven feed pump
2 to 5 x	• Auxiliary feed pump • Backup service water pump (dilution pump)
1.5 to 2 x	• Low pressure injection pump • Emergency instrument air compressor • Station blackout diesel generator • PORV • Emergency diesel generator
1 to 1.5 x	• Startup feedwater pump • Makeup pump • Service air compressor • High pressure injection pump • Standby component cooling water pump • Containment spray pump
1 x	• Containment air cooler • Spare component cooling water pump • Spare service water pump

The component importance can also be characterized by the actual LERF contribution due to a particular failure. This measure is referred to as the Fussell-Vesely importance and is equivalent to the percentage by which the LERF could be reduced if the component could be made 100% reliable. Figure 6 illustrates the Fussell-Vesely importance for a selection of significant components. This ranking is valuable because it can be used as a guide in determining where resources could be applied most effectively in reducing LERF.

Some of the insights gained from considering these importance measures are adderssed in the discussion that follows.

Electrical Power Buses

The components with the highest RAW values are electrical buses. These components rank high according to RAW because the loss of a bus can affect a number of components that may be required to prevent both core damage and a release. Despite the high RAW the electrical buses are small contributors to the nominal core damage frequency because electrical buses are generally very reliable components.

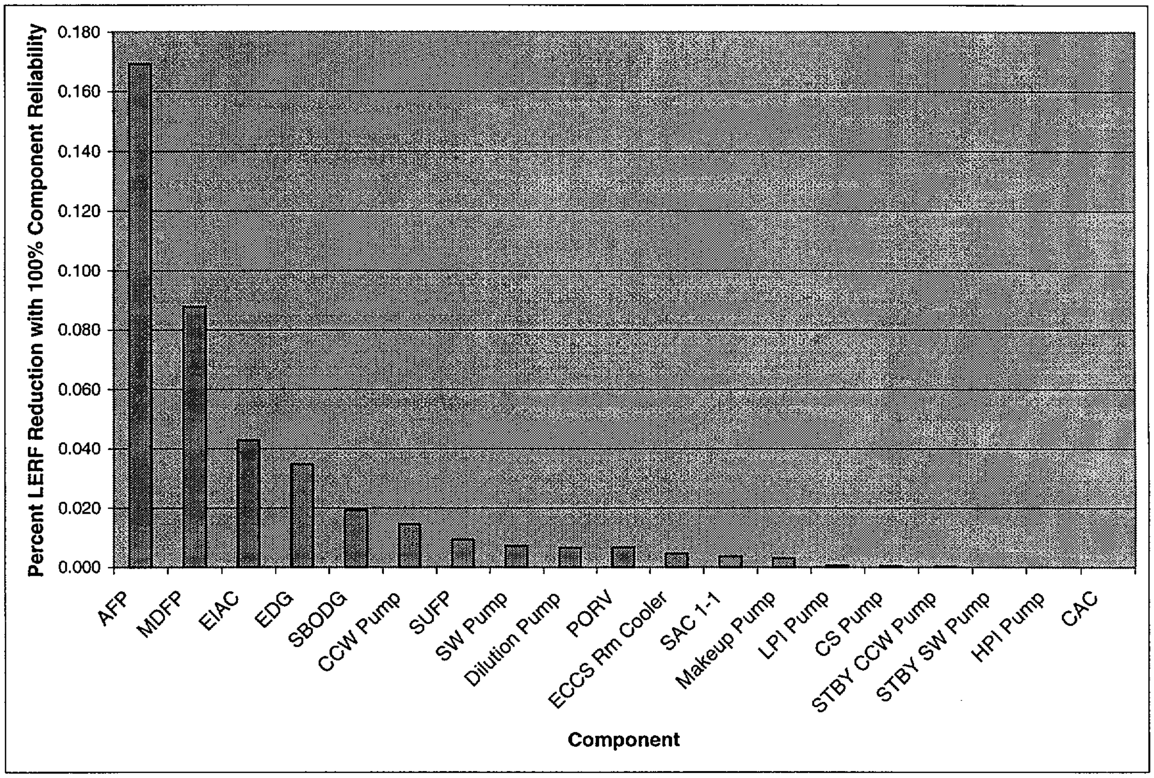

Figure 6. Components Failure Contributions to Large Early Release Frequency

Auxiliary Feedwater Pumps

The auxiliary feedwater pumps and the motor-driven feed pump have the highest RAW values other than those of electrical buses. The high RAW values of these components reflects their importance both in preventing core damage by providing core cooling and in preventing conditions from occurring that could lead to an induced SGTR. Additionally, feedwater provides fission product scrubbing in the event of a steam generator tube rupture.

Emergency Diesel Generator

The emergency diesel generators have a relatively high level of importance in both measures relative to LERF. This is contrary to the results based on core-damage frequency, which show the emergency diesel generators not to be especially high in either measure. The greater importance with respect to release frequency reflects the significant contribution of station blackout sequences to release frequency.

Backup Service Water Pump (Dilution Pump)

The backup service water pump has a higher RAW based on LERF than the RAW calculated based on core-damage frequency. This reflects the increased importance of service water in the level 2 analysis. Service water is especially significant with respect to release frequency because the both the containment air coolers and the containment spray pumps are dependent on service water cooling. Therefore, loss of service water can lead to core damage and degrade the containment cooling systems.

Containment Air Cooling and Containment Spray

Containment air cooling and containment spray are insignificant in both importance measures. This result may seem contrary to expectations but is actually a reflection of the amount of redundancy in containment cooling systems. The probability that containment cooling will fail due to a series of independent failures is quite low. Therefore, support systems, including cooling water and electrical distribution are more important than individual containment system components.

