ITS NRC Questions

Id	**1471**
NRC Question Number	**ALK-009**
Category	**Editorial**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	**A-2**
JFD Number	
JFD Bases Number	
Page Number (s)	**Vol. 5, pg 13 of 63**
NRC Reviewer Supervisor	**Rob Elliott**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**Please verify if this is a typo that needs to (will) be corrected: Discussion of Changes for ITS 3.0, A02, first paragraph in 3rd bullet: "… upon discovery *or* failure to meet an LCO…" should be "… upon discovery *of* failure to meet an LCO …."**
Attach File 1	
Attach File 2	
Issue Date	**1/11/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**1/11/2010 11:54 AM**
Notification	**NRC/LICENSEE Supervision Audrey Klett**

Licensee Response/NRC Response/NRC Question Closure

Id	**1791**
NRC Question Number	**ALK-009**
Select Application	**Licensee Response**
Response Date/Time	**1/19/2010 7:55 AM**
Closure Statement	
Response Statement	**The NRC reviewer is correct in that the word "or" should be "of." A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS conversion amendment.**
Question Closure Date	
Attachment 1	**ALK-009 Markup.pdf** (699KB)
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**1/19/2010 7:54 AM**
Modified By	
Date Modified	

DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the Kewaunee Power Station (KPS) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 3.0, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.0.a states, "Compliance with the LIMITING CONDITIONS FOR OPERATION contained in the succeeding TSs is required during the operational MODES or other conditions specified therein; except that upon failure to meet the LIMITING CONDITIONS FOR OPERATION, the associated ACTION requirements shall be met." ITS 3.0.1 states, "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8." This results in several changes to the CTS.

- Certain phrases are revised to be consistent with the equivalent phrase used in the ITS. Specifically, "operational MODES or other conditions specified therein" is changed to "MODES or other specified conditions in the Applicability" to be consistent with the ITS definition of MODE and the terminology used in the ITS.

  These changes are acceptable because they result in no change in the intent or application of the Technical Specifications, but merely reflect editorial preferences used in the ITS.

- The phrase "Compliance with the LIMITING CONDITIONS FOR OPERATION contained in the TS is required" is replaced with "LCOs shall be met." This change is made consistent with the ITS.

  This change is acceptable because it is an editorial change that does not change the intent of the requirements.

- The phrase "except that upon failure to meet the LIMITING CONDITIONS FOR OPERATION, the associated ACTION requirements shall be met" is moved from CTS 3.0.a to ITS LCO 3.0.2, which states that upon discovery ~~of~~ of failure to meet an LCO, the Required Actions of the associated Conditions shall be met.

  The change is acceptable because moving this information within the Technical Specifications results in no change in the intent or application of ACTIONS.

- The phrase "except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8" is added in ITS LCO 3.0.1. ITS LCO 3.0.2 describes the appropriate actions to be taken when ITS LCO 3.0.1 is not met. ITS LCO 3.0.7 describes Test

Licensee Response/NRC Response/NRC Question Closure

Id	**1801**
NRC Question Number	**ALK-009**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**1/19/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision**
Added By	**Audrey Klett**
Date Added	**1/19/2010 8:14 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1531**
NRC Question Number	**ALK-010**
Category	**Editorial**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	**A-3**
JFD Number	
JFD Bases Number	
Page Number (s)	**Vol. 5, page 14 of 63**
NRC Reviewer Supervisor	**Rob Elliott**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**Please verify that the addition of the word "not" in the quotation contained in the following statement is a typo that needs to (will) be corrected: Discussion of Changes for ITS 3.0, A03, first paragraph in 2nd bullet: "The sentence is change, in ITS LCO 3.0.2, to state 'If the LCO is *not* [emphasis added] met or is no longer applicable prior to expiration of the specified Completion Time(s) ….'"**
Attach File 1	
Attach File 2	
Issue Date	**1/22/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**1/22/2010 7:51 AM**
Notification	**NRC/LICENSEE Supervision**

Licensee Response/NRC Response/NRC Question Closure

Id	**1951**
NRC Question Number	**ALK-010**
Select Application	**Licensee Response**
Response Date/Time	**1/28/2010 8:50 AM**
Closure Statement	
Response Statement	**The NRC reviewer is correct in that the word "not" is a typo and is incorrect. The sentence should not include the word and should read "If the LCO is met or is no longer applicable..." Furthermore, KPS has noted that a typo exists in the same bullet and will also be corrected at this time. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS conversion amendment.**
Question Closure Date	
Attachment 1	**ALK-010 Markup.pdf** (699KB)
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Melissa Krcma** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**1/28/2010 8:50 AM**
Modified By	
Date Modified	

Exceptions LCOs, which are exceptions to other LCOs. ITS LCO 3.0.8 addresses snubber inoperabilities, which is also an exception to other LCOs.

This change is acceptable because adding the exceptions for LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8 prevent a conflict within the Applicability section. This addition is needed for consistency in the ITS requirements and does not change the intent or application of the Technical Specifications.

These changes are designated administrative because they are editorial and result in no technical changes to the Technical Specifications.

A03 CTS 3.0.b states, "Noncompliance with a TS shall exist when the requirements of the LIMITING CONDITIONS FOR OPERATION and associated ACTION requirements are not met within the specified time intervals. If the LIMITING CONDITIONS FOR OPERATION is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required." ITS LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated." This results in several changes to the CTS.

- CTS 3.0.b is revised to include an exception for LCO 3.0.6. LCO 3.0.6 is a new allowance that takes exception to the ITS LCO 3.0.2 requirement to take the associated ACTION requirements when a LIMITING CONDITION FOR OPERATION is not met. This exception is included in LCO 3.0.2 to avoid conflict between the applicability requirements.

  This change is acceptable because it includes a reference to a new item in the ITS and results in no change to the CTS. Changes resulting from the incorporation of LCO 3.0.6 are discussed in Discussion of Change (DOC) A08.

- The second sentence of CTS LCO 3.0.b states, "If the LIMITING CONDITIONS FOR OPERATION is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not d required." The sentence is change, in ITS LCO 3.0.2, to state "If the LCO is ~~not~~ met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated."

  This change is acceptable because, while worded differently, both the CTS and ITS state that ACTIONS do not have to be completed once the LCO is met or is no longer applicable. ITS LCO 3.0.2 also adds the phrase "unless otherwise stated." There are some ITS ACTIONS that must be completed, even if the LCO is met or is no longer applicable. This change is acceptable because it reflects a new feature in the ITS which does not exist in the CTS. The technical aspects of these changes are discussed in the appropriate ITS sections.

Attachment 1, Volume 5, Rev. 0, Page 14 of 63

Licensee Response/NRC Response/NRC Question Closure

Id	**1981**
NRC Question Number	**ALK-010**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**2/2/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision**
Added By	**Audrey Klett**
Date Added	**2/2/2010 8:21 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1651**
NRC Question Number	**ALK-011**
Category	**Technical**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	**A-6**
JFD Number	
JFD Bases Number	
Page Number(s)	**Vol. 5, page 15 of 63**
NRC Reviewer Supervisor	**Rob Elliott**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**DOC A06 for ITS 3.0 states, in part, "Certain CTS contain ACTIONS such as 'Declare the {supported system} inoperable and take the ACTIONS of {its Specification}.' In many cases, the supported system would likely already be considered inoperable. The implication of this presentation is that the ACTIONS of the inoperable supported system would not have been taken without the specific direction to do so." For each CTS that contains ACTIONS such as, "Declare the {supported system} inoperable and take the ACTIONS of {its Specification}", please clarify if adopting the STS LCO 3.0.6 would result in a less restrictive change for these TS. If it does, then please denote this change as less restrictive.**
Attach File 1	
Attach File 2	
Issue Date	**1/29/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**1/29/2010 2:27 PM**
Notification	**NRC/LICENSEE Supervision Audrey Klett**

Licensee Response/NRC Response/NRC Question Closure

Id	**2181**
NRC Question Number	**ALK-011**
Select Application	**Licensee Response**
Response Date/Time	**2/15/2010 3:50 PM**
Closure Statement	
Response Statement	**The statement in question is to clarify that the CTS wording means that actions of supported systems do not have to be taken until directed by the support system actions, which is consistent with CTS use and application. In the ISTS, a similar approach is taken. That is, when a support system is inoperable, it may direct, either immediately or after some length of time, to cascade to the supported system Actions (i.e., take the actions in the support system). This change simply specifies that this CTS approach will be maintained in the ITS, thus the change is administrative. However, if the CTS does not allow a delay in the cascade requirement and the ITS does, then this delay is evaluated and justified in each individual Specification. For example, ITS 3.8.3 allows a delay in declaring the EDG inoperable when fuel oil or staring air requirements are not met. These less restrictive changes are justified in ITS 3.8.3 DOCs. KPS believes that each individual Specification adequately justifies any extension in declaring the support system inoperable and taking the Actions of the support system. Furthermore, this Discussion of Change is consistent with the DOCs used in other ITS conversions and approved by the NRC.**
Question Closure Date	
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**2/15/2010 3:50 PM**
Modified By	
Date Modified	

Licensee Response/NRC Response/NRC Question Closure

Id	**2201**
NRC Question Number	**ALK-011**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**2/17/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**
Added By	**Audrey Klett**
Date Added	**2/17/2010 10:06 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1671**
NRC Question Number	**ALK-013**
Category	**Technical**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	**L-1**
JFD Number	
JFD Bases Number	
Page Number (s)	**Att.1, v.5, pg 55 of 63**
NRC Reviewer Supervisor	**Rob Elliott**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**The 10 CFR 50.92 evaluation for L01 in Attachment 1, Volume 5, page 55 of 63, discusses TSTF-359, Rev. 8; however, in Enclosure 3 to the ITS license amendment request letter, this TSTF is not listed to be adopted in whole or in part in the KPS ITS. Is it KPS's intention to fully adopt this TSTF?**
Attach File 1	
Attach File 2	
Issue Date	**2/18/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**2/18/2010 9:53 AM**
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**

Licensee Response/NRC Response/NRC Question Closure

Id	**2271**
NRC Question Number	**ALK-013**
Select Application	**Licensee Response**
Response Date/Time	**2/25/2010 10:20 AM**
Closure Statement	
Response Statement	**Enclosure 3 to the ITS Submittal includes only those TSTFs that are not already incorporated into NUREG-1431, Rev. 3.0. That is, it lists TSTFs that are to be incorporated into Revision 3.0 of NUREG-1431 (the latest issued revision of the NUREG). TSTF-359, Rev. 8 (and Rev. 9, which is the most current version of the TSTF) is already incorporated completely into NUREG-1431, Rev. 3.0, therefore, it is not listed in Enclosure 3. Kewaunee has incorporated this TSTF as typed into NUREG-1431, Rev. 3.0, as modified by minor changes identified in the ISTS Markup and justified by the applicable JFDs.**
Question Closure Date	
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Robert Hanley** **Jerry Jones** **Bryan Kays** **Audrey Klett**
Added By	**David Mielke**
Date Added	**2/25/2010 10:22 AM**
Modified By	
Date Modified	

Licensee Response/NRC Response/NRC Question Closure

Id	**2421**
NRC Question Number	**ALK-013**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**3/4/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**
Added By	**Audrey Klett**
Date Added	**3/4/2010 7:47 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1681**
NRC Question Number	**ALK-014**
Category	**Technical**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	
JFD Number	
JFD Bases Number	**9**
Page Number (s)	**att.1, v.5, pg 53 of 63**
NRC Reviewer Supervisor	**Rob Elliott**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**Please clarify the WOG STS section and text for which JFD #9 for ITS 3.0 Bases (Volume 5, page 53 of 63) applies. The JFD 9 designator does not appear in the margins of the marked-up WOG STS Bases.**
Attach File 1	
Attach File 2	
Issue Date	**2/18/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**2/18/2010 9:56 AM**
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**

Licensee Response/NRC Response/NRC Question Closure

Id	**2281**
NRC Question Number	**ALK-014**
Select Application	**Licensee Response**
Response Date/Time	**2/25/2010 10:25 AM**
Closure Statement	
Response Statement	**Kewaunee has reviewed the ISTS Bases Markup and determined that JFD 9 is not used in the ITS submittal for this Section. The JFD will be deleted. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS conversion amendment.**
Question Closure Date	
Attachment 1	**ALK-014 Markup.pdf** (703KB)
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Robert Hanley** **Jerry Jones** **Bryan Kays** **Audrey Klett**
Added By	**David Mielke**
Date Added	**2/25/2010 10:28 AM**
Modified By	
Date Modified	

JUSTIFICATION FOR DEVIATIONS
ITS 3.0 BASES, LCO AND SR APPLICABILITY

8. This statement has been deleted since the Kewaunee Power Station ITS submittal does not state the snubbers do not meet the 10CFR50.36(c)(2)(ii) criteria. This is also consistent with the NRC SERs for DC Cook Units 1 and 2 and Monticello ITS amendments.

Not used.

9. ~~This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.~~

10. These changes are made to be consistent with the wording of LCO 3.0.7.

11. The ITS SR 3.0.1 Bases allows credit to be taken for unplanned events that satisfy Surveillances. The Bases further states that this allowance also includes those SRs whose performance is normally precluded in a given MODE or other specified condition. This portion of the allowance has been deleted. As documented in Part 9900 of the NRC Inspection Manual, Technical Guidance - Licensee Technical Specifications Interpretations, and in the Bases Control Program (ITS 5.5.12), neither the Technical Specifications Bases nor Licensee generated interpretations can be used to change the Technical Specification requirements. Thus, if the Technical Specifications preclude performance of an SR in certain MODES (as is the case for some SRs in ITS Section 3.8), the Bases cannot change the Technical Specifications requirement and allow the SR to be credited for being performed in the restricted MODES, even if the performance is unplanned.

12. Changes have been made for consistency with similar discussions/terminology in the Bases.

13. The ISTS SR 3.0.1 Bases provides two examples of when it is not possible to perform post maintenance testing in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. The second example is not valid for Kewaunee Power Station and therefore it has been deleted.