4.2.5 System Importance Measures for LERF

System importance was calculated using methods similar to those used for individual components. Since entire systems are not normally removed from service, the system importance for this report was evaluated by calculating the percent reduction in LERF that would be achieved by 100% reliable system performance. As with individual components, this method is useful because it can be used as a guide in determining where resources can be applied in improving plant safety. The Fussell-Vesely ranking by system are illustrated in Figure 7.

The results of this assessment highlight the importance of feedwater systems and electrical or cooling water support systems. Feedwater systems are very important because a large fraction of the core damage frequency involves sequences with failures of feedwater to the steam generators that lead to core damage due to loss of cooling. Loss of cooling sequences are significant contributors to LERF because core damage may occur at high reactor coolant pressure which increases the probability of containment failure at vessel breach. Also the probability of induced SGTR is greatest for loss of cooling sequences with high reactor coolant pressure and no feedwater flow to the steam generators.

Support systems are important because failures in these systems can result in failures of multiple components that are needed to prevent core damage and a radionuclide release. The probability that individual failures of multiple trains and systems will lead to a large release is much smaller than the probability that the release will be caused by the failure of a support systems. Therefore, support systems including the emergency diesel generators, 4160v distribution, service water, instrument air, and 13.8 kv distribution, are more important than front-line systems including high pressure injection, low pressure injection and containment spray.

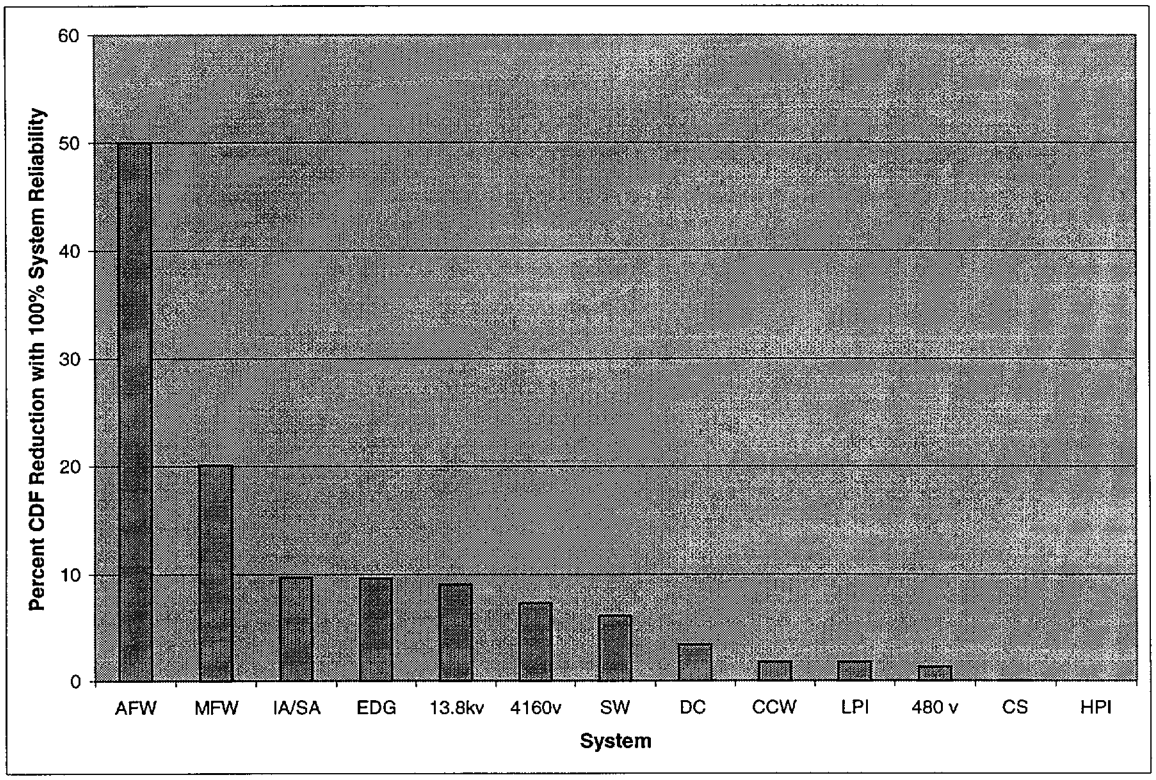

Figure 7. System Contributions to Core-Damage Frequency

4.3 Results for Release Categories

As described in Section 2, the large number of outcomes represented by combinations of plant-damage states and containment failure modes were sorted into 34 release categories. These release categories are intended to capture the features that could be important with respect to the calculation of offsite consequences. They account for the timing, energy, and fractions of radionuclide inventories released.

The frequency for each release category was calculated based on the frequencies of the plant-damage states and conditional probabilities of the relevant containment outcome. These frequencies can be broken down to understand the important contributors, as has been done in the preceding sections. These results, however, correspond to the results for the various containment failure modes discussed in Section 4.1. It would be needlessly repetitive to discuss these results again relative to the release categories.

Section 5
Conclusions and Applications of the PSA

5.1 Conclusions

The overall core-damage frequency is lower than that assessed for the IPE for reasons that are largely reflective of improvements that have been achieved in the performance of Davis-Besse. The most important of these are the reductions in the frequency of plant trips, and in particular in the frequency of the loss of main feedwater. Improved component reliability, especially for motor-operated valves, also contributed to this reduction.

The objectives for the update seem to have been achieved. The PSA models and associated data bases more accurately reflect the current plant configuration and recent plant experience. They can be used much more effectively and efficiently to perform a variety of applications to address plant safety and operational issues. The PSA staff is able to modify and exercise the PSA to accomplish these applications. The updating of the back-end analysis was performed in a way that will greatly facilitate the calculation of the large early release frequency. Some of the applications that have been made and that may be made in the future are described in the next section.

5.2 Risk-Informed Applications

The PSA constitutes a tool that can be, and indeed has been, used in a wide variety of applications. These applications have the potential to become increasingly common and important as the NRC moves toward risk-informed regulations [Ref. 13]. The general types of applications that can be made are outlined in an earlier document, and include the following [Ref. 14]:

- Systematic examination of plant safety,
- Evaluation of regulatory issues,
- Assessment of the justification for and optimal approach to modifications,
- Guidance for procedure development and training,
- Support for accident management,
- Protection of plant investment, and
- Support of plant life extension.

Several useful applications of the PSA have been made or are underway. These applications are summarized in the following discussion.

Support for the Maintenance Rule and Maintenance Planning

The Maintenance Rule represents one of the first major attempts by the NRC to incorporate risk-based insights into regulations for nuclear power plants. The PSA for Davis-

Besse provided several types of information for the development of the plant's Maintenance Rule program. These included the following:

- The risk results from the PSA performed for the IPE served as the primary means for determining the risk significance of systems. This guided the selection of systems and functions to be covered in the Maintenance Rule program.
- The data and results from the PSA were used to define criteria for equipment performance. These criteria are used to assure that equipment unavailability tracks with expected performance, and to identify expected rates of failure to monitor equipment reliability.
- A series of sensitivity studies was performed to identify combinations of equipment that, if removed from service for maintenance at the same time, would result in a temporarily higher relative level of risk. A matrix was formed that indicated which combinations of equipment outages should be avoided. Using this matrix in planning maintenance activities allows the risk of operations to be reduced without significant costs [Ref. 15].

The support for Maintenance Rule activities continues, and the matrix of equipment unavailabilities has been expanded and updated to reflect the most recent risk results. This effort has evolved further into providing ongoing support to maintenance planning. This is accomplished through the development of a Weekly Risk Profile. This profile examines the maintenance activities planned for the coming week, and examines the time-dependent risk implied by the expected equipment outages. This process yields much more detailed information regarding the implications of equipment outages to augment the general cases covered in the matrix. The objective of this effort is to ensure that risk implications are adequately understood and, where feasible, to suggest changes to maintenance plans that will reduce risk without incurring undue costs or complications.

Extended Allowed Outage Times

The allowed outage times (AOTs) in most technical specifications were based on engineering judgment, without a clear or objective technical basis. One of the most useful applications of risk informed analysis could be to provide a justification for extending the specific AOTs. Longer AOTs can afford further flexibility for plant operations and maintenance, without a significant impact on the risk of core damage.

Such an analysis has been completed recently with respect to extending AOTs for the decay heat removal and containment spray systems. Plant-specific analyses of the impact on risk of extending the AOT to one week for a train in each system were completed for each of the Babcock & Wilcox (B&W) plants operating in the U.S., and compiled into a single report on behalf of the B&W Owners Group [Ref. 16]. The analyses were prepared in conformance with recently developed guidance from the NRC [Ref. 17]. The analyses are currently being reviewed by the NRC. Following its review of the technical bases, the technical specifications themselves may be changed to reflect the longer AOTs.

Risk Prioritization

Relative risk measures can be particularly useful for establishing priorities for considering measures to address potential concerns. Among such applications that are underway for Davis-Besse is the use of risk ranking to determine which of the air-operated valves (AOVs) in safety-related applications at Davis-Besse are most important with respect to the potential for core damage. This ranking will be used to focus inservice-testing (IST) programs for AOVs, applying criteria developed by the industry and NRC [Refs. 18—20]. A similar effort is envisioned for motor-operated valves.

Assessment of Potential Plant Modifications

Because PSA encompasses a systematic approach to examining plant safety, it can sometimes offer unique perspectives on potential safety issues. It can also be used effectively to determine whether modifications to address such issues are merited, and to identify the most effective design option to reduce or eliminate the impact of the issues.

A relatively recent application of this type related to the potential for flooding in the turbine building due to a break in the condenser circulating water system. In the IPE, the assumption was made that flooding from the condenser water system would be self-limiting because the circulating water pumps would be flooded and would fail before the flood level could rise above the basement of the turbine building. Subsequent review determined that this assumption was not correct. A detailed assessment was made of the potential for flooding from the system, of the effects such flooding could have on equipment needed for maintaining core cooling, and of the frequency of core damage that could be attributed to such floods [Ref. 21].

As a result of that assessment, it was concluded that steps should be taken to reduce the risk associated with this flooding. The assessment was extended to consider the relative merits of a variety of proposed modifications. These modifications were primarily aimed at providing early indication to the control room operators of the existence of flooding, and for more effective automatic termination of the flooding. The results of this assessment were used to identify the modification that would be the most effective in reducing risk, balanced against concerns that the automatic functions might cause inadvertent losses of circulating water and consequential plant trips.

Status for Future Applications

As the PSA is now configured, it should be possible to make more effective and more meaningful applications in the future. In addition to being a more realistic reflection of the plant as it currently exists, the PSA has been refined so that it can be used to examine a broader range of issues. Together with these refinements, the use of newer tools to track and exercise the models will greatly facilitate these applications.

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Appendix A
Glossary

AND gate. A logical operator used in constructing a fault tree. An AND gate signifies an event that occurs if and only if all of the events or conditions that are inputs to it occur.

back-end analysis. The portion of a PSA that concerns itself with the investigation of the physical progression of the accident, and especially the period after melting of the core begins. The primary objective of the back-end analysis is to determine, for the challenges presented by each core-damage sequence (or group of similar sequences), the likelihood of various containment responses and failure modes (e.g., whether the containment retains its integrity throughout the accident, fails at some point due to overpressurization, experiences meltthrough of its basemat, etc.). The back-end analysis may also include an evaluation of the severity of the radionuclide releases associated with each accident. The back-end analysis is typically comprised of a series of deterministic thermal-hydraulic calculations to predict the accident progression; the construction of a containment event tree to depict the possible pathways for containment response; estimation of the probabilities of the events that make up the branch points for the containment event tree; and quantification of the containment event tree paths to determine the relative probabilities of the accident outcomes. The scope of a "level 2 (or II)" PSA includes both the front-end and back-end analyses.

basic event. The event at the lowest level to which the modeling is carried in a fault tree. A basic event is usually a hardware failure or human interaction to which a probability can be assigned directly.