14. The proper LCO title and Applicability have been provided.

15. Changes are made to be consistent with changes made to the Specification.

Attachment 1, Volume 5, Rev. 0, Page 53 of 63

Licensee Response/NRC Response/NRC Question Closure

Id	**2431**
NRC Question Number	**ALK-014**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**3/4/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**
Added By	**Audrey Klett**
Date Added	**3/4/2010 7:48 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1751**
NRC Question Number	**ALK-015**
Category	**Technical**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	**M-1**
JFD Number	
JFD Bases Number	
Page Number(s)	**Att. 1, vol. 5, pg 19 of 63**
NRC Reviewer Supervisor	**Carl Schulten**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**Attachment 1, Volume 5, page 19 of 63, DOC M01 discusses KPS's intent to incorporate LCO 3.0.8 into the KPS ITS. Please provide the following:** **a. corrected pages that show the removal of CTS LCO 3.14.b as an "R" DOC (Relocated Specifications), and** **b. a safety basis for why the 12-hour requirement in LCO 3.0.8b, as described in TSTF 372, Revision 4, and the NRC's model safety evaluation of this TSTF, cannot be met and therefore presents a hardship (e.g., operational challenges, reductions in safety, etc.) for KPS. Include in the response plant-specific design and licensing basis information, deterministic insights, and risk insights that justify the proposed time period for ITS LCO 3.0.8b. Describe the extent of the hardship as it applies to all KPS TS systems with snubbers. Also describe how the proposed 24-hour time allowance was derived from this safety basis analysis.**
Attach File 1	
Attach File 2	
Issue Date	**2/23/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	

Date Added	**2/23/2010 2:08 PM**
Notification	**NRC/LICENSEE Supervision** **Audrey Klett** **Carl Schulten**

Licensee Response/NRC Response/NRC Question Closure

Id	**2491**
NRC Question Number	**ALK-015**
Select Application	**Licensee Response**
Response Date/Time	**3/10/2010 7:00 AM**
Closure Statement	

Response Statement

As stated in the NRC Model Safety Evaluation for TSTF-372, Section 2.0, LCO 3.0.8 was added into the ISTS because of a difference of interpretation as to whether or not the 72 hour delay to enter the actions for the supported equipment in the old STS Snubber Technical Specification could be applied after the Snubber Technical Specification had been relocated to a licensee-controlled document as part of a licensee's conversion to the ISTS. The Safety Evaluation states that the NRC's position is that after relocation, the 72 hour delay cannot be used if the inoperable snubber also results in inoperability of the supported system. Thus, plants that had already converted to ISTS and had relocated the Snubber Technical Specification to a licensee-controlled document needed an allowance to delay entry into the supported systems when a required snubber is inoperable. TSTF-372 and the NRC CLIIP provided the requirements for these plants to adopt a new delay time for when a required snubber is inoperable. However, this CLIIP requirement on how to justify a new 72 hour delay time is only for plants that have already adopted the ISTS and relocated their old CTS Snubber Technical Specification.

Kewaunee Power Station (KPS) currently has a 72 hour delay time in CTS 3.14. KPS has not, as yet, relocated the Snubber Technical Specification to a licensee-controlled document. Therefore, KPS does not believe that we have to re-justify the currently allowed 72 hour delay time for one train or the proposed 24 hour delay time for two trains to adopt LCO 3.0.8. The times and requirements of proposed LCO 3.0.8 are all equal to or more restrictive than the current actions provided in CTS 3.14 when a snubber is inoperable. KPS provided a More Restrictive DOC to justify the changes from the CTS to ITS LCO 3.0.8, with respect to the actions to take when a required snubber is inoperable. This More Restrictive DOC is consistent with a previously approved ITS conversion, specifically, Davis-Besse. The NRC approved the adoption of LCO 3.0.8 from their current Snubber LCO, which is similar to the KPS Snubber LCO (i.e., 72 hours is allowed for one or more snubbers to be OPERABLE). Furthermore, the NRC asked a similar question (i.e., requesting that Davis-Besse do the requirements, including the evaluations, listed in the CLIIP). Davis-Besse proposed to either maintain the DOC as an M DOC, but also incorporate the CLIIP Section 3.2 requirements (which include only the Tier 2 restrictions) or to maintain current licensing basis by not adopting the LCO 3.0.8 and including a

Technical Specification identical to the CTS Snubber Specification, converted to ITS format.

The NRC accepted the first Davis-Besse option, which was to adopt LCO 3.0.8 but commit to the applicable Tier 2 requirements. A copy of the original RAI and all responses is attached (attachment 1). This RAI was docketed by Davis-Besse in a letter dated May 16, 2008 (ML081480464 and ML081480465). The Davis-Besse discussion of the two options is in the licensee response of 2/15/08 and the NRC acceptance of Option 1 is in the NRC reviewer response dated 4/1/2008. In addition, the NRC SER for the Davis-Besse ITS conversion, dated November 20, 2008 (ML082900600 and ML082910112), also stated that the change from the CTS Snubber Actions to adopting ITS LCO 3.0.8 was a more restrictive change. A copy of the applicable portions of the SER Table is attached (attachment 2).

In the KPS ITS submittal, KPS ensured that all the final requirements the NRC imposed on Davis-Besse as part of their Option 1 proposal were included in the more restrictive DOC. Furthermore, the commitments the NRC required of Davis-Besse are also included in the KPS submittal. These commitments are located in Attachment 5 to the ITS submittal cover letter, commitments 2 through 5. Based on the above, KPS believes that the ITS submittal, classifying the change as a more restrictive change, not requiring any CLIIP-related technical evaluations, and adopting the Tier 2 restrictions of the CLIIP (including the proposed commitments), is acceptable.

However, similar to Davis-Besse, KPS will provide the NRC the option of KPS maintaining the Current Licensing Basis of CTS 3.14. Specifically, this will require KPS to generate a new Technical Specification that is not currently in the ISTS, and not adopt LCO 3.0.8. The new Specification will maintain all the requirements of CTS 3.14 and CTS 4.14, in ISTS format. It should be noted that the current delay times and Action requirements are less restrictive than what is currently required by LCO 3.0.8.

Question Closure Date	
Attachment 1	**Attachment 1 for ALK-015.pdf** (454KB)
Attachment 2	**Attachment 2 for ALK-015.pdf** (39KB)
Notification	**NRC/LICENSEE Supervision** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**3/10/2010 7:02 AM**
Modified By	

Date Modified

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RAI Screening Required: No

Status: Approval Not Required

This is a Non RAI Dialogue

This document will not be relied upon by staff for disposition of the LAR

This document has been reviewed and information in this question contains NO SUNSI sensitive material (the checkbox to the right must be selected before this question can be submitted) **Yes**

NRC ITS TRACKING

NRC Reviewer

ID	**200712101502** Conference Call Requested? **No**
Category	**In Scope**
ITS Information	ITS Section: **3.0 Carl Schulten**; TB POC: ; JFD Number: **None**; Page Number(s): **8**; ITS Number: **None**; OSI: **None**; DOC Number: **M.2**; Bases JFD Number: **None**
Comment	**Section 3.0 Volume 5, Page 8 of 62 DOC M02 LCO 3.0.8 is added to CTS per TSTF-372, Revision 4 by M02. Adopting the TSTF changes requires submittal of the content of the CLIIP TSTF Model Application and Model Safety Evaluation in accordance with Federal Register Notice (69 FR 68412, November 24, 2004) (ADAMS ML051160013) before these changes to CTS can be processed. To efficiently process the incoming license amendment applications, the staff requests each licensee applying for the changes addressed by TSTF-3372, Revision 4, as modified, using the CLIIP to include bases for the proposed technical specification consistent with the bases proposed in the TSTF-372, Revision 4 as stated FRN. Identify and submit documentation for all differences between the content of DOC M02 and the content of a license application submittal required by the TSTF-372 CLIIP.**
Issue Date	**12/10/2007**
Close Date	**04/09/2008**

Responses

Licensee Response by Jerry Jones on 01/04/2008	The CLIIP application is based on a plant that has already adopted the ISTS. Prior to TSTF-372, the ISTS did not include any snubber requirements - they were to be relocated to a plant-controlled document. Thus, for an ITS plant to adopt TSTF-372, they would have to justify the allowances provided in ITS LCO 3.0.8. Furthermore, the CLIIP discussed this issue specifically, in that it stated that prior to ITS development, old CTS already had most of the allowances provided by ITS LCO 3.0.8, and that the TSTF was just adding them back into the ITS since they appeared to be lost when a plant adopted the ITS and relocated the old CTS requirements to plant controlled documents. Davis-Besse has not relocated the current snubber

	Techncial Specification, CTS 3.7.7, to a plant controlled document. Thus, in order to adopt the ITS LCO 3.0.8, Davis-Besse only has to justify the differences from the CTS requirements to the ITS requirements. The allowances provided in ITS LCO 3.0.8 are all more restrictive that what is allowed in CTS 3.7.7. Since the allowances provided in LCO 3.0.8 are all more restrictive than what is allowed by the Actions of CTS 3.7.7, Davis-Besse does not believe that a plant specific evaluation to justify the current 72 hour time (as is required by the CLIIP) is required. This manner of adopting LCO 3.0.8 is also consistent with a previously approved ITS conversion, Monticello, which adopted TSTF-372 as part of the conversion. Furthermore, the NRC asked a similar question, and did not require any changes to the more restrictive Discussion of Change submitted as part of the Monticello conversion. The question is still viewable on the EXCEL website (question 200512151125).
NRC Response by Carl Schulten on 01/30/2008	LCO 3.0.8 is a risk-informed TS. The three tiered approach for assessing the acccetability of risk-informed TS changes are dicussed in RG 1.177. The submittal information contained in the model application for TSTF-372 which follows the 3-tiered approach has not been provided to the NRR staff for review. There are conditions and speCific limitions to be met to adopt the allowances of TSTF-372. The shortest path to revising the Davis Besse licensing basis to incorporate LCO 3.0.8 risk-informed allowances is to submit the TSTF-372 CLIIP model application for staff review.
Licensee Response by Bryan Kays on 02/15/2008	As stated in the NRC Model Safety Evaluation for TSTF-372, Section 2.0, LCO 3.0.8 was added into the ISTS because of a difference of interpretation as to whether or not the 72 hour delay to enter the actions for the supported equipment in the old CTS Snubber Technical Specification could be applied after the Snubber Technical Specification had been relocated to a licensee-controlled document as part of a licensee's conversion to the ISTS. The Safety Evaluation states that the NRC's position is that after relocation, the 72 hour delay cannot be used if the inoperable snubber also results in inoperability of the supported system. Thus, plants that had already converted to ISTS and had relocated the Snubber Technical Specification to a licensee-controlled document needed an allowance to delay entry into the supported systems when a required snubber is inoperable. TSTF-372 and the NRC CLIIP provided the requirements for these plants to adopt a new delay time for when a required snubber is inoperable. However, this CLIIP requirement on how to justify a new 72 hour delay time is only for plants that have already adopted the ISTS and relocated the old CTS Snubber Technical Specification. Davis-Bess currently has a 72 hour delay time in CTS 3.7.7. We have not, as yet, relocated the Snubber Technical Specification to a licensee-controlled document. Therefore, Davis-Besse does not believe that we have to re-justify the currently allowed 72 hour delay time to adopt LCO 3.0.8. The times and requirements of LCO 3.0.8 are all equal to or more restrictive than the current actions provided in CTS 3.7.7 when a snubber is inoperable. Davis-Besse provided a More Restrictive DOC to justify the changes from the CTS to ITS LCO 3.0.8, with respect to the actions to take when a required snubber is inoperable. This More Restrictive DOC is consistent with a previously approved ITS conversion, specifically, Monticello Nuclear Generating Plant. The NRC approved the adoption of LCO 3.0.8 from their current Snubber LCO, which is similar to the Davis-Besse Snubber LCO. Furthermore, the NRC asked a similar question (i.e., requesting that Monticello do the requirements listed in the CLIIP); however, the question was resolved and closed without Monticello having to adopt the CLIIP requirements. The Monticello ITS submittal CTS markup and DOC, the Monticello NRC question, and the NRC SER pages

	applicable to this change are provided as an attachment. Based on the above, Davis-Besse believes that one of the two options provided below are available to resolve this issue: 1. Davis-Besse will adopt LCO 3.0.8 using the NRC precedent set for Monticello. The currently provided M DOC will be sufficient to justify the adoption of LCO 3.0.8 and the adoption will not be considered a beyond scope issue. Furthermore, Davis-Besse has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation, Section 3.2, and will commit to ensuring appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8.a is used; b) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Davis-Besse will confirm that at least one train of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Davis-Besse CTS. 2. Davis-Besse will maintain Current Licensing Basis with respect to the Snubber Technical Specification 3.7.7. This will require Davis-Besse to generate a new TS that is not currently in the ISTS, and not adopt LCO 3.0.8. It should be noted that the current delay times and Action requirements are less restrictive than what is currently required by LCO 3.0.8.
Licensee Response by Bryan Kays on 04/01/2008	Based on further discussion with the NRC reviewer, Davis-Besse has modified Discussion of Change M02 (Volume 5, Pages 22 and 23) to include the information required from the CLIIP for this change. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS Conversion Amendment. Furthermore, Davis-Besse has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation, Section 3.2, and will commit to ensuring appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8.a is used; b) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Davis-Besse will confirm that at least one train of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Davis-Besse CTS.
NRC Response by Carl Schulten	Your stated option 1 will be accceptable, provided the response to this comment

on 04/01/2008	restates Option 1, in its entirety, as the Davis Besse ITS conversion commitment for incorporating LCO 3.0.8 and resolving ID 200712101502.
Licensee Response by Bryan Kays on 04/07/2008	Based on the NRC reviewer's posting on 4/1/2008, Option #1 from the Davis-Besse 2/15/2008 response is acceptable, provided it is restated in its entirety as the commitment for incorporating LCO 3.0.8. During a phone call with the NRC reviewer on 4/7/08, the NRC reviewer stated that the Tier 2 Restrictions should be stated in the Discussion of Change (DOC) as documentation of the commitment. The following response supersedes the Davis-Besse response and draft markup posted on 4/1/08: Davis-Besse will adopt LCO 3.0.8 using the NRC precedent set for Monticello. The currently provided DOC M02 (Volume 5, Pages 22 and 23) will be sufficient to justify the adoption of LCO 3.0.8 and the adoption will not be considered a beyond scope issue. Furthermore, Davis-Besse has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation, Section 3.2, and will modify DOC M02 (Pages 22 and 23) to clearly state that the Tier 2 Restrictions will be complied with. DOC M02 will include a discussion that Davis-Besse will ensure appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8.a is used; b) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Davis-Besse will confirm that at least one train of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Davis-Besse CTS. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS Conversion Amendment.
NRC Response by Carl Schulten on 04/09/2008	The response submitted 04/01/2008 stated Davis-Besse will commit to ensuring appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. The specific commitments given in the 04/01/2008 response to 200712101502 need to be added to DOC M.2 (Section 3.0) to complete the record of the commitment in the Davis-Besse ITS license amendment application and finalize the response to 200712101502. The text that should be added to DOC M.2, which is in the 04/01/2008 response, is as follows: "Davis-Besse has reviewed the additional provisions discussed in the CLIIPS Model Safety Evaluation, Section 3.2, and will commit to ensuring appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8.a is used; b) at least one EFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Davis-Besse will confirm that at

	least one train of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection."