Bayesian updating. In PSA, a process in which a preliminary (or prior) estimate of a failure rate or other parameter of interest is combined (updated) with plant-specific evidence (e.g., a certain number of failures in a defined number of operating hours). The purpose of this updating process is usually to modify an estimate of the parameter based on generic experience to account for plant-specific experience when the plant-specific experience is not sufficient to be statistically meaningful on its own.

common-cause failure. A single cause of failure that affects (or has the potential to affect) more than one (usually identical) component. An example of a common-cause failure would be the simultaneous failure of a set of pressure transmitters to respond as intended because all of them had been miscalibrated using the same faulty calibration equipment.

containment bypass. An accident that has the potential simultaneously to lead to core damage and to cause a release of radionuclides outside the containment. An interfacing-systems LOCA is the most common example of such an event.

core-damage frequency. The rate at which accidents that could cause core damage would occur, nearly always expressed on a per-year basis. For most light water power reactors, the core-damage frequency is typically in a range of once per one hundred thousand to once per ten thousand years.

core-damage sequence. A combination of events, including an initiating event and subsequent failures, that would lead to core damage. The sequence may be defined at different levels. For example, the event trees used in the Davis-Besse PSA define core-damage sequences at the level of a general category of initiating event and the failures of various safety functions.

cut set. A set of basic events (system or component faults, human interactions, etc.) that would cause failure of the top event in a fault tree. For example, a cut set for failure of the high pressure injection (HPI) system at Davis-Besse would be failure of one of the HPI pumps to start at a time when the other pump was in maintenance. This is also a minimal cut set, because both failure states would have to exist simultaneously for the system to fail.

dependency. A feature of two or more elements of the risk models or results that makes the occurrence of the multiple elements more likely than would be implied by the combination of the probabilities for the individual elements. A common-cause failure is one example of a dependency. Multiple human interactions are usually considered to be subject to dependency as well, since the same persons may be responsible for deciding on and implementing the implied actions. Dependency also arises through the reliance of different systems or components on the same support (such as the same electrical bus or same train of cooling water). One of the most critical aspects of any PSA is the need to ensure that dependencies are properly identified and characterized, since the risk measures could otherwise be significantly underestimated.

event tree. A graphical modeling tool used in PSAs to delineate the various types of accidents that could result from an initiating event. In the PSA for Davis-Besse, a set of event trees was used to define the core-damage accidents that might occur. A separate event tree was also used to describe the possible ways in which the physical progression and containment response for an accident could take place after the onset of core damage.

external event. An initiating event that results from some condition, usually a change in environment. External events may be either naturally occurring or manmade. Examples include earthquakes, tornados, hurricanes, external floods (i.e., flooding of the plant site from a nearby lake or river, etc.), explosions or releases of toxic gases from nearby industrial sites, and aircraft impact. The definition is often fuzzy, and depends largely on historical usage. For example, damage due to turbine missiles and fires originating inside the plant are usually defined to be external events, but loss of offsite power due to loss of a power grid is commonly treated as an internal event. Internal flooding may be an internal or an external event, depending on the preferences of those responsible for the PSA. One reason for making the distinction between internal and external events is that the methods for analyzing external events are often not as well developed as those for internal events, and there may not be as much applicable data (for example, there is little historical experience to support estimating the frequencies of earthquakes large enough to threaten nuclear power plants). Hence, the results of external events assessments are usually more uncertain than those for internal events.

fault tree. A graphical modeling technique that allows a high-level event (such as failure of a particular system to function) to be developed down to the level of component or sub-component faults through the use of Boolean logic operators (such as AND and OR gates). A fault tree is typically evaluated using a computer program that performs Boolean reduction to determine cut sets for the top event and to calculate the frequencies of the cut sets based on the input reliability data.

fault-tree linking. A process by which the frequency of an accident sequence (or set of sequences) is calculated by combining the fault trees for individual systems or safety functions represented as top events in an event tree into a "master" fault tree. This linked fault tree, is then evaluated to determine sequence cut sets, which form the basis for quantifying the frequency of the sequence(s).

fragility. The probability of failure of a structure, system, or component as function of the level of severity of some hazard or stress to which it is exposed. The most common use of fragility in PSA is for earthquakes; in this case, fragility is expressed as probability of failure as a function of earthquake magnitude.

front-end analysis. The portion of a PSA in which the potential core-damage accidents are identified and their frequencies are estimated. This process entails identifying the initiating events of interest; defining the general sequences of events that could lead to core damage for those initiators (usually through the construction of event trees); relating these general event sequences to specific failures and human interactions at the system and component levels (usually by constructing fault trees); evaluating the frequencies of the initiating events and the probabilities of the system- and component-level events from operating experience (from Davis-Besse if it is available, or from the nuclear industry in general if it is not); and integrating these elements to identify the combinations of initiating events and basic failures to define the ways in which the accidents could occur and to permit their frequencies to be estimated. Also referred to as the "level 1 (or I)" analysis.

Fussell-Vesely importance. A measure of the contribution of a basic event or group of events to the probability of the top event in a fault tree. It is calculated by summing the probabilities of all cut sets that contain the event or events of interest, and dividing by the total probability for the top event. Another way to look at Fussell-Vesely importance is as the fraction of the total probability of failure for the top event that would be eliminated if the event of interest could be made perfect (i.e., with a zero probability of failure).

generic data. Component failure rates, initiating event frequencies, and other data that is based on the combined experience of other operating units, including, in some cases, non-nuclear industrial facilities.

hazard. A condition or event that has the potential to cause damage to plant structures, systems, or components. Usually, the hazard can be defined in quantitative terms as a frequency of occurrence or as a function of magnitude or intensity (e.g., frequency of lake flooding as a function of height of flood, or frequency of earthquakes as a function of effective ground acceleration).

human interaction. An event in a logic model or cut set relating to a human action. To be consistent with the conventions of fault trees, the event usually represents a failure (e.g., the failure to perform a required action, or the performance of an incorrect action). This term has generally replaced the term "human error" in PSAs, to reflect the fact that many failures associated with human interactions stem from deficiencies in procedures or other aspects of the man-machine interface, rather than from actual mistakes made by personnel.

human reliability analysis. The identification of events (i.e., human interactions) that represent the role of operators, maintenance personnel, or others in the logic models (that is, the event trees and fault trees) that define the accident sequences, and the estimation of the probabilities for those events through characterization of the conditions that influence their success or failure.

importance analysis. A calculation that yields any of several formal measures to characterize the relative significance of factors that contribute to the results in reliability and risk assessments. Three of the most common measures in importance analyses are Fussell-Vesely importance, risk achievement worth, and risk reduction worth.

individual plant examination (IPE). An evaluation requested of every operating nuclear power plant by the NRC in Generic Letter 88-20 to investigate the potential for severe accidents to occur. Although other approaches were permitted, most utilities (including Toledo Edison) performed a PSA to satisfy the generic letter. In referring to the study completed for Davis-Besse, the terms IPE and PSA are often used interchangeably. The scope of the IPE for Davis-Besse includes a full range of internal initiating events, including internal flooding. Only accidents initiated during nominal full-power operation were considered.

individual plant examination for external events (IPEEE). This is an extension of the scope of the IPE to cover external events, and requested by the NRC in a supplement to Generic Letter 88-20. Included in this expanded scope are earthquakes, tornados and other high winds, external flooding, internal fire, aircraft impact, and any other site-specific events that could be important.

initiating event. An event that has the potential to be the first step in a chain leading to an accident condition. Since the focus of most of the PSA work completed thus far for Davis-Besse is on the risks associated with power operation, most initiating events are failures that create a demand for a plant trip. In some cases, the initiating event may actually represent a broad set of individual events, such as all the failures that could cause a turbine trip but that would not have any other significant direct effect on plant response. An initiating event may occur at any mode of operation, depending on the scope of the analysis.

initiator. Same as initiating event.

internal event. An initiating event due to a fault within a plant system. The term is used primarily to distinguish the scope of an analysis that may or may not include external events. Examples of internal initiating events included in the PSA are various sizes of loss-of-coolant accidents (LOCAs), turbine trip, loss of main feedwater, loss of offsite power, loss of RCS makeup, loss of a dc bus, and partial or total losses of component cooling water or service water.

internal flood. A flood originating within plant equipment, such as due to a break in a condenser circulating water or service water line, spurious actuation of a fire suppression system, or loss of isolation when a heat exchanger is opened for maintenance. Traditionally, internal floods have been grouped with other external events because of their potential for acting as sources of common-cause failure and because it is their effects external to the systems needed to preserve core cooling that are of interest, rather than the unavailability created by the initiating fault itself. To add to the confusion, however, (beginning with the IPE program) it is becoming more common to include internal flooding as an internal event.

living PSA. A probabilistic safety assessment that, after initial completion, is updated on a periodic or continuous basis as changes are made to the plant and as additional experience is gained so that any conclusions made regarding risk or safety issues reflect the current configuration of the plant.

logic model. Usually, in PSA, an event tree or fault tree. The logic model lays out in a logical sequence of events the way in which a system could fail or an accident could occur, and can be evaluated in some manner to support estimation of the probability of system failure or the frequency of the accident.

minimal cut set. A cut set in which all of the events contained in it must occur for the top event of the fault tree to occur (i.e., it does not contain more than the minimum set of failures needed to cause the top event to occur).

OR gate. A logical operator used in constructing a fault tree. An OR gate signifies an event that occurs if any of the events or conditions that are inputs to it occurs.

probabilistic risk assessment (PRA). Same as probabilistic safety assessment (PSA).

probabilistic safety assessment (PSA). A systematic evaluation of the frequencies and consequences of accidents for a facility or process. This entails carefully identifying the various ways in which accidents could result from initiating events, component failures, human interactions, etc.; estimating the frequencies and probabilities of these individual events to permit the overall frequencies of the accidents to be estimated; and characterizing the severity of the accidents by some measure (i.e., core damage vs. no core damage, various conditions in containment, or offsite consequences).

release category. A convenient grouping of large numbers of accident types that would have similar characteristics with respect to severity of radiological release. Typically, in a PSA that includes an assessment of offsite consequences, a set of consequences is calculated for each release category.

reliability. The characteristic of an item expressed by the probability that it will perform its intended function when it is called upon and for the duration of the time interval(s) for which it is required to perform.

risk achievement worth. A measure of importance for an event that is calculated by setting the probability of the event to unity (1.0, corresponding to failure of the component), then dividing the resulting overall probability of the top event by the probability of the top event with the event of interest at its nominal probability. Risk achievement worth (RAW) is useful as a means to identify events for which it is important to maintain adequate reliability, or for which the results could be particularly sensitive because of uncertainty.