Date Created: 12/10/2007 03:02 PM by Carl Schulten
Last Modified: 04/09/2008 11:17 AM

Table M – More Restrictive Changes

ITS/CTS No. and DOC No.	Description of Change	ITS Requirement	CTS Requirement
3.0 M02	CTS 3.7.7 Action a provides the actions for inoperable snubbers, and requires one of the following (1, 2, or 3) within 72 hours when one or more snubbers are inoperable: 1) replace or restore the inoperable snubber(s) to OPERABLE status; 2) verify system operability with the snubber(s) inoperable by engineering evaluation; or 3) declare the supported system inoperable and follow the appropriate ACTION statement for that system. In the ITS, the actions for inoperable snubbers are incorporated into ITS limiting condition for operation (LCO) 3.0.8. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period (i.e., 12 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected system LCO(s) shall be declared not met. This changes the CTS by requiring the risk associated with inoperable snubbers to be assessed and managed and requires the snubbers to be restored to OPERABLE status in all cases, and in certain cases within a more restrictive CT.	LCO 3.0.8	3.7.7 Action a

Licensee Response/NRC Response/NRC Question Closure

Id	**2851**
NRC Question Number	**ALK-015**
Select Application	**Licensee Response**
Response Date/Time	**4/29/2010 7:05 AM**
Closure Statement	
Response Statement	**This response supplements the previous KPS response. The previous response was posted on 3/10/2010. Subsequent to posting this response, this issue has been informally discussed with the NRC. The NRC requested further information regarding the proposed 24 hour allowance when a snubber affecting both trains of a safety related system was found to be non-functional.**

The following is additional information verbally discussed with the NRC on subsequent phone conversations:

Kewaunee Nuclear Power Plant, Individual Plant Examination of External Events Summary Report," Wisconsin Public Service Corporation, June 28, 1994 (docketed number NRC-94-079) provides earth quake frequency for Kewaunee. Table 3-1, Mean Seismic Hazard Curves for the Kewaunee Site (page 3-64) identifies that the Peak Ground Acceleration (g) of 0.11 has a Mean Annual Probability of Exceedence of 1.10E-04.

TSTF-372 provides a discussion and formula for assessing risk associated with the use of LCO 3.0.8.b. The formula uses earth quake frequency for Peak Ground Acceleration (g) of 0.1 and failure probability for insulators to determine a frequency for seismic-induced LOOP. The formula also includes a time period for repairing a snubber which is used to determine incremental core damage probability during the snubber repair.

Table 3-1 of IPEEE does not provide an earth quake frequency for 0.1 g. However the frequency for 0.11 g is 1.10E-04 and the frequency for 0.01 g is 2.10E-03. By interpolation on a logarithmic scale, the frequency for 0.1 g would be 1.5E-04.

Using the TSTF-372 formula, Kewaunee's earthquake frequency is 1.5E-04. Kewaunee's proposed snubber repair period is 24 hours. Therefore, the incremental core damage probability (ICDP) for a snubber repair would be:

$$(1.5E\text{-}04/yr)(5E\text{-}02) \quad X \quad \frac{24 \text{ hours}}{8760 \text{ hours/yr}} \quad = \quad 2\ E\text{-}08$$

Using the same assumptions as TSTF-372, it can be conservatively assumed that the delta-large energy release frequency (LERF) values resulting from the snubber out of service would be at least an order of magnitude less than the delta-CDF and ICDP values, respectively. Therefore, the incremental LERF (ILERF) for Kewaunee would conservatively be 2 E-09.

For Kewaunee, the ICDP and ILERF are two orders of magnitude less than the risk assumed for normal work controls as identified in TSTF-372.

Therefore, Kewaunee considers the allowance of 24 hours to perform maintenance on a snubber associated with more than one train or subsystem of a multiple train or subsystem supported system to be acceptable for safe plant operation.

The NRC also questioned whether or not the change was properly classified as an M change, since the CTS required the affected system to be isolated after 72 hours in order to use the option of declaring the affected system inoperable. KPS believed that maintaining the system properly aligned during this additional time is actually safer with regard to operating the unit; therefore, we classified it as a more restrictive change. However, deleting this requirement could be considered as less restrictive, since KPS will not have to isolate the affected system when ITS LCO 3.0.8 is adopted. Therefore, KPS will reclassify the change as an overall L change (there remains a portion of the change that is more restrictive). A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS conversion amendment.

Question Closure Date	
Attachment 1	**ALK-015 Markup.pdf** (845KB)
Attachment 2	**ALK-015 KPS IPEEE.pdf** (15MB)
Notification	**NRC/LICENSEE Supervision** **Victor Cusumano** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele** **Carl Schulten**
Added By	**Robert Hanley**
Date Added	**4/29/2010 7:17 AM**
Modified By	
Date	

Modified

L04 ~~M01~~

INSERT 8

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 24 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.14 SHOCK SUPPRESSORS (SNUBBERS)

APPLICABILITY

Applies to the OPERABILITY of shock suppressors which are related to plant safety.

OBJECTIVE

To ensure that shock suppressors, which are used to restrain safety-related piping under dynamic load conditions, are functional during reactor operation. See CTS 3.14

SPECIFICATION

a. The reactor shall not be made critical unless all safety-related shock suppressors are OPERABLE except as noted in 3.14.b.

LCO 3.0.8

b. During power operation or recovery from inadvertent trip, if any safety-related shock suppressor is found inoperable one of the following actions shall be taken within 72 hours:

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1. The inoperable shock suppressor shall be restored to an OPERABLE condition or replaced with a spare shock suppressor of similar specifications; or

~~M01~~

2. The fluid line restrained by the inoperable shock suppressor shall, if feasible, be isolated from other safety-related systems if otherwise permitted by the TS and thereafter operation may continue subject to any limitations by the TS for that fluid line; or

3. Actions shall be initiated to shut down the reactor and the reactor shall be in a HOT SHUTDOWN condition within 36 hours.

DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

This change is designated as administrative as there is no change in the intent of the CTS and no additional flexibility is granted.

MORE RESTRICTIVE CHANGES

Not used.

M01 CTS 3.14.b provides the action for inoperable snubbers, and requires one of the following (1, 2, or 3) within 72 hours when one or more snubbers are inoperable: 1) replace or restore the inoperable snubber; 2) isolate the fluid line restrained by the inoperable snubber from other safety related systems; or 3) shut down the unit to HOT SHUTDOWN within 36 hours. In the ITS, the actions for inoperable snubbers are incorporated into ITS LCO 3.0.8. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 24 hours. At the end of the specified period (i.e., 24 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected support system LCO(s) shall be declared not met and the associated ACTIONS taken (which may include a unit shutdown). This changes the CTS by requiring the risk associated with inoperable snubbers to be restored to OPERABLE status in all cases, and in certain cases within a more restrictive Completion Time.

The purpose of CTS 3.14.b is to provide a short time (72 hours) prior to requiring the affected systems to be declared inoperable, to either restore or replace inoperable snubbers or to isolate the snubber from the safety related system. ITS LCO 3.0.8 requires the risk associated with inoperable required snubbers to be assessed and managed in all instances of snubber inoperability. ITS LCO 3.0.8 also requires all "required" inoperable snubbers to be restored to OPERABLE status within the specified Completion Times. ITS LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable, provided only a single subsystem is affected. This 72 hour time is consistent with the CTS. However, ITS LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one subsystem of a multiple subsystem supported system, and allows 24 hours to restore the snubber(s) before declaring the supported system inoperable. This 24 hour time is more restrictive than the CTS. The 24 hour Completion Time is acceptable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function. Furthermore, ITS LCO 3.0.8 requires that risk be assessed and managed. This risk assessment is not required in the CTS. The Bases for ITS LCO 3.0.8 provides guidance on how the risk must be assessed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of ITS LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. Furthermore, Kewaunee Power Station has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation (in Federal Register Notice 69 FR 68412, November 24, 2004), Section 3.2, and will ensure appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one AFW train including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s) must be available when LCO 3.0.8.a is used; b) at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Kewaunee Power Station will confirm that at least one train of systems supported by the inoperable snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restriction, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Kewaunee Power Station CTS. This change is designated as more restrictive because inoperable snubbers must be restored to OPERABLE status under certain conditions within a more restrictive Completion Time and the risk associated with inoperable snubber must always be assessed and managed.

M02 CTS 4.0.b states, "Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per …" basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, "For Frequencies specified as "once," the above interval extension does not apply." The remaining changes to CTS 4.0.b are discussed in DOC A09 and DOC L03.

The purpose of the 1.25 extension allowance to Surveillance Frequencies is to allow for flexibility in scheduling tests. This change is acceptable because Frequencies specified as "once" are typically condition based Surveillances in which the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Frequency without extension in order to avoid operation in unacceptable conditions. This change is designated as more restrictive because an allowance to extend Frequencies by 25% is eliminated in some Surveillances.

tripping an inoperable instrument channel. To allow the performance of Surveillance Requirements to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment or variable within limits, which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. ITS LCO 3.0.5 documents a formal allowance for accepted industry practices utilized in the restoration of inoperable equipment under CTS. Without this allowance, certain components could not be restored to OPERABLE status and plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. This change is designated as less restrictive because LCO 3.0.5 will allow the restoration of equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS.

L03 CTS 4.0.b states, in part, "specific surveillance intervals with a maximum allowable extension not to exceed 25% of the specified interval." ITS SR 3.0.2 includes a similar requirement, but adds the following: "If a Completion Time requires periodic performance on a "once per… " basis, the above Frequency extension applies to each performance after the initial performance." This changes the CTS by adding an allowance that if a Required Action's Completion Time requires periodic performance on a "once per…" basis, the 25% Frequency extension applies to each performance after the initial performance.

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time without extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because additional time is provided to perform some periodic Required Actions.

INSERT L04

L04 CTS 3.14.b provides the action for inoperable snubbers, and requires one of the following (1, 2, or 3) within 72 hours when one or more snubbers are inoperable: 1) replace or restore the inoperable snubber; 2) isolate the fluid line restrained by the inoperable snubber from other safety related systems; or 3) initiate action to shut down the unit to HOT SHUTDOWN within 36 hours. In the ITS, the actions for inoperable snubbers are incorporated into ITS LCO 3.0.8. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 24 hours. At the end of the specified period (i.e., 24 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected support system LCO(s) shall be declared not met and the associated ACTIONS taken (which may include a unit shutdown). This changes the CTS by: a) requiring the risk associated with inoperable snubbers to be assessed and managed; b) allowing the associated safety related system to be declared inoperable yet remain unisolated, in lieu of requiring it to be isolated to use this option; c) deleting the option to shutdown the unit; and d) decreasing the amount of time allowed to perform one of the options if snubbers in both trains of a safety related system are affected.

The purpose of CTS 3.14.b is to provide a short time (72 hours) to either restore or replace inoperable snubbers prior to requiring the affected safety related system to be isolated from the inoperable and the affected safety realted system to be declared inoperable. If the safety related system cannot be isolated from the snubber, then the CTS requires the unit to be shutdown. ITS LCO 3.0.8 requires the risk associated with nonfunctional required snubbers to be assessed and managed in all instances of snubber nonfunctionality. ITS LCO 3.0.8 also requires all "required" nonfunctional snubbers to be restored to functional status within the specified Completion Times. ITS LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable, provided only a single subsystem is affected. Furthermore, the ITS does not require the supported system to be isolated from the snubber. This 72 hour time is consistent with the CTS. However, ITS LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one subsystem of a multiple subsystem supported system, and allows 24 hours to restore the snubber(s) before declaring the supported system inoperable. Again, the ITS does not require the supported system to be isolated from the snubber. This 24 hour time is more restrictive than the CTS. The 24 hour Completion Time is acceptable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function. Furthermore, ITS LCO 3.0.8 requires that risk be assessed and managed. This risk assessment is not required in the CTS. The Bases for ITS LCO 3.0.8 provides guidance on how the risk must be assessed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of ITS LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. Furthermore, Kewaunee Power Station has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation (in Federal Register Notice 69 FR 68412, November 24, 2004), Section 3.2, and will ensure appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one AFW train including a minimum set of supporting equipment required for its successful operation) not associated with the nonfunctional snubber(s) must be available when LCO 3.0.8.a is used; b) at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the nonfunctional snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Kewaunee Power Station will confirm that at least one train of systems supported by the nonfunctional snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the nonfunctional snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restriction, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Kewaunee Power Station CTS.

The deletion of the requirement to isolate the affected system is acceptable since it will allow the system to remain in service during the time the affected train is declared inoperable and the ACTION for the inoperable train is being taken. Thus, if an accident occurs in which the train is needed, it will function as assumed. This portion of the change could be considered as less restrictive, since the system is now not required to be isolated in order to use the CTS option to declare the train inoperable, in lieu of using the CTS shutdown option. Therefore, the change is designated overall as less restrictive. The remaining portions of this overall change are more restrictive because nonfunctional snubbers must be restored to functional status under certain conditions within a more restrictive Completion Time (24 hours versus 72 hours) and the risk associated with nonfunctional snubbers must always be assessed and managed.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

Response: No.

The proposed change allows the Completion Time for periodic actions to be extended by 25 percent. This change will not physically alter the plant (no new or different type of equipment will be installed). Also, the change does not involve any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change allows the Completion Time for periodic actions to be extended by 25 percent. The 25 percent extension allowance is provided for scheduling convenience and is not expected to have a significant effect on the average time between Required Actions. As a result, the Required Actions will continue to provide appropriate compensatory measures for the subject Condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, there is a finding of "no significant hazards consideration."

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Attachment 1, Volume 5, Rev. 0, Page 63 of 63

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L04

Kewaunee Power Station (KPS) is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, Rev. 3, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

CTS 3.14.b provides the action for inoperable snubbers, and requires one of the following (1, 2, or 3) within 72 hours when one or more snubbers are inoperable: 1) replace or restore the inoperable snubber; 2) isolate the fluid line restrained by the inoperable snubber from other safety related systems; or 3) initiate action to shut down the unit to HOT SHUTDOWN within 36 hours. In the ITS, the actions for inoperable snubbers are incorporated into ITS LCO 3.0.8. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 24 hours. At the end of the specified period (i.e., 24 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected support system LCO(s) shall be declared not met and the associated ACTIONS taken (which may include a unit shutdown). This changes the CTS by: a) requiring the risk associated with inoperable snubbers to be assessed and managed; b) allowing the associated safety related system to be declared inoperable yet remain unisolated, in lieu of requiring it to be isolated to use this option; c) deleting the option to shutdown the unit; and d) decreasing the amount of time allowed to perform one of the options if snubbers in both trains of a safety related system are affected.

The purpose of CTS 3.14.b is to provide a short time (72 hours) to either restore or replace inoperable snubbers prior to requiring the affected safety related system to be isolated from the inoperable and the affected safety realted system to be declared inoperable. If the safety related system cannot be isolated from the snubber, then the CTS requires the unit to be shutdown. ITS LCO 3.0.8 requires the risk associated with nonfunctional required snubbers to be assessed and managed in all instances of snubber nonfunctionality. ITS LCO 3.0.8 also requires all "required" nonfunctional snubbers to be restored to functional status within the specified Completion Times. ITS LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable, provided only a single subsystem is affected. Furthermore, the ITS does not require the supported system to be isolated from the snubber. This 72 hour time is consistent with the CTS. However, ITS LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one subsystem of a multiple subsystem supported system, and allows 24 hours to restore the snubber(s) before declaring the supported system inoperable. Again, the ITS does not require the supported system to be isolated from the snubber. This 24 hour time is more restrictive than the CTS. The 24 hour Completion Time is acceptable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function. Furthermore, ITS LCO 3.0.8 requires that risk be assessed and managed. This risk assessment is not required in the CTS. The Bases for ITS LCO 3.0.8 provides guidance on how the risk must be assessed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of ITS LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. Furthermore, Kewaunee Power Station has reviewed the additional provisions discussed in the CLIIP's Model Safety Evaluation (in Federal Register Notice 69 FR 68412, November 24, 2004), Section 3.2, and will ensure appropriate plant procedures and administrative controls will be used to implement the applicable Tier 2 Restrictions. Specifically: a) at least one AFW train including a minimum set of supporting equipment required for its successful operation) not associated with the nonfunctional snubber(s) must be available when LCO 3.0.8.a is used; b) at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the nonfunctional snubber(s), or some alternative means of core cooling must be available when LCO 3.0.8.b is used; and c) every time the provisions of LCO 3.0.8 are used, Kewaunee Power Station will confirm that at least one train of systems supported by the nonfunctional snubbers would remain capable of performing their required safety or support functions for postulated design loads other than seismic loads. In addition, a record of the design function of the nonfunctional snubber (i.e., seismic vs. non-seismic), implementation of any applicable Tier 2 restriction, and the associated plant configuration shall be available on a recoverable basis for NRC staff inspection. These Tier 2 Restrictions are also more restrictive than what is currently required by the Kewaunee Power Station CTS.

The deletion of the requirement to isolate the affected system is acceptable since it will allow the system to remain in service during the time the affected train is declared inoperable and the ACTION for the inoperable train is being taken. Thus, if an accident occurs in which the train is needed, it will function as assumed. This portion of the change could be considered as less restrictive, since the system is now not required to be isolated in order to use the CTS option to declare the train inoperable, in lieu of using the CTS shutdown option. Therefore, the change is designated overall as less restrictive. The remaining portions of this overall change are more restrictive because nonfunctional snubbers must be restored to functional status under certain conditions within a more restrictive Completion Time (24 hours versus 72 hours) and the risk associated with nonfunctional snubbers must always be assessed and managed.

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change allows the affected train to be unisolated during the time provided to restore the affected train. Furthermore, this change decreases the amount of time allowed to restore an inoperable snubber. Not isolating the affected system or reducing the time allowed to restore a snubber is not an initiator of any accident previously evaluated. Consequenctly, the probability of an accident previously evaluated is not significantly increased. The consequences of any accident previously evaluated are the same during the Completion Time or during any reduction of the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change allows the affected train to be unisolated during the time provided to restore the affected train. Furthermore, this change decreases the amount of time allowed to restore an inoperable snubber. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. While the affected system is not required to be isolated from the inoperable snubber, normal plant operations are not affected. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change allows the affected train to be unisolated during the time provided to restore the affected train. Furthermore, this change decreases the amount of time allowed to restore an inoperable snubber. Maintaining the affected system unisolated during the system restoration time will allow the system to function as designed if an accident were to occur during this short restoration time. If the system were isolated, it would not be able to function. Furthermore, less time is now provided to restore the snubber if it affects both trains of a safety related system. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, there is a finding of "no significant hazards consideration."

NRC-94-079

WPSC (414) 433-1598
TELECOPIER (414) 433-5544

WISCONSIN PUBLIC SERVICE CORPORATION

June 28, 1994

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Response to Generic Letter 88-20, Supplement 4,
Individual Plant Examination for External Events

References:

1) Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events" (IPEEE) For Severe Accident Vulnerabilities - 10CFR50.54(f), dated June 28, 1991

2) Letter from C. R. Steinhardt (WPSC) to NRC Document Control Desk, dated December 20, 1991

3) Letter from C. R. Steinhardt (WPSC) to NRC Document Control Desk, dated September 18, 1992

Generic Letter 88-20, Supplement 4 (reference 1), requested that all licensees perform Individual Plant Examinations of External Events (IPEEE). Wisconsin Public Service Corporation (WPSC) submitted the response to Generic Letter 88-20, Supplement 4 for the Kewaunee Nuclear Power Plant in letters dated December 20, 1991, and September 18, 1992 (references 2 and 3). In reference 3, WPSC stated that the final results of the IPEEE for the Kewaunee Nuclear Power Plant would be submitted to the NRC by June 28, 1994. The enclosure with this letter provides the IPEEE submittal for Kewaunee.

The WPSC IPEEE submittal follows the format recommended in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." The report describes plant improvements already implemented, in the process of being implemented, or being considered as a result of the

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Document Control Desk
June 28, 1994
Page 2

Kewaunee IPEEE project. This submittal completes WPSC's response to Generic Letter 88-20, Supplement 4, for the IPEEE. If there are any questions regarding this report, please contact a member of my staff.

Sincerely,

C.A. Schrock for

C. R. Steinhardt
Senior Vice President - Nuclear Power

FBS/cjt

Attach.

cc - US NRC Region III
US NRC Senior Resident Inspector

Subscribed and Sworn to
Before Me This 28th Day
of June 1994

Notary Public, State of Wisconsin

My Commission Expires:
June 18, 1995

LIC\NRC\20 WP

ATTACHMENT 1

Letter From C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

June 28, 1994

KEWAUNEE NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS SUMMARY REPORT

June 28, 1994

WISCONSIN PUBLIC SERVICE CORPORATION

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS SUMMARY REPORT

TABLE OF CONTENTS

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS SUMMARY REPORT

TABLE OF CONTENTS (Cont'd)

SECTION PAGE

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS SUMMARY REPORT

TABLE OF CONTENTS (Cont'd)

SECTION PAGE

1. Executive Summary

In November 1988, the U.S. Nuclear Regulatory Commission (NRC) staff issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," which established a formal request for utilities to perform an Individual Plant Examination (IPE). In addition to the performance of the IPE, this letter requested utilities to identify potential improvements to address the important contributors to plant risk and implement improvements that they believed were appropriate for their plant.

In June 1991, the NRC issued Supplement 4 to Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)," which was accompanied by NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," which provided guidance for the information to be submitted to the NRC.

This report provides the requested information for the Kewaunee Nuclear Power Plant regarding external events, excluding internal flooding. The internal flooding analysis was included in the Kewaunee IPE.

1.1 Background and Objectives

In its Severe Accident Policy Statement (50FR43621), issued in 1985, the NRC concluded that operating nuclear plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be removed with low-cost improvements. As a result, the Commission issued Generic Letter 88-20 in 1988, requesting that each licensee conduct an IPE for internally initiated events, including internal flooding.

In December 1987, an External Events Steering Group (EESG) was established by the NRC to make recommendations regarding the scope, methods and coordination of the IPEEE. Ultimately, Supplement 4 to Generic Letter 88-20 was issued regarding external events.

The objectives of the IPEEE, as outlined in NUREG-1407, are:

1. To satisfy the requirements of GL 88-20, Supplement 4.

2. To develop an appreciation of severe accident behavior.

3. To understand the most likely severe accident sequences that could occur under full power operating conditions.

4. To gain a qualitative understanding of the overall likelihood of core damage and fission product releases.

5. To identify potential improvements in the plant design and/or operation that will reduce the overall core damage frequency and/or the containment failure frequency.

Wisconsin Public Service Corporation (WPSC) has completed and documented the IPEEE for Kewaunee, which meets these objectives. This report, containing a summary of the methods, results, and conclusions, provides our response to the NRC request for information contained in Generic Letter 88-20, Supplement 4. In addition, the IPEEE was conducted according to the applicable sections of 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." WPSC has retained all supporting analyses, descriptions and files pertaining to the IPEEE. These are available at WPSC offices for NRC review as necessary.

1.2 Plant Familiarization

The WPSC IPEEE program for Kewaunee involved an extensive plant familiarization effort because the undertaking of a full-scope realistic IPEEE requires careful analysis of the as-built, as-operated plant. To the extent possible, information gained during the internal events IPE for Kewaunee is used for the IPEEE. Nevertheless, additional walkdowns of the plant were performed and documented for the seismic, fire and other external events analyses.

The Kewaunee Nuclear Power Plant is a 2-loop pressurized water reactor licensed at 1650 MW (thermal). It is located in Kewaunee County, Wisconsin, along Lake Michigan's western shoreline and is jointly owned by WPSC, Wisconsin Power and Light Company, and Madison Gas and Electric Company. Kewaunee is the only nuclear power plant operated by WPSC. The nuclear steam supply system was supplied by Westinghouse Electric Corporation as was the turbine-generator, which is rated at 535 MW (net electrical). The architect/engineer was Pioneer Service and Engineering. The operating license was granted on December 21, 1973. Initial criticality was achieved on March 7, 1974. Initial power generation was reached April 8, 1974. and the plant was declared commercial on June 16, 1974. As of May 31, 1994, Kewaunee has operated with an availability factor of 84.4%.

The following is a summary of some of the important design features at the Kewaunee plant.

1. High Pressure Injection

   - Two centrifugal safety injection (SI) pumps deliver flow if Reactor Coolant System (RCS) Pressure is less than 2200 psig.

   - Two SI accumulators each contain 1250 ft^3 of borated water and are ready to inject if reactor coolant system (RCS) pressure is less than 700 psig.

   - SI pumps require support from the component cooling water and service water systems.

2. Low Pressure Injection

- Two residual heat removal (RHR) pumps deliver approximately 2000 gpm each when the RCS is depressurized.
- RHR heat exchangers downstream of each pump provide recirculation heat removal.
- Recirculation mode takes suction from containment sump B and discharges to the RCS, SI pump suction, and/or containment spray pump suction.
- RHR pumps and heat exchangers require support from the component cooling water system.
- RHR pump fan coil units are supplied by service water.

3. Auxiliary Feedwater

- Two motor-driven and one turbine-driven auxiliary feedwater (AFW) pumps supply cooling water to the steam generators in the event of a loss of main feedwater. Each pump and associated lube oil is cooled by the fluid being pumped.
- Pumps take suction through a single supply header from the condensate storage tanks.
- An alternate supply of water to the AFW pumps is provided by the Service Water (SW) System. AFW pump A is supplied by service water train A. AFW pump B is supplied by service water train B. The turbine driven AFW pump can be supplied by either service water train. Each pump is equipped with a low discharge pressure trip, so that if its normal source, the condensate storage tank, is lost, the pump is not damaged but immediately trips.

4. Emergency Power System

- Two 4160VAC buses feed two 480VAC buses each.
- Two diesel generators provide power to the 4160VAC buses should off-site power become unavailable.
- DC power is provided by four 8 hour station batteries and four battery chargers (2 vital and 2 non-vital).
- Vital instrument power is provided to four instrument buses from the 480VAC buses by way of 480VAC/120VAC instrument bus transformers, or from the vital DC system by way of four instrument bus inverters.

5. Component Cooling

- Consists of two pumps, two heat exchangers and one surge tank.
- Cools reactor coolant pumps (RXCPs), RHR pumps, and SI pumps.
- Component cooling heat exchangers are cooled by service water.

6. Service Water

- Consists of two normally cross-connected headers with two pumps in each header. These two headers are isolated from each other by a SI signal and thereby made separate and independent.
- Cools component cooling heat exchangers, containment fan coil units, SI pump lube oil heat exchanger and stuffing box, diesel generator coolers, and safeguard fan coil units, and provides the emergency water supply to the AFW pumps.

7. Containment

- Containment is of the Westinghouse large dry type.
- Primary containment consists of a low leakage steel vessel.
- Secondary containment consists of a medium leakage concrete shield building surrounding the primary containment vessel.
- Containment vessel free volume is 1.32×10^6 cubic feet.
- The containment vessel design pressure is 46 psig and design temperature is 268°F.

8. Containment Spray

- Two independent spray headers with one pump in each header deliver 1300 gpm each.
- Water is supplied by the refueling water storage tank (RWST) and the sodium hydroxide tank initially. When the RWST is depleted recirculated fluid can be supplied to internal containment spray (ICS) pump suction from the RHR pumps.

9. Containment Fan Coil Units

- Four containment fan coil units, two supplied by each service water header, cool the containment during normal and accident conditions.
- Service water is supplied at maximum flow during accident conditions.

10. Chemical and Volume Control

- Three positive displacement/air cooled charging pumps provide 60.5 gpm each for RCS makeup and RXCP seal injection.

- One of the three charging pumps is provided with a variable DC drive for speed control and is therefore not dependent on instrument air for attaining maximum pump output.

- Two boric acid transfer pumps provide the capability of supplying concentrated boric acid to the suction of the charging pumps for reactivity control.

1.3 Overall Methodology

In the IPEEE, standard systems analysis and external event assessment practices as outlined in NUREG-1407 are used. Seismic, internal fires and other external events (e.g., high winds, floods, etc.) are analyzed in the IPEEE using the following methodologies:

Seismic: The seismic IPEEE is a Level 1 effort with a qualitative and quantitative containment performance analyses. A seismic PRA (SPRA) approach using guidance described in NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150" is used for Kewaunee. Both Electric Power Research Institute (EPRI) seismic hazard curves and hazard curves developed by the Lawrence Livermore National Laboratory (LLNL) are used in the analysis. The seismic accident event trees and plant system models are taken from the internal events IPE and modified as necessary for seismic events. The Westinghouse WLINK computer code is used for fault tree quantification. The J. R. Benjamin SHIP computer code is used to perform the core melt quantification and the plant damage state quantification. The plant damage state quantification serves as a link between the Level 1 SPRA and the containment performance analysis.

Internal Fires: The internal fires analysis of the IPEEE is performed for Kewaunee using a Level 1 PRA and a qualitative and quantitative containment performance evaluation. A screening study based on the plant walkdowns and the EPRI Fire-induced Vulnerability Examination (FIVE) Methodology is used to screen out the less important fire area, while a full PRA is performed for the remaining areas. This analysis is a new fire PRA and follows the guidance identified in NUREG-1407. The deficiencies of past fire PRAs identified in NUREG/CR-5088 "Fire Risk Scoping Study" are addressed in the Kewaunee Fire PRA. The Westinghouse WLINK code is used for fault tree and core melt quantification.

Other External Events: The other external events analysis of the IPEEE use a screening approach that meets the intent of that described in NUREG-1407.

Sensitivity studies were performed on the model to determine the variability in the core damage frequency as influenced by such factors as changes in the cutoffs, operator actions, random failures, etc. Importance analyses were performed to identify the important accident sequences, system failures, component failures, and operator errors that contribute to the core damage frequency. Detailed notebooks were developed for each section of the Kewaunee IPEEE to provide documentation of the decisions and assumptions that served as input to the models and calculations.

One of the important steps in the Kewaunee IPEEE project was the reviews performed by the WPSC PRA staff, independent WPSC reviewers, and independent external reviewers. The PRA group members thoroughly reviewed the results of every iteration of the core melt quantification using their operations background to identify invalid cutsets. The models were reviewed to identify the problem or problems that caused the invalid cutset, and then the problems were corrected. An independent group of experienced Kewaunee plant staff members performed an extensive review of the different sections of the Kewaunee IPEEE, and identified numerous improvements that were then incorporated into the IPEEE.