risk monitor. A tool for incorporating information on the current status of plant equipment into the models or results from a PSA to determine the instantaneous level of risk. Such a tool can aid in planning equipment maintenance and other activities in such a manner as to minimize the potential for a core-damage accident.

risk reduction worth. A measure of importance for an event that is calculated by setting the probability of the event to zero, then dividing the resulting overall probability of the top event by the probability of the top event with the event of interest at its nominal probability. Risk reduction worth is of interest primarily with respect to identifying areas in which improvements or further redundancy would be particularly effective in reducing risk.

severe accident. An accident that results in catastrophic fuel rod failure, core degradation, and the release of fission products into the reactor vessel, the reactor containment building, or to the environment.

sequence (minimal) cut set. A cut set that is obtained from the solution of a fault tree representing a complete accident sequence. Such a fault tree is formed by combining the fault trees for each of the individual events that make up the event tree defining the sequence of interest.

standby failure. A failure that occurs while a component is idle and that renders the component unavailable to respond when there is a demand for it. For example, corrosion of the stem for a motor-operated valve could cause the valve to fail to operate when it was needed.

top event. Most commonly, the highest level of failure for which a fault tree is developed. For example, the top event for a fault tree of the auxiliary feedwater system might be stated as "insufficient flow from AFW pumps to steam generators; flow from at least one of three pumps to at least one of two steam generators required". This high-level definition of system failure would then be developed layer by layer down to the train-level, and then component-level, faults that could cause it to occur.

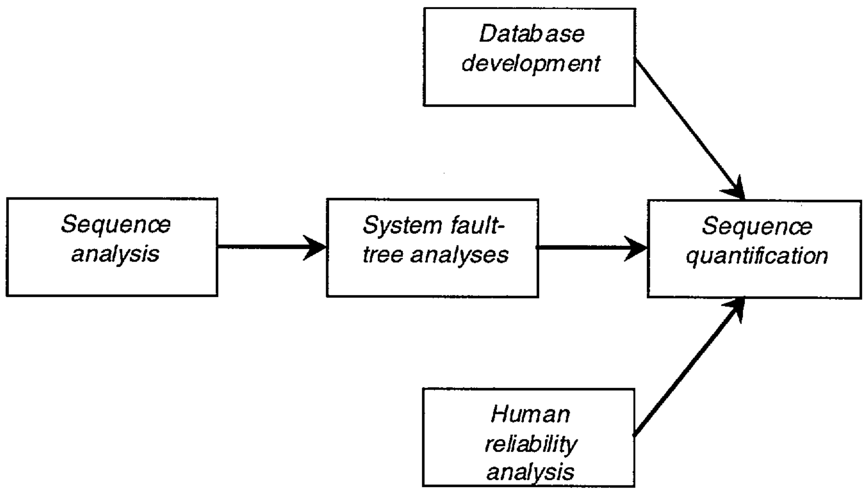
unavailability. The probability that a component, system, etc. will be unable to perform its intended function at the time it is called upon. Also, the fraction of a prior period of time during which the component, system, etc. was unable to fulfill its function.

vulnerability. Generally, a feature of the plant design, operating, or maintenance practices, etc., that could result in a particularly large chance of contributing to occurrence of core damage or failure of containment during a severe accident. The NRC's stated purpose in requesting the IPE and IPEEE was to search for severe-accident vulnerabilities, although they left the definition to the individual utilities. Toledo Edison, in its IPE submittal, considered a vulnerability to be a plant feature that would compel action on the part of the plant owner to reduce risk, irrespective of regulatory pressure. This would arise if the one or a few aspects of the plant were assessed to cause the core-damage frequency to be much higher than once in 10,000 years; if one or a few features were substantially higher contributors to core-damage frequency than all other contributors; or if the frequency of core damage was very sensitive to a highly uncertain aspect of plant response.

Appendix B
Index to Detailed Documentation

As described in Section 2, the PSA consists of many different types of analytical tasks. Each of these has been documented extensively to facilitate review of the work and to make any future updates or applications more efficient. This appendix provides a road map to the detailed documentation maintained in the Nuclear Engineering files (mostly in the form of task-specific notebooks).

The documentation is organized according to the five major task areas, as shown below:



Sequence Analysis

The sequence analysis encompasses two major efforts: the identification of initiating events, and the delineation of core-damage sequences through the development of event trees. The notebooks for the initiating event analysis document the identification of the initiating events to be considered and the estimation of their frequencies of occurrence. The sequence analysis notebooks describe the success criteria applied in the development of the core-damage sequences; the event trees that define the core-damage sequences for LOCA and transient initiators; the fault-tree logic that ties the event-tree sequences to the system failures developed in the fault trees; and the logic for actually quantifying the core-damage frequencies. A copy of each of the documents or analyses used as a basis in developing the success criteria is also provided for ease of reference. The contents of the notebooks are summarized in Table B1.

Table B1. Contents of Notebooks for Sequence Analyses

Volume	Contents
	Initiating Event Analysis
1	Summary • Approach to identifying initiating events • Definition of events selected for analysis • Summary of initiating event frequencies
2	Initiator Frequencies from Generic Data • Critical hours for US nuclear power plants • Frequency calculations from generic experience for – Large and medium LOCA – Steam generator tube rupture – Reactor vessel rupture – Spurious SFAS signal – Feedwater or steam line break
3	Initiator Frequencies from Plant-Specific Data • Davis-Besse transient history • Critical hours for Davis-Besse • Frequency calculations from plant-specific experience for – Reactor trip – Loss of main feedwater – Loss of offsite power – Loss of power from bus YAU – Loss of power from bus YBU – Loss of instrument air – Condenser/condensate faults following trip • Summary of transient initiating event frequencies
4	Initiator Frequencies from Plant-Specific Fault Trees • Frequencies for interfacing-system LOCAs • Frequencies for flood initiators • Frequencies for system-specific intiators – Loss of makeup – Loss of component cooling water – Loss of service water
	Sequence Development
1	Description of Sequence Development • Event trees, supporting logic, and sequence descriptions for – Loss-of-coolant accidents – Transients • Assignment of sequences to plant-damage states
2	References and Supporting Calculations