1.4 Summary of Major Findings

This section summarizes the major findings of the Kewaunee IPEEE. First, the results of the core damage frequency quantification for each initiator are presented. Second, the dominant contributors leading to core damage for significant initiating events are described. Detailed discussion of these events can be found in their respective sections of this report.

The Kewaunee IPEEE does not deviate from the plant abnormal and emergency procedures. Therefore, there are no accident management actions modeled. By not taking credit for these actions, the results provided in this report are higher than they would be had credit been taken. The overall core damage frequency (CDF) for the Kewaunee Plant, considering both internal and external events is calculated to be 2.0E-04/year. This reflects the conservative approach taken by WPSC. Figure 1-1 provides a summary of the contribution to the overall core damage frequency by the different events. It is conservatively assumed that the core melt frequency due to external events other than fire and seismic events is at the screening value (1.0E-06/year). Figure 1-2 shows the Level 2 overall results by release categories. These release categories are defined in Table 1-1. The internal events portion of these figures represents not the IPE as submitted, but the present state of the Kewaunee internal events PRA which includes some revisions.

Seismic

Seismic Core Damage

In general, no significant seismic concerns were discovered in the SPRA. The CDF based upon the EPRI seismic hazard curve is 1.10E-05/year, whereas core damage frequency based upon the 1993 LLNL seismic hazard curve is 1.15E-05/year. Rankings of the dominant contributors to seismic CDF remain the same regardless of which seismic hazard curve is used.

The following are the dominant basic events/component failures that contribute to seismic risk:

- Loss of off-site power,
- Surrogate component in the following systems: Containment or Steam Generator failure, Reactor Vessel or Building failures, SW, DC Power, and AC Power,
- Operator Error - failure to shift AFW pumps from the condensate storage tanks (CST) to service water.

Kewaunee Seismic Core Damage Frequency Results - Approximately 93 percent of the CDF is determined by 6 accident sequences. A discussion of the dominant sequences is provided below.

Sequence SCSF - This sequence corresponds to failure of the containment structure or the SG. Failure of either of these components leads directly to core damage. These structures were screened using the first column in EPRI NP-6041, which corresponds approximately to a PGA HCLPF level of 0.30g. As a result, these structures are modeled by the surrogate component. The seismic CDF associated with this sequence is 3.6E-06 per year. This is conservative, since the surrogate component is a conservative representation of the capacity of components at the Kewaunee plant.

Sequence SSWS - This sequence corresponds to failure of the SW System. Failure of the SW System is dominated by failure of the intake structure, which is modeled using the surrogate component. The intake structure was screened based on a HCLPF level of 0.30g. All other components in the SW System have median capacities of 0.66g PGA or greater. The seismic risk associated with failure of the SW System is 1.99E-06 per year which is conservative, since the capacity of the SW System is dominated by the surrogate component.

Sequence SRVB - This sequence corresponds to failure of the screenhouse, the auxiliary building, turbine building or the reactor vessel. Failure of any of these components is assumed to lead directly to core damage. These structures were screened using the middle column in EPRI NP-6041, which corresponds approximately to a PGA HCLPF level of 0.30g.

As a result, these structures are modeled by the surrogate component. The seismic risk associated with this sequence is 1.74E-06 per year. This is conservative since the surrogate component is a conservative representation of the capacity of the structures at the Kewaunee plant.

Sequence SLSP01 - This sequence corresponds to the loss of off-site power and failure of the AFW System. Failure of the AFW System is attributed to failure of the operator to shift AFW pumps from the CST to SW and failure of the surrogate element. The seismic risk associated with this sequence is 1.35E-06 per year.

Sequence SACP - This sequence corresponds to failure of emergency AC power system, including the diesel generators, and supporting mechanical and electrical equipment. All components in the AC power system have median capacities of 1.86g PGA or greater. As a result, failure of AC power is dominated by failure of the surrogate component. The seismic risk associated with failure of the AC power system is 1.26E-06 per year. This is conservative, since the system capacity is dominated by the surrogate element.

Sequence SDCP - This sequence corresponds to failure of DC power system, including failure of the station batteries, battery chargers, cable trays and electrical support equipment. All components in the DC power system have median capacities of 1.10g PGA or greater. As a result, failure of DC power is dominated by failure of the surrogate component. The seismic risk associated with failure of the DC power system is 3.48E-07 per year. This is conservative, since the system capacity is dominated by the surrogate element.

In the Kewaunee SPRA, small, medium and large LOCA events are modeled. As part of the Kewaunee SPRA, reactor vessel failure and seismically initiated small, medium and large LOCA events are considered. The frequency of these events is:

LOCA	Size	Frequency
Small	1.5 - 3"	2.23E-07/Yr
Medium	3 - 6"	7.68E-09/Yr
Large	> 6"	1.77E-07/Yr
Reactor Vessel Failure		1.74E-06/Yr

Contribution of Ground Motions to Plant Risk - Approximately 76 percent of the CDF is contributed by PGA values in the range 0.25g to 0.65g.

Summary

This section summarizes the results of the Kewaunee SPRA. The seismic CDF is 1.10E-05 which is considerably less than the internal events CDF (8.73E-05). The median capacity of the plant is 0.38g peak ground acceleration (PGA), which is greater than a factor three times that of the safe shutdown earthquake (SSE). The high confidence of a low probability of failure (HCLPF) acceleration for Kewaunee is 0.26g PGA or approximately two times the SSE. The HCLPF is based on an evaluation of seismic systems model for seismic failures only.

Based on sensitivity evaluations and a review of the SPRA results, the following conclusions are made:

1. There does not exist a single failure mode for Kewaunee that dominates the seismic CDF.

2. Failure of a surrogate component, which is a conservative measure of the capacity of components that are screened out, is for many systems the important mode of failure. Since the surrogate does not specifically model the failure of a particular component, this observation is a further reinforcement of the conclusion that there does not exist any component specific failure modes that dominate the seismic CDF.

3. Operator actions are not a major contributor to the seismic CDF or plant capacity.

4. Loss of off-site power is an important contributor to the seismic risk.

5. As a group, random failures and operator actions are an important part of the seismic CDF. In a relative sense, variation in the random failure probabilities produced the largest change in the seismic CDF (a range corresponding to a factor of 2.5).

Seismic Containment Performance

As part of the seismic containment walkdowns, containment mechanical penetrations and the containment isolation valves were analyzed for the ability to withstand seismic events. The penetrations and isolation valves from both inside and outside of containment were analyzed. Based upon these plant walkdowns, no significant seismic hazards were found to exist and it was determined that these components possess a high capability to withstand seismic events.

The seismic containment failure frequency is 6.24E-06 which is about the same as the containment failure frequency due to internal events (8.03E-06). The seismic containment failure frequency is 57% of the seismic core damage frequency. The containment failure median capacity and HCLPF for Kewaunee are 0.51 and 0.30g PGA, respectively. The median capacity is about four times the SSE and the HCLPF is two and a half times the SSE. The HCLPF is based on an evaluation of the seismic containment systems model that considers both random and seismic failures. No additional Level 2 vulnerabilities were discovered.

The results of the evaluations performed indicated that the containment as well as the systems designed to ensure containment integrity are seismically sound and no vulnerabilities could be identified.

Fire

Fire Core Damage

In general, no significant fire concerns were discovered in the Fire PRA. The core damage frequency due to fire is 9.8E-05/year. This is dominated by fires in the A and B auxiliary feedwater (AFW) pump rooms, which contribute 84% to the total fire core melt frequency. These areas each contain cabling for one train of safe shutdown equipment and cabling for numerous non-safety related equipment, such as transformers supplying offsite power. Kewaunee meets all the requirements of 10 CFR 50, Appendix R, (other than exemptions approved by the NRC) and an additional equipment failure or human error in addition to the fire is necessary for core melt to occur.

Due to manual actions necessary to respond to a fire, human error is an important contributor to fire core damage frequency, contributing 56% (based on Fussel Vesely importance) to the fire core damage frequency. Of these, the largest contributor is failure to locally establish power to the dedicated (A) train of safe shutdown equipment, contributing 17% to the fire core damage frequency.

Following is a description of each of the thirteen dominant sequences, which represent 99.95% of the total core melt frequency.

Sequence #1 - AFW pump A oil fire followed by failures of AFW and bleed and feed. This results in a loss of heat sink and early core damage. The primary means by which heat sink is lost is a station blackout. Since offsite power and the A diesel generator are unavailable as a result of the fire, a failure of the B diesel generator or the B train of service water, which cools the diesel generator, results in a station blackout. Since the fire also disables control cabling for the turbine driven AFW pump, there is no heat sink and therefore core melt occurs. The primary human errors in this sequence are failure to manually establish electrical power from the B diesel generator and failure to locally restore instrument air for the pressurizer PORVs, used for bleed and feed. The frequency of this sequence is 4.21E-05 per year.

Sequence #2 - AFW pump B oil fire followed by failure of AFW. This results in a loss of heat sink and early core damage. Other means of removing heat, (main feedwater, condensate, bleed and feed) are unavailable due to the unavailability of offsite power and safety injection pumps from the dedicated shutdown panel (DSP). This sequence is dominated by human error. Failure to establish electrical power from the A diesel generator locally and failure to establish service water and AFW from the DSP are major contributors to this sequence. The frequency of this sequence is 2.91E-05 per year.

Sequence #3 - Oil fire in A AFW pump followed by failure of CCW. This results in a loss of cooling to the reactor coolant pump (RXCP) thermal barrier, which can result in a small LOCA due to RXCP seal failure. Since the safety injection (SI) and residual heat removal (RHR) pumps are cooled by component cooling water (CCW) and charging is not available because of cabling damaged by the fire, there is no way to get makeup flow to the RCS to replace that lost through the RXCP seals and late core damage results. The only human error in this sequence is the failure to manually restart CCW and this is a relatively minor contributor. The frequency of this sequence is 9.65E-06 per year.

Sequence #4 - Fire near motor control center (MCC) 62A in B diesel generator room followed by a failure of charging and CCW. This results in a loss of cooling to the RXCP seals and late core damage. The primary means of loss charging and CCW is loss of power to the emergency buses, buses 5 and 6. Human error is not important in this sequence. The frequency of this sequence is 4.50E-06 per year.

Sequence #5 - Oil fire in B diesel generator followed by a failure of charging and CCW. This results in a loss of cooling to the RXCP seals and late core damage. The primary contributor is a mechanical failure of CCW and operator error in establishing charging flow. The frequency of this sequence is 3.56E-06 per year.

Sequence #6 - Fire near safeguards 480V electrical buses 51 and 52 followed by a failure of CCW. This results in a loss of cooling to the RXCP seals and late core damage. Human errors in this sequence include failure to manually establish on-site power, service water and CCW. The frequency of this sequence is 3.13E-06 per year.

Sequence #7 - Oil fire in B diesel generator followed by failure of AFW and bleed and feed. This results in a loss of heat sink and early core damage. All B train equipment, powered by safeguards 4160V electrical bus 6, is unavailable due to the fire. Therefore, the primary means of failure is mechanical failure of the A motor driven and turbine driven auxiliary feedwater

(AFW) pumps followed by failures of the SI, CCW or Instrument Air Systems, all of which are needed for bleed and feed. Another means of failure is a failure of DC bus BRA-104, which is needed to start both the A motor driven and turbine driven AFW pumps, and the A SI pump. Human error, failure to establish bleed and feed, is a minor contributor to this sequence. The frequency of this sequence is 2.18E-06 per year.

Sequence #8 - Oil fire in B diesel generator followed by failure of AFW, success of bleed and feed, and failure of high pressure recirculation. This results in a loss of heat sink and late core damage. All B train equipment, powered by bus 6, is unavailable due to the fire. Therefore, the primary means of failure is mechanical failure of the A motor driven and turbine driven AFW pumps followed by failures of valves associated with switchover to containment sump recirculation. Human error, in this case failure to stop the A RHR pump to limit its time in miniflow, is a minor contributor to this sequence. The frequency of this sequence is 1.70E-06 per year.

Sequence #9 - Oil fire in A AFW pump followed by failure of AFW, success of bleed and feed, and failure of high pressure recirculation. This results in a loss of heat sink and late core damage. Since the fire destroys both the A AFW pump and control cabling for the turbine driven pump, the primary means of failure is a mechanical failure of the B AFW pump followed by failure of valves associated with switchover to containment sump recirculation. Human error is not important in this sequence. The frequency of this sequence is 9.83E-07 per year.

Sequence #10 - Oil fire in B AFW pump followed by failure of charging and CCW. This results in a loss of cooling to the RXCP seals and late core damage. Both CCW and charging must be established from the dedicated shutdown panel (DSP). This sequence is dominated by the combination of failure to establish CCW and failure to establish charging. In this case, moderate dependency is applied. The frequency of this sequence is 5.76E-07 per year.

Sequence #11 - Fire in the relay room followed by failure of AFW. This results in a loss of heat sink and early core damage. The dominant contributor to this sequence is human error. The A diesel generator must be started locally, and service water and either charging or CCW must be started from the DSP. The frequency of this sequence is 3.14E-07 per year.

Sequence #12 - Fire near buses 51 and 52 followed by failure of AFW and bleed and feed. This results in a loss of heat sink and early core damage. Since AFW, SI, and instrument air are all started locally, human error is the dominant contributor to this sequence. Of these human errors, the largest contributor is failure to establish AFW and instrument air. These are not modeled as dependent actions because AFW is started from the control room and the instrument air compressor is simultaneously started locally. The frequency of this sequence is 1.79E-07 per year.

Sequence #13 - Fire near MCC 62A in B diesel generator room followed by failure of AFW, main feedwater, and bleed and feed. This sequence consists mostly of mechanical failures of AFW followed by the operator failing to stop the RXCPs, in order to prevent the additional heat input into the Reactor Coolant System. The frequency of this sequence is 1.48E-07 per year.

Fire Containment Performance

Due to the robust design of Kewaunee's large dry containment, no containment failures due to overpressurization are produced by any fire-initiated core damage sequence within the 48 hour containment mission time. The containment failure frequency of 3.64E-05/year is due exclusively to failure to isolate containment. No additional containment failure modes unique to internal fires were identified.

Other External Events

This analysis examines all credible external events other than seismic events, internal floods, or internal fires. Specifically examined in the other external events analysis are external flooding, aircraft accidents, severe winds, ship impact accidents, off-site and on-site hazardous materials accidents, and external fires. No vulnerabilities are identified that require detailed quantification of any accident events. It is therefore concluded that the effects from any of the other external events described here are not a significant concern at Kewaunee. The results and conclusions are presented in summary as follows.

High Winds and Tornadoes

Due to the low frequency of high winds and tornadoes at the Kewaunee site, it is concluded that the contribution to plant risk from severe wind events is insignificant.

The severe wind protective measures and design features instituted at Kewaunee are consistent with a highly safe plant design with very low risk contribution. Therefore, no design changes are currently recommended to protect the plant from severe winds.

External Floods

In view of the low frequencies and maximum flood levels, as well as the plant elevation, the topographical layout of the site, and the elevation of plant penetrations relative to safety-related equipment, it is concluded that the contribution to plant risk from external flooding is greatly dominated by other risk contributors.

An analysis was also performed to evaluate flooding based on the probable maximum precipitation (PMP) criterion. It is concluded that Kewaunee is not endangered by the flooding based on the PMP criteria.

In fact, in 1993 when flood levels across the Midwest were at all time highs, there were no problems experienced at Kewaunee.

The external flooding protective measures and design features instituted at Kewaunee are consistent with a highly safe plant design with very low risk contribution. Therefore, no design changes related to external flooding prevention are currently recommended.

Transportation and Nearby Facility Accidents

Based on the location of Kewaunee with respect to the major roads, rail transportation, air traffic and nearby industrial facilities, there is no threat to the plant safety from any of these sources.

Unless flight patterns change, the air or ground traffic significantly increases, or any new industrial facility is opened in the vicinity of the plant, no design changes in plant are recommended due to these events.

Hazardous Materials

This analysis began with a review of the Updated Control Room Habitability Report (Reference 34), which was completed in 1989. This report was the result of a study performed in response to NUREG-0737 and includes an assessment of hazardous materials on-site as well as off-site.

A plant walkdown and a review of plant records was performed to verify that the control room habitability study assumptions were still valid. It was determined that the results and conclusions stated in the study are still valid, and that there is no threat to control room personnel from hazardous spills or releases.

The analysis was further expanded to consider the effects of a release of hazardous materials on safety-related equipment or the local operation of plant systems during emergencies. It was determined that a release of hazardous material would have no effect on safety related equipment, and furthermore, no hazardous materials were located near safety-related equipment. It was further determined that there are no credible hazardous material releases that would prevent an operator from locally operating plant equipment during plant emergencies.

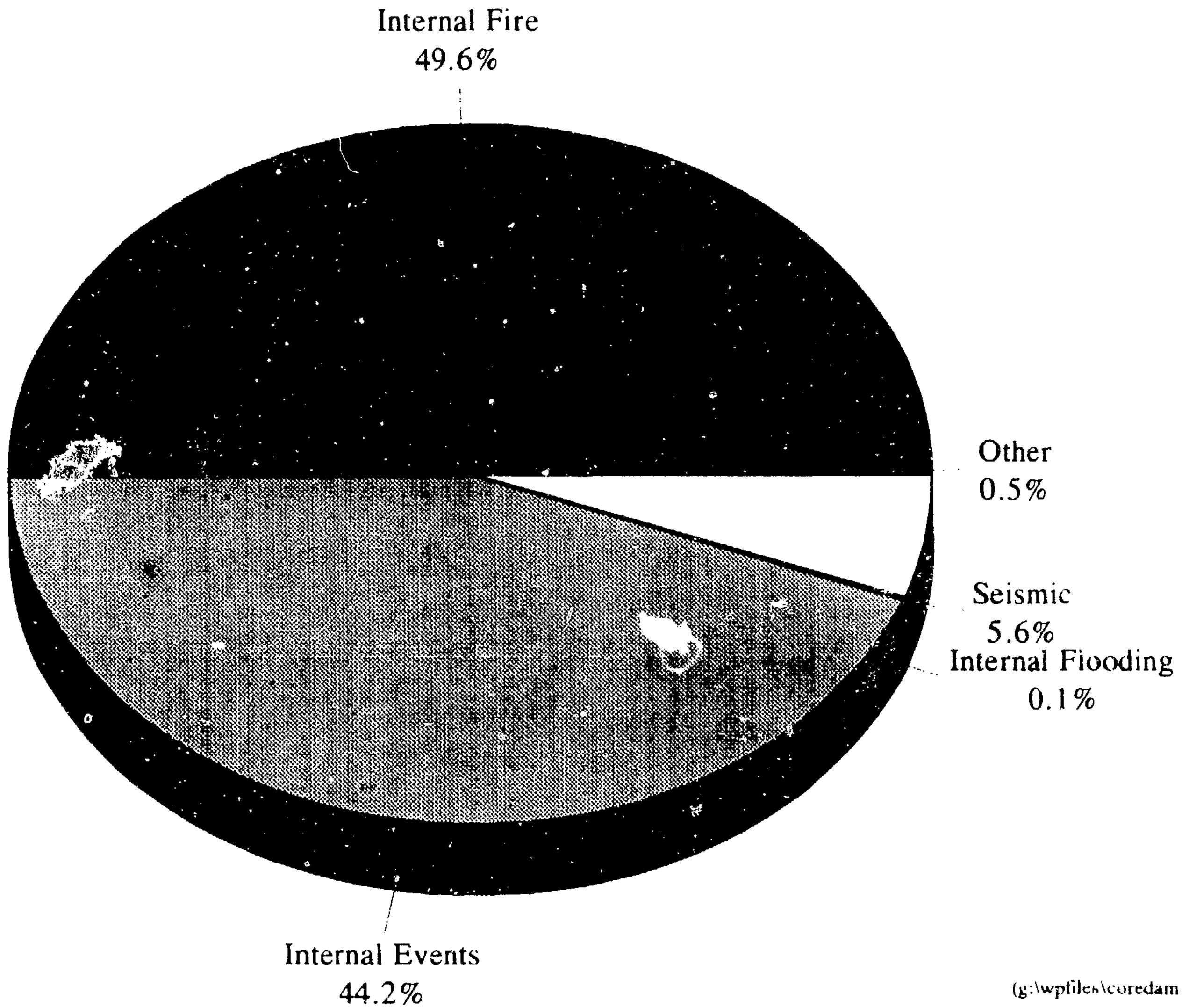
TABLE 1-1

RELEASE CATEGORY DEFINITIONS

Release Category	Definition
A	No containment failure occurs with 48 hour mission time but failure could eventually occur without accident management action; noble gases and less than 0.1% volatiles released.
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired).
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful).
T	Containment bypassed with noble gases and more than 10% of the volitiles released.
U	Containment failure prior to vessel failure with noble gases and more than 10% of the volatiles released (containment isolation impaired).

FIGURE 1-1

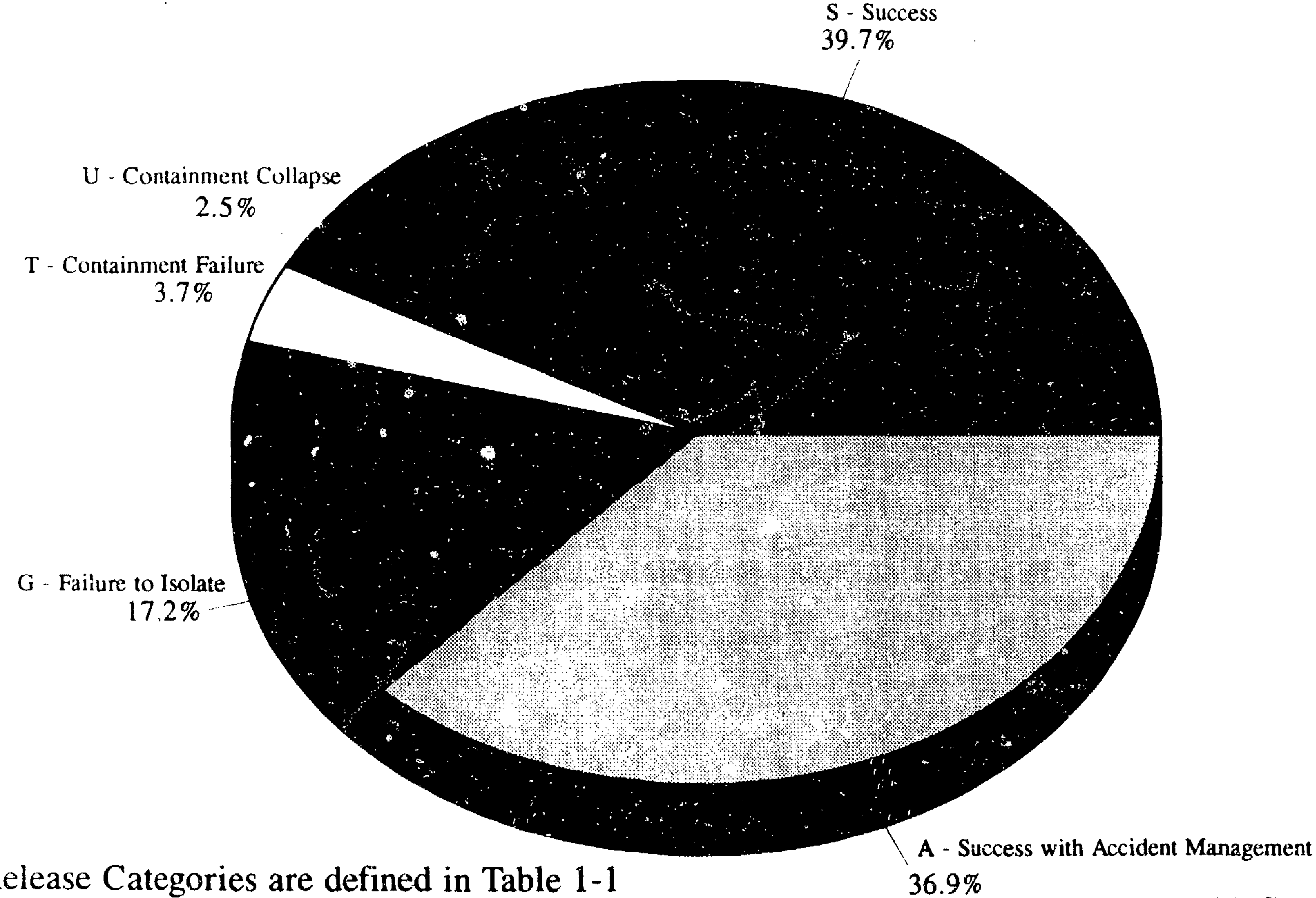
TOTAL CORE DAMAGE FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS



(g:\wpfiles\coredam1.prs)

FIGURE 1-2

RELEASE CATEGORIES FOR INTERNAL AND EXTERNAL EVENTS



Release Categories are defined in Table 1-1

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2. Examination Description

2.1 Introduction

The Kewaunee Nuclear Power Plant Individual Plant Examination for External Events (IPEEE) has been performed to identify and resolve plant specific severe accident issues stemming from external events.

Wisconsin Public Service Corporation (WPSC) has conducted the IPEEE in full compliance with the requirements of the NRC Generic Letter 88-20, Supplement 4. WPSC's approach to the IPEEE has been to perform realistic evaluations of Kewaunee's capabilities to respond to external events.

The Kewaunee External Events program consisted of the following major tasks:

Project Management
Data Collection and Analysis
Initiating Event Analysis
Event Tree Analysis
Systems Analysis
Human Reliability Analysis
Systems Interaction
Fault Tree and Accident Sequence Quantification
Sensitivity Analysis
Peer Reviews
Training and Technology Transfer

The Kewaunee IPEEE Containment Performance Analysis is a qualitative as well as a quantitative analysis analogous to the internal events Level 2 analysis as appropriate.

The models developed in the IPEEE are drawn from the internal events analysis and modified as necessary for external events. These models represent the as-built, as-operated Kewaunee Nuclear Power Plant. Efforts were taken to ensure that only formal procedures that the operators are trained to use have been credited.

2.2 Conformance With Generic Letter and Supporting Material

Generic Letter 88-20, Supplement 4, which was issued on July 22, 1992, requested each utility to perform an IPEEE for the purpose of:

(1) developing an appreciation of severe accident behavior,

(2) understanding the most likely severe accident sequences that could occur at its plant,

(3) gaining a more quantitative understanding of the overall probabilities of core damage and fission product releases, and if necessary,

(4) reducing the overall probabilities of core damage and fission product releases.

General requirements provided in the Generic Letter for fulfilling the stated purpose are:

(1) The utility staff should be used to the maximum extent possible in the performance of the IPEEE to insure that they:

- understand the plant procedures, design operation, maintenance and surveillance,
- understand the quantification/evaluation of the expected sequence frequencies,
- determine the leading contributors to core damage and unusually poor containment performance,
- identify proposed plant improvements for prevention and mitigation,
- examine each of the proposed improvements, and
- identify which proposed improvements will be implemented and their schedule.

(2) The method of examination should be as described (for each of the external events) in the Generic Letter using the guidance of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities."

(3) The utility should resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," as impacted by external events in the IPEEE.

(4) The utility should carefully examine the results of the IPEEE to determine if there are worthwhile prevention or mitigation measures that could be taken to reduce the frequency of core damage or improve containment performance.

(5) The utility should report the results of the IPEEE to the NRC consistent with the criteria provided in the Generic Letter and subsequent guidance provided in NUREG-1407.

(6) The utility should document the examination in a traceable manner and retain it for the duration of the license unless superseded.

In response to the Generic Letter, WPSC issued two letters dated December 20, 1991 and September 18, 1992 stating its intent to perform an IPEEE for Kewaunee in order to identify, evaluate, and resolve severe accident issues germane to the plant. The IPEEE addresses the requirements set forth in Supplement 4 to the Generic Letter.

WPSC has invested substantial personnel time in addition to financial resources for the efforts of contractors J. R. Benjamin & Associates, Westinghouse Electric Corporation and Stevenson & Associates in the performance of an IPEEE that meets or exceeds the NRC directives listed in Generic Letter 88-20, Supplement 4. A permanently assigned staff, knowledgeable in the design and operation of Kewaunee and the IPE, has been involved in all aspects of the IPEEE. Other WPSC personnel have been involved in various aspects of the evaluation as needed. In addition, steps were taken to insure that WPSC personnel who had a need for understanding of the evaluation or parts thereof developed an appreciation for the risk significance of the results and the plant response as well as an understanding of the bases of the IPEEE.

2.3 General Methodology

The Kewaunee IPEEE program, as previously identified, consisted of eleven major tasks. The IPEEE was conducted using standard systems analysis practices such as those mentioned in NUREG-1407. A comprehensive task breakdown was developed for the Kewaunee PRA in order to organize the work to be accomplished. An overview of each of the tasks is provided below. More specific information regarding each of the analyzed external events is found within the applicable sections of this report.

IPEEE Tasks

(1) Project Management - Development and monitoring of detailed project planning and scheduling provided necessary technical direction of project analyses and proper review of results.

(2) Data Collection and Analysis - Plant-specific information was collected through plant walkdowns, review of WPSC calculations and review of the history of external events at Kewaunee. This data was analyzed and formatted for input into the IPEEE.

(3) Initiating Events Analysis - The selection of accident initiating events for the Kewaunee IPEEE considered both actual plant data and results of previous studies and published NUREGs.