System Analysis Notebooks

A system notebook was developed for each of the systems modeled in the Davis-Besse PSA. Table B2 lists the system notebooks that are maintained to provide detailed documentation for the system fault trees. Each of the individual system notebooks contains the following sections:

- **Overview.** A brief description of the system and its role in the PSA.
- **System Description.** A description of the system including the following:
  - A functional diagram of the system and description of normal and off-normal configurations,
  - System boundaries in the PSA model,
  - Dependencies modeled in the PSA,
  - Actuation and control,
  - Operating practices,
  - Test and maintenance activities.
- **System Model.** A description of the role of the system in the PSA including the following:
  - Top events and functions supported by the top events,
  - Review of potential initiating events associated with the system,
  - Description of the development of the system fault tree including modeling assumptions and selection of common-cause failures.
- **Reliability Data**. Summary of quantification of any events unique to the particular system, including the following:
  - Quantification of special events other than nominal component faults,
  - Exposure times for time-dependent failures,
  - Unavailabilities for test and maintenance activities.
- **Summary of System Reliability.** Summary of important system failure modes based on review of the system cut sets and system-specific initiating event frequencies.
- **References.** Listing of all references used to develop the system fault tree model.
- **Attachments.** A printout of the system fault tree, listing of all system basic events, listing of system gates and a copy of system cut sets.

Data Base Notebooks

Separate notebooks have been assembled for each type of reliability data needed for the PSA. Table B3 provides a list of the specific notebooks and a summary of their contents. The types of data documented include the following:

Table B2. Summary of System Notebooks

Volume	System
1	125/250 vdc and 120 vac
2	13.8 kv System
3	4160 v System
4	Auxiliary Feedwater
5	Component Cooling Water
6	Circulating Water
7	Condenser/Condensate
8	Containment Air Cooling
9	Containment Isolation
10	Containment Spray
11	ECCS Room Coolers
12	Emergency Diesel Generators and Station Blackout Diesel Generator
13	Decay Heat Removal / Low Pressure Injection
14	High Pressure Injection
15	Instrument Air
16	Low Voltage Switchgear Ventilation
17	Low Voltage Electric Power
18	Main Feedwater
19	Main Steam
20	Makeup
21	Reactor Coolant System—Reactor Coolant Pump Seals
22	Reactor Coolant System—Pressurizer Spray
23	Reactor Coolant System—PORV / Pressurizer Safety Valves
24	Service Water
25	Safety Features Actuation System
26	Turbine Plant Cooling Water

- **Generic Data**. Reliability parameter estimates based on generic data were calculated for all components included in the PSA models. Generic data was used in the PSA models for components for which plant-specific data was not readily available. Generic data was also combined with available plant-specific experience through a process of Bayesian updating to produce plant-specific failure rates. Various generic data sources were reviewed and relevant sources were aggregated into composite estimates for each component type and failure mode. The specific generic data sources used for each component aggregation are identified in the data sheets contained in the generic data notebook.

- **Plant-Specific Data.** Plant-specific data was collected for those components determined to be most risk-significant in previous PRA studies. The method used in developing the plant specific data applied the principles of Bayesian analysis to combine generic data with plant-specific data. This approach was used to supplement the limited amount of data available from Davis-Besse plant records with the significant amount of industry experience. The notebooks for plant-specific data include a report describing in detail the methods used in the data analyses and the failure rates obtained, and the details supporting the development of each individual failure rate. This includes the following for each type of component:
  - A summary sheet for each failure event collected from the Davis-Besse experience base, including the date, failure mode, applicable reference to a source record (a PCAQR or maintenance work order), and a review of common-cause potential;
  - A summary of operating experience, including demands and operating hours and, if applicable, standby hours, with references and clearly stated assumptions;
  - A worksheet with the actual failure rate calculation, in which the raw data is evaluated to arrive at a lognormal probability distribution. The raw failure information is also used to update the generic data to arrive at the plant-specific failure rate actually used in the PSA. This sheet also graphically shows the generic distribution, the raw plant-specific distribution and the updated distribution

- **Common-Cause Data.** The estimation of common-cause factors was based on state-of-the art methods and industry-wide experience. The notebook contains a description of the methods used and a summary of all the common-cause parameters developed for use in the PSA. A calculation sheet is provided for each type of component and failure mode.

- **Test and Maintenance Unavailabilities.** Plant-specific data was collected for system unavailability due to testing and maintenance activities. The corresponding notebook contains a description of the methods used to calculate unavailability, a summary of all testing and maintenance unavailability events, and the worksheets used to calculate the relevant probabilities.

- **Offsite Power Data.** Summary of the calculations relating to the frequency of loss of offsite power due to various causes, the distribution of restoration times, and the probabilities of failure to restore power within the relevant times for various sequence conditions.