(4) Event Tree Analysis - Plant-specific event tree models were drawn from the internal events analysis and modified as necessary for external events. This task entailed reviewing accident progression as modeled within the internal event trees and modifying these event trees based upon equipment and operator availability following initiation of the external event.

(5) Systems Analyses - Similar to the event trees, the internal events system fault trees were modified as necessary to reflect plant system availability following initiation of the external event.

(6) Human Reliability Analysis - The human reliability analysis task established suitable models to represent the interaction of operators and other plant staff with plant systems and equipment during normal operation and during transient and accident conditions. For this task, those human tasks important to the analysis were identified, and the full range of plant procedures was examined to determine the types of human actions that are routinely performed and what kinds of actions operators are trained to take. All accident sequences and system failure modes developed in the event and fault tree models were

carefully evaluated to determine those areas where operator intervention can, should, and must occur. Finally, the kinds of errors in all identified human actions critical to the analysis of plant risk were assessed in the Kewaunee IPEEE.

After potentially important human errors were identified, detailed models were developed and were quantified so that their effects could be incorporated into the event and fault tree models. The Technique for Human Error Rate Prediction (THERP) methodology was used for the human reliability analysis.

(7) Systems Interaction - Possible system interactions due to external events were identified by conducting detailed system walkdowns.

(8) Fault Tree and Accident Sequence Quantification - The Kewaunee external events fault trees and event tree accident sequences were integrated and quantified to obtain accident sequence cutsets, frequencies for all accident sequences resulting in core damage, and to identify dominant accident sequences among all event tree results. The Westinghouse WLINK Code System was used to perform the initial fault tree quantification. The seismic IPEEE used the J. R. Benjamin SHIP code for system and accident sequence quantification. The internal fire analysis also employed the COMPBRN IIIe code to back up engineering evaluations. The other external events analysis employed a screening approach with engineering judgement as described in Generic Letter 88-20, Supplement 4.

(9) Sensitivity and Importance Analyses - The response of the core damage frequency to changes in input parameters and modeling assumptions was examined to identify important actions and equipment and to study the sensitivity to those assumptions.

(10) Review Program - In all stages of the Kewaunee PRA, numerous levels of review were performed to ensure accuracy and completeness. Extensive reviews were performed by the WPSC PRA staff, independent WPSC reviewers. The PRA group members thoroughly reviewed the results of every core melt quantification iteration using their operations background to identify invalid cutsets. The models were reviewed to identify the problem that caused the invalid cutset, and then the problems were corrected. A independent group of experienced Kewaunee plant staff members performed an extensive review of the different phases of the Kewaunee IPEEE, and identified numerous improvements that were made to the IPEEE.

(11) Training and Technology Transfer - Training was conducted by contractor employees for utility personnel to provide the in-house ability to understand, evaluate, modify, and update the IPEEE to reflect proposed or actual changes in the plant design, operation or to account for future industry updates impacting external event analyses.

2.4 Information Assembly

A tremendous amount of information was needed to perform the detailed Kewaunee IPEEE study. The project team reviewed and assembled information from plant specific sources, similar plant studies, and generic sources. Plant walkdowns were key to the data collection effort. Walkdowns were specifically used to search for plant external event vulnerabilities and to group data into specific areas. This data was ultimately used to determine important initiating events, quantify their frequency and determine component and system failure rates or provide information for various screening analyses.

The Kewaunee IPEEE team modeled the plant condition as it currently exists. All major changes to plant operation or design to date have been identified and included in the PRA results. All information used in the project is available at the WPSC offices in Green Bay, Wisconsin. Copies of some information are also housed at the contractor offices previously mentioned.

Detailed IPEEE project notebooks were developed for seismic events, internal fires and other external events (external floods, winds, etc.). Information sources used to develop the IPEEE models are identified in the individual sections of this report. Both plant specific and generic sources identified were used to define component availabilities, initiating events and initiating event frequency, important accident sequences and potentially important modeling features. Subsequent sections of this report provide a more detailed discussion of the use of the information collected.

Plant walkdowns were conducted by IPE team members and contracted personnel who were responsible for the evaluation of specific external events. WPSC IPEEE analysts accompanied the walkdown team members so as to observe first hand any identified plant vulnerabilities and take part in the IPEEE from start-to-finish.

Walkdowns were conducted for the systems and plant environment of most concern to the IPEEE. These areas are contained primarily in the auxiliary building and the containment. Several other buildings or areas were examined, however, because important systems and components are located therein. The areas or buildings in which walkdowns were made are:

- Containment
- Auxiliary Building
- Turbine Building Basement
- Relay Room
- Screenhouse
- Battery Rooms
- Technical Support Center

- Control Room
- Administration Building Basement
- Outside Grounds Including Switchyards

General arrangement drawings of these areas are contained in the USAR. The individual external event walkdowns are described in more detail in subsequent sections of this report.

3. Seismic Analysis

SECTION 3

SEISMIC ANALYSIS

TABLE OF CONTENTS

SECTION **PAGE**

SECTION 3

SEISMIC ANALYSIS

LIST OF TABLES

SECTION 3

SEISMIC ANALYSIS

LIST OF FIGURES

SECTION 3

SEISMIC ANALYSIS

LIST OF FIGURES (Cont'd)

3.0 Methodology Selection

The methodology applied in the Kewaunee Nuclear Power Plant seismic PRA (SPRA) is consistent with the methodology outlined in NUREG-1407 (Reference 1). A SPRA has not been previously performed for Kewaunee. Thus this is a new PRA study.

3.1 Seismic PRA

The Kewaunee SPRA was performed using a methodology that allowed for the incorporation of the analysis that was performed in the Kewaunee Individual Plant Examination (IPE). Using such a methodology was beneficial for two reasons: first, since the basic structure of the internal events fault trees and event trees could be carried over into the seismic analysis, this saved the effort of creating all new logic models. Therefore, it was only necessary to make modifications to existing models. Second, by using the same logic model structure, for both the IPE and SPRA, the core damage contribution results could be compared on a common basis. Since the SPRA logic models were built by making modifications to the internal events logic models, the event tree and fault tree sections of this report only describe the modifications required to convert the IPE logic models into the SPRA logic models. The IPE submittal (Reference 2) provides the basis and description of the fault tree and event tree logic models.

The methodology used for the SPRA analysis is summarized below:

Hazard Analysis

The seismic hazard curve is a description of the probability that during a given period of time, one or more earthquakes will occur that results in a specific peak ground acceleration (PGA) level. As recommended in Generic Letter 88-20, Supplement 4, the site specific seismic hazard curves generated by both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) are analyzed in the Kewaunee SPRA. This is discussed further in Section 3.1.1.

Plant Walkdown

The Kewaunee SPRA plant walkdowns were conducted to take advantage of the overlapping requirements between the IPEEE and Unresolved Safety Issue (USI) A-46 examination programs. All equipment were treated as if they were USI A-46 items, even if they were designated as SPRA items with specific fragility levels assigned. Seismic walkdowns were also conducted to assess the primary site structures. The plant walkdowns are discussed in detail in Section 3.1.2.

Fragility Analysis

In the SPRA, three types of basic events are considered in the plant logic model:

- component (i.e., structure or equipment item) failures that are initiated by a seismic event (seismic fragility)
- random (non-seismic) failures, and
- operator errors that occur following a seismic event.

Each basic event type is described below.

Seismic Fragility - The purpose of the seismic fragility analysis is to estimate the capacity of structures and equipment at Kewaunee in terms of peak ground acceleration (PGA). The seismic fragility of an individual component (structure or equipment item) is defined as the conditional probability of failure for levels of PGA. The assessment of the seismic fragility of a component is a function of its seismic design basis, safe shutdown earthquake (SSE), the factors of safety incorporated in the design process and the variability in earthquake ground motion and factors that influence structure response. As a result of these variabilities, the capacity of a component to withstand earthquake loading, as defined in terms of PGA, is a random variable. For a specified PGA level (a) the fragility (conditional probability of failure) of a component is:

$$f = P[A = a] \qquad (3\text{-}1)$$

The fragility of a component is modeled by a lognormal distribution. The lognormal distribution is defined by two parameters, the median capacity, A_m, and logarithmic standard deviation, β. The conditional probability of failure is calculated as:

$$f = \Phi\left[\frac{\ell n(a/A_m)}{\beta_C}\right] \qquad (3\text{-}2)$$

where (Φ) is the standard normal cumulative distribution function. For each component incorporated in the SPRA the median capacity, A_m, and logarithmic standard deviation, β_c, are determined.

The variability in the seismic capacity of a component is quantified by the logarithmic standard deviation which is divided in two parts, randomness and uncertainty. The logarithmic standard deviation for randomness, β_R, quantifies the inherent or natural variation in material properties and earthquake ground motion. The uncertainty in the assessment of the seismic capacity of a component is attributed to limited data to determine the ground motion that causes failure and limitations of analytical methods. This logarithmic standard deviation for variability is denoted β_U.

The composite variability, β_C, is defined by:

$$\beta_C = \sqrt{\beta_R^2 + \beta_U^2} \tag{3-3}$$

In the Kewaunee SPRA the mean seismic fragility curve is used to quantify the plant risk. The mean conditional probability of failure is determined through the use of the median capacity, A_m, and the composite logarithmic standard deviation, β_C, in equation 3-2.

For a component, the ground motion (PGA) level for which there is a high confidence of a low probability of failure (HCLPF) is defined as:

$$HCLPF = A_m \exp(-1.65(\beta_R + \beta_U)) \tag{3-4}$$

The HCLPF provides a measure of the seismic integrity of a component that accounts for the uncertainty in its capacity and randomness in response.

Random (Non-Seismic) Failures - As part of the Kewaunee SPRA, random or non-seismically initiated failure of equipment is considered. These random events are the same as those used in the IPE plant model.

Operator Actions - During a non-seismically initiated accident, the potential for operator error is related to the level of stress associated with the accident and the complexity of the operations that must be performed. Given the occurrence of an earthquake, stress conditions are amplified, leading to the greater chance for operator error. In addition, depending on the level of ground motion, the operator must contend with the disruption that may hypothetically occur in the control room (i.e., falling ceiling tiles, overturned bookcases, items falling off desks) as well as damage to the plant that may make access to critical areas difficult. In the Kewaunee SPRA, the increased likelihood of operator error with increasing levels of ground motion is considered. Operator error rates are defined for specified ground motion intervals. For low ground motions that are not likely to cause major damage to the plant, the operator error rates are the same as those used in the IPE. These rates are then increased for specified ground motion intervals. This is discussed further in Section 3.1.5.3.

Seismically Induced Initiating Events

To model the performance of the plant following the occurrence of a seismic event, the potential structure and equipment failures that could initiate an accident are considered. These events are modeled in a seismic event tree and fault trees. To develop the logic structure for seismically initiated accidents the following analysis procedure is used:

1. The system analysis identified the structures and equipment items that may be required to determine the status of the plant and to bring the plant to and maintain hot shutdown following a seismic event.

2. For the items identified in the previous step, a screening evaluation is performed to determine which components may be vulnerable to seismically initiated failure. (The screening procedure is described in Section 3.1.4.)

3. A seismic event tree is developed to model the sequence of seismically initiated failures in the plant. The initiating event is the occurrence of earthquake ground motion at Kewaunee. The top events in the tree correspond to seismically initiated structure and equipment failures, front line and support system failures, etc.

4. For each plant system, a seismic fault tree is developed to model the seismic and random failures that could lead to system failure. Included in the system model are operator actions required to mitigate an accident.

5. For structures and equipment items modeled in the seismic event and fault trees, fragility parameters are determined.

6. For operator actions that are modeled, seismic failure probabilities are estimated. (The estimation technique is described in Section 3.1.5.3.)

Seismically initiated accidents included in the SPRA are:

- Reactor Vessel rupture
- Large (> 6") LOCA
- Medium (3" - 6") LOCA
- Small (1.5" - 3") LOCA
- Transients
- Loss of Off-Site Power
- Steam Line Break

The logic model developed in the above steps is quantified using the Seismic Hazard Integration Package (SHIP) code (Reference 3) to determine the plant seismic risk. The seismic risk quantification process is described in Section 3.1.5.

Seismic Event Trees

Seismic event trees are developed to model the event sequences that can follow the occurrence of an earthquake. The top events in the SPRA event trees define the frontline systems that need to be modeled in the seismic analysis and thus, the necessary support systems are defined by the frontline system requirements. Top events in the event tree include:

- Structure Failure
- Reactor Vessel Failure
- Support System Failure (i.e., electric power and cooling water)
- LOCAs
- Mitigation System Failure

A complete description of the methodology and assumptions used in the event tree development is found in Section 3.1.5.2.

Seismic Fault Trees

Seismic fault trees are developed using the Level 1 IPE analysis as the foundation for the fault trees in the SPRA. Once the systems needed for the SPRA analysis are defined, it is necessary to determine the components in these systems that could be adversely affected by seismic activity. The vulnerable components were identified during the seismic walkdown. A seismic fault tree is developed for each internal events fault tree that contains seismically vulnerable components. The seismic fault trees only contain the seismic failures of a system's components. After each system seismic fault tree is completed, it is linked with the internal events fault tree it supports. The resulting system fault tree cutset files contain both seismic and random failures. To simplify the system fault trees and more explicitly represent the intersystem dependencies, support systems such as AC power, DC power, and service water are removed from the system level analysis and placed into the sequence level analysis. A complete description of the methodology and modeling assumptions used in the fault tree development is found in Section 3.1.5.3.

3.1.1 Hazard Analysis

In the Kewaunee SPRA, the EPRI mean seismic hazard curve for PGA is used. Table 3-1 shows the EPRI hazard estimates for Kewaunee. The mean annual hazard curve defines the mean probability of exceedence of specified PGA levels at the plant site as determined in the EPRI seismic hazard study. The mean PGA hazard curve, which is defined to a maximum of 1.59g, is used in the seismic risk quantification. The EPRI mean annual seismic hazard curve for the Kewaunee site is presented in Figure 3-1.

The site specific seismic hazard curve generated by LLNL is used in performing sensitivity analyses as discussed in Section 3.1.5.4. The LLNL mean hazard estimates are presented in Table 3-1. The LLNL mean annual hazard curve is presented in Figure 3-2. These are the revised hazard estimates that are presented in NUREG-1488 (Reference 28) and endorsed by NRC Information Notice 94-32 (Reference 23) as the best available information on seismic hazard estimates.

3.1.2 Review of Plant Information and Walkdown

3.1.2.1 Plant Information

Kewaunee is designed to withstand the effects of unusual natural phenomena including earthquakes. The plant is designed to withstand a design basis earthquake (DBE) (also known as the SSE with a PGA of 0.12 g (12% of gravity). The operating basis earthquake (OBE) is one-half of the DBE event.