Table B3. Reliability Data Base Notebooks

Volume	Contents
	Generic Failure Rates
♦	Generic Data—Aggregation Reports
	Plant-Specific Failure Rates
1	Summary Report • Data collection and analysis methods • Tabulation of calculated failure rates
2	Failure Rate Calculations for Mechanical Equipment • Auxiliary feedwater pumps • Motor-driven feed pump • High pressure injection pumps • Low pressure injection pumps • Makeup pumps • Reactor coolant pumps • Component cooling water pumps • Service water pumps • Dilution pump • Containment spray pumps • Containment air coolers • Low voltage switchgear room ventilation • Instrument air compressors • Air receivers
3	Failure Rate Calculations for Mechanical Equipment (continued) • Air filters • Air dryers
4	Failure Rate Calculations for Electrical Equipment • 4160 v buses • 13.8 kv buses • 480 v buses • 13.8 k bus-tie breakers • 4160 v bus-tie breakers • 480 v breakers • Emergency diesel generators • Station blackout diesel generator • Batteries • Battery chargers • C/D bus-tie breakers
5	Failure Rate Calculations for Motor-Operated Valves • Summary of failure rates for motor-operated valves • Auxiliary feedwater valves • Component cooling water valves • Core flood valves • Decay heat valves

Table B3. Reliability Data Base Notebooks (continued)

Volume	Contents
	Plant-Specific Failure Rates (continued)
6	Failure Rate Calculations for Motor-Operated Valves (continued) • High pressure injection valves • Makeup valves • Reactor coolant valves • Service water valves
7	Failure Rate Calculations for Air-Operated Valves • Summary of failure rates for air-operated valves • Component cooling water valves • Core flood valves • Decay heat valves • Makeup valves
8	Failure Rate Calculations for Miscellaneous Valves • Solenoid valves • Pressurizer pilot-operated relief valve • Atmospheric vent valves • Main steam isolation valves • Main steam safety valves • Turbine bypass valves • Temperature control valves
9	Failure Rate Calculations for Manual valves • High pressure injection valves • Decay heat valves • Core flood valves • Reactor coolant valves • Makeup valves • Auxiliary feedwater valves • Service water valves • Component cooling water valves
10	Failure Rate Calculations for Check Valves • High pressure injection valves • Decay heat valves • Core flood valves • Makeup valves • Auxiliary feedwater valves • Component cooling water valves

Table B3. Reliability Data Base Notebooks (continued)

Volume	Contents
	Common-Cause Failure Rates
♦	Summary Report • Summary of generic common-cause experience • Calculation of common-cause (MGL) parameters • Calculation of common-cause factors
	Test and Maintenance Unavailabilities
♦	Summary Report • Summary of test and maintenance unavailabilities • Worksheets for maintenance unavailability calculations • Worksheets for test unavailability calculations
	Offsite Power Data
♦	Summary Report • Summary of data needs, methods, and results • Calculation details for frequency of loss of offsite power • Calculation sheets for non-recovery of power

Human Reliability Analysis Notebooks

The human reliability analysis (HRA) notebooks provide a description of the methods used to quantify human interactions and provide copies of the worksheets used to quantify all interactions and combinations of interactions. Table B4 summarizes the contents of the HRA notebooks.

Sequence Quantification Notebooks

The set of sequence quantification notebooks provides a guide to the methods used for quantifying the PSA model and collects the results. Sequence quantification involves solving the master fault tree to identify the combinations of equipment failures that could lead to core damage (the sequence cut sets). A master PRAQUANT file is used to automate the PSA accident sequence quantification. A truncation limit, flag file, mutually exclusive file, success events, data base, and master fault tree file are defined for each sequence. The program QRECOVER is used to automate the recovery process. The quantification notebook provides a copy of the master PRAQUANT file, the master flag files, the mutually exclusive file, the success events and the master recovery file. The cut sets for each of the core-damage sequences are maintained in the results notebooks. Table B5 summarizes the contents of the sequence quantification notebooks.

Table B4. Human Reliability Analysis Notebooks

Volume	Contents
1	Summary Report • Description of methods used for human reliability analyses • Summary of results
2	Human Reliability Analysis Worksheets • Type A (pre-initiator) human interactions • Type C_P (procedure-based post-initiator) human interactions • Type C_R (non-proceduralized recovery) human interaction
3	Human Reliability Analysis Worksheets (continued) • Combinations of interactions • Recovery modules • Miscellaneous notes and supporting calculations

Table B5. Sequence Quantification Notebooks

Volume	Contents
	Quantification Summary
♦	Summary Report • Master PRAQUANT file • Master fault tree • Flag files • Mutually exclusive files • Recovery files • Success trees
	Core-Damage Results
1	Core-Damage Cut Sets for LOCA Sequences • Loss-of-coolant accidents • Steam generator tube ruptures
2	Core-Damage Cut Sets for Transient Sequences • Sequence TQU • Sequence TQX • Sequence TKBU • Sequence TKBP • Sequence TKBL
3	Core-Damage Cut Sets for Transient Sequences (continued) • Sequence TBU • Sequence TBWX • Sequence TBLX • Sequence TBQX • Sequence TBP • Sequence TKBL

Back-End Analysis Notebooks

The back-end analysis analysis notebooks include a quantification summary report, a plant-damage state notebook, containment event tree notebooks, and a radionuclide release notebook. These notebooks describe the development and quantification of the Level 2 analysis. The contents of the notebooks are summarized in Table B6.

Table B6. Back-End Analysis Notebooks

Volume	Contents
	Quantification Summary
♦	Summary Report • Master PRAQUANT File • Master Fault Tree • Flag Files • Mutually exclusive files • Recovery Files
	Plant-damage state
♦	Summary Report • Attributes of Plant-damage states • Definition of Core-Damage Bins • Bridge Tree • Quantification of the Frequencies for Plant-damage states
	Containment Event Tree
1	Summary Report • Development of the Containment Event Tree • Top Events in the Containment Event Tree
2	Investigation of Specific Issues • Ex-Vessel Corium Coolability • Submerged Vessel Corium Cooling • Creep Rupture • Flammable Gas Generation and Combustion • Containment Failure Characterization
3	Hydrogen Burn Calculations
4	Containment Event Tree Quantification • Master Fault Tree • Flag Files • Database
	Radionuclide Release Category
♦	Summary Report • Estimation of Release Fractions • Definitions of Release Categories