The Kewaunee site is a soil site with the safety-related power block structures founded on a clay-sand soil to an approximate depth to bedrock of 76 feet. The original building model is used in a nodal response analysis to obtain amplified floor response spectra (FRS) for the LLNL median ground spectral shape corresponding to the 10,000 year return period for the following buildings:

Containment
Auxiliary Building
Turbine Building
Administration Building
Technical Support Center

Since the screenhouse is entirely at or below grade (a deeply embedded structure) the ground spectrum is conservatively used as seismic demand for all elevations of the screenhouse. All safety-related systems and equipment are contained in the aforementioned power block structures.

Plant systems are chosen that were required to safely shut the plant down in the event of an earthquake. The list started as the Safe Shutdown Equipment List (SSEL) developed for the USI A-46 program. Since the USI A-46 program addresses mechanical and electrical equipment only, piping systems and plant structures were added to the list. Additionally, other equipment was added to the list to provide functions that are not addressed by the (USI) A-46 program.

3.1.2.2 Information Sources

As stated in the Kewaunee Updated Safety Analysis Report (USAR) (Reference 4), the plant buildings and systems are seismically designed. The USAR was used to obtain seismic design criteria for the DBE earthquake. The safety-related power block structures were evaluated by John A. Blume Associates, Engineers under contract to Pioneer Services, the Architect-Engineering firm responsible for the Kewaunee design. The Blume reports, (Reference 5) and (Reference 6), evaluate the seismic, dynamic response of the buildings providing amplified FRS for the DBE, and evaluate the seismic loads and stresses for the DBE, respectively.

The existing seismic evaluations of the safety-related piping and mechanical and electrical equipment were primarily found in the Kewaunee project engineering files originally developed by Pioneer Engineering, Inc. Safety-related piping was re-evaluated in accordance with requirements set forth in the IEB 79-14 piping seismic analysis program (Reference 7). This effort evaluated piping in the as-built configuration in accordance with current seismic-dynamic analysis procedures. Piping stress summaries and equipment stress analyses were obtained from these files. As-built and original installation drawings were used to obtain routing, equipment weights, and anchorage details.

Evaluation of site soil seismic adequacy and liquefaction potential was conducted by GEI Consultants, Inc. (Reference 8). Original site soil properties which formed the basis for the study were obtained from a geological study of Kewaunee performed in 1967 (Reference 9).

Much of the methodology of the seismic fragility program is based on the procedures prescribed in EPRI NP-6041 (Reference 10) which establishes bases for seismic "binning" and screening of nuclear power plant equipment, mechanical and electrical distribution systems, and power block structures. The basis for the procedures in EPRI NP-6041 use the same experience data as the Generic Implementation Procedures (GIP) (Reference 11) developed for resolution of the USI A-46 issue. Ancillary support documentation for the GIP and EPRI NP-6041 that are used in this study for Kewaunee include EPRI Reports NP-5228 (Reference 12) for anchorage issues, NP-7146 (Reference 13) for electrical cabinet amplification characteristics, and NP-7147 (Reference 14) for relay generic seismic ruggedness levels.

3.1.2.3 Plant Walkdowns

The Kewaunee SPRA takes advantage of the overlapping requirements between the IPEEE and A-46 examination programs. Seismic Review Teams (SRT) conducted the Kewaunee SPRA walkdowns following the walkdown procedures detailed in EPRI NP-6041. Each team consisted of two Seismic Capability Engineers trained by EPRI both in the USI A-46 walkdown requirements, and also in the IPEEE add-on requirements.

Wisconsin Public Service Corporation (WPSC), Stevenson & Associates (S&A), Jack R. Benjamin & Associates (JRBA) and RPK Structural Mechanics supplied the Seismic Capability Engineers. At least one WPSC engineer with systems expertise participated on each SRT throughout the plant walkdowns. The walkdowns were conducted between March 15 and April 2, 1993.

Walkdowns were conducted to evaluate plant equipment. For the sake of documentation, all equipment was treated as if it were an USI A-46 item, even if it was designated as SPRA equipment items only. As such, each equipment item has a Screening Evaluation Worksheet (SEWS) completed for it in accordance with GIP requirements as well as a fragility level assigned to it. Safety-related piping, electrical raceways and ductwork were walked down separately to assess fragility capabilities. Essential relays were evaluated based on seismic screening rules and then on circuit analyses. In accordance with GIP rules, spot checks of relays were made during walkdowns to confirm type (model number and manufacturer), location and installation adequacy. Structural screening walkdowns were conducted by Dr. John Reed of JRBA to assess the primary site structures and determine building fragilities.

Peer Review

An independent evaluation and peer review of the walkdown process was performed in accordance with the GIP by Dr. Paul Smith of The Readiness Operation on March 28 - March 30, 1993. As required, the review included an assessment of the walkdown and analyses by audit and sampling to identify any gross errors. Dr. Smith personally conducted two days of walkdowns with the SRTs and on his own to ascertain completeness and correctness of the SPRA and USI A-46 walkdown. His review included both comparing completed SEWS with equipment previously inspected by the SRTs and witnessing the SRTs performing actual evaluations in the field. Dr. Smith also reviewed the documentation packages the SRTs used to determine equipment design details that could not be readily determined by walkdown. Dr. Smith concluded that the walkdowns were being conducted competently and the findings made were appropriate. Documentation of Dr. Smith's peer review is provided in Reference 15.

3.1.3 Analysis of Plant Systems and Structure Response

This section discusses the development of the plant systems considered in the SPRA. The systems needed and the specific equipment comprising those systems are presented. Supporting systems such as electrical raceways are also presented. Structures containing these systems are identified and their seismic response characteristics are also discussed. Finally, site soil conditions and soil stability are presented in this section.

3.1.3.1 Plant Frontline Systems Included in the SPRA

Low Pressure Safety Injection System

The Low Pressure Safety Injection (LPSI) System and the safety injection accumulators are subsystems of Kewaunee's Emergency Core Cooling System (ECCS). It can be operated in the low pressure injection (LPI) mode and the low pressure recirculation (LPR) mode. The LPSI System and the accumulators provide emergency core cooling in the event of a break in either the Reactor Coolant System (RCS) or the secondary system. The purpose of the injection mode of operation for the various accidents is to terminate any reactivity increase following the postulated accidents, cool the core, and replenish coolant lost from the RCS. Upon depletion of the refueling water storage tank (RWST), the recirculation mode of operation is initiated to provide long term heat removal by recirculating the water that accumulates in the containment sump.

High Pressure Safety Injection System

The High Pressure Safety Injection (HPSI) System is a subsystem of Kewaunee's Emergency Core Cooling System (ECCS). It can be operated in the high pressure injection (HPI) mode, and the high pressure recirculation (HPR) mode.

The HPSI system provides emergency core cooling in the event of a break in either the RCS or the secondary system. The purpose of the injection mode of operation is to terminate any reactivity increase following the postulated accidents, cool the core, and replenish coolant from the RCS.

Upon depletion of the RWST, the recirculation mode of operation may be initiated to provide long-term cooling by recirculating the water that accumulates in the containment sump by way of LPR.

Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) System is an engineered safeguard system designed to supply high pressure feedwater to the steam generators (SGs) following an interruption of the Main Feedwater (FW) System supply. Periods when the AFW System may be required for the removal of residual heat from the core include startup, safety injection, failure of the FW System and for long term decay heat removal. AFW System operation prevents the release of reactor coolant through the pressurizer safety valves and removes the residual heat from the reactor core by heat transfer in the SGs.

3.1.3.2 Mechanical and Electrical Support Systems Included in the SPRA

Service Water System

The Service Water (SW) System is designed to provide redundant cooling water supplies to the diesel generators, safeguards compressors, safety injection pumps, containment fan coil units, turbine and auxiliary building safeguard fan coil units, control room air conditioners, component cooling heat exchangers, charcoal filter deluge for control room post accident recirculation, special zone ventilation, and shield building ventilation. The SW System provides an emergency supply of water to the Component Cooling Water (CCW) System, spent fuel pool and a backup source of water to the AFW System. The SW System also provides non-redundant cooling water supplies for balance of plant equipment.

Component Cooling Water System

The CCW System is designed to remove sensible heat from the RCS via the Residual Heat Removal (RHR) System during plant cooldown, startup and shutdown. It also provides cooling of water taken from the containment sump during the recirculation phase of emergency core cooling. The CCW System cools the RCS letdown and reactor coolant pump (RXCP) seal leakoff flows to the Chemical and Volume Control System (CVCS). The CCW System provides cooling for the RXCPs, low head safety injection (RHR) pumps, high head safety injection (SI) pumps and the internal containment spray (ICS) pumps. The CCW System serves as an intermediate loop between systems processing or containing radioactive fluids and the SW System. The CCW System thereby minimizes the chance of contaminating the SW System with leakage from radioactive system coolers.

Electric Power System

The Electric Power System (EPS) provides a reliable source of power to all plant systems required during normal or emergency plant operation.

The primary functions of the EPS is to:

- Provide a reliable source of motive power to those components whose operation is necessary for the mitigation of any abnormal event affecting the reactor core, its heat removal systems, or systems that could effect the release of radioactivity to the environment.
- Provide a reliable source of control power for the operation of these systems and for the initiation of safeguards systems actuation signals.
- Provide a reliable source of power to instrumentation necessary for the monitoring of emergency system functions, for the monitoring of key plant parameters, and for inputs to safeguards systems actuation logic matrices.

Reactor Protection System

The protective actions initiated by the Reactor Protection System (RPS) are broadly classified into two major categories, reactor trips and actuations of engineered safety features. Therefore, the RPS protective functions are addressed from two functionally defined subsystems, the Reactor Trip System (RTS) and the Engineered Safety Features Actuation System (ESFAS). These two subsystems perform all of the safety-related actions associated with the RPS.

The overall RPS consists of: Foxboro process instrumentation, nuclear instrumentation system, reactor protection, and safeguards logic relay cabinets, SI sequencer, and reactor trip switchgear. For reactor trip and engineered safety features actuation, the safeguards and reactor protection logic relay cabinets each contain two redundant logic trains, A and B, that are physically and electrically independent. The logic relay cabinets receive inputs from process instrumentation, nuclear instrumentation, field contacts, and directly from main control board switches.

3.1.3.3 Supporting Components Included in the SPRA

Piping

As discussed in Section 3.1.3.1, numerous systems are identified as being considered in the SPRA. These systems are reviewed to ensure that the mechanical piping is seismically designed in all instances. Seismically designed piping is screened out at a relatively high PGA level as discussed in Section 3.1.4. As part of the IPEEE walkdown a candidate piping system was walked down from end to end to verify design adequacy. Piping inertial failure is not the issue. Inadequate piping system flexibility and excessive relative support deflections are more likely contributors to seismically induced failures. Specific items that could diminish seismic capacity include:

- threaded or Victualic connections
- cast iron pipe
- inflexibly attached branch line
- excessive nozzle loads
- proximity of valve operators to structures, components and other systems
- poor supports
- lack of flexibility across seismic gaps

The A train of the SW System was inspected at Kewaunee in detail to investigate for the aforementioned concerns. Also, in general, other piping systems were observed during the course of the walkdowns for these concerns.

Electrical Raceways

A cable and conduit raceway review was conducted in accordance with Section 8 of the GIP in all safety-related areas of the plant. The raceway review consisted of: (1) a plant walkdown in which the raceways were evaluated against a set of walkdown guidelines, and (2) an analytical check of selected worst-case supports using a set of Limited Analytical Review Guidelines. The screening procedure for the raceways was based on earthquake experience and shake table test data. The plant walkdown focused on an inspection for design details and features that might result in poor seismic performance. The walkdowns were documented on Plant Area Summary Sheets (PASS). Representative, bounding support hangers were selected for ductility and load capacity evaluations for the worst-case Limited Analytical Reviews.

HVAC Ducting

Ductwork was inspected throughout the site buildings. The major concern in these inspections were anchorage adequacy and support details, such as no missing bolts or connections. Particular attention was given to ducting inside containment supported along the steel shell and inside the battery rooms where collapse could short circuit the emergency station batteries.

3.1.3.4 Site Buildings Included in the SPRA

All buildings containing systems to be considered in the SPRA as well as support systems included in the SPRA were assessed. A site walkdown of the structures was performed by Dr. John Reed of JRBA and was presented in a report to Stevenson and Associates (Reference 31). Included in the building assessment walkdown were:

- Containment vessel (including the internal structures)
- Administration building basement (diesel generator rooms)
- Shield building
- Auxiliary building
- Screenhouse building
- Screenhouse tunnel
- Turbine building mezzanine (battery rooms)
- Turbine building basement (safeguards equipment alley)
- Technical support center

The evaluations were made in accordance with EPRI NP-6041, Table 2-3. Field walkdowns were supported by a thorough review of the USAR, seismic stress evaluations by Blume (Reference 5), and design calculations and drawings generated by the Architect-Engineer, Pioneer/Fluor.

3.1.3.5 Structural Response

The original DBE or SSE seismic analysis FRS at Kewaunee are based on simple dynamic models and soil springs, with peak input ground acceleration of 0.12g. The IPEEE seismic motion of interest for the SPRA is well in excess of the design basis PGA. Given the availability of advanced soil-structure interaction (SSI) analysis techniques, it is expected that the effects of radiation damping in the soil and kinematic interaction would result in floor response spectra that would be more realistic. Although this was considered, the cost of this benefit was determined to exceed the value, given the relatively low seismic hazard for this site. Therefore, the original building models and soil springs were used to generate amplified floor response spectra for the IPEEE SPRA study. The FRS are considered conservative. The original model is based on three-dimensional lumped mass models with elastic half-space springs to represent the soil properties. Soil damping is assumed to be 5% of critical damping.

Uniform hazard spectra are used for the seismic input. In accordance with the provisions of NUREG-1407, the median shape for the 10,000 year return period as provided by the Lawrence Livermore National Laboratory Eastern Seismicity Report (Reference 16) is used for this study. Structural damping for all modes is set to 7% in accordance with the recommendation of EPRI NP-6041.

3.1.3.6 Soil Properties and Soil Failure Analysis

NUREG 1407 specifically requires the consideration of soil failure effects in the seismic IPEEE. Soil failure effects are considered from two perspectives: 1) soil liquefaction potential; and, 2) differential soil displacements under seismic conditions, as an input to buried component fragilities.

The stratigraphy of Kewaunee consists of alternating layers of heavily overconsolidated glacial till and glacial lacustrine deposits underlain by very dense glacial outwash overlying dolomite bedrock. The glacial till and glacial lacustrine deposits consist of a clay matrix with varying amounts of sand and gravel included. The outwash consists of relatively clean sand and gravel.

In the Dames & Moore report (Reference 9), a shear wave geophysical refraction survey was conducted across the site, which indicates that the shear wave velocities of the glacial deposits and the bedrock are 2,500 fps and 11,500 fps, respectively. GEI estimates the shear wave velocities of the glacial clays and the outwash using empirical relationships by Hardin (Reference 17) and by Ohta and Goto (Reference 18), which range from 500 fps near the surface to 1050 fps at a depth of 62.5 ft. for the glacial clay. The outwash estimated shear wave velocity ranges from 1000 to 1300 fps based on the Hardin and Ohta, et al, empirical relationship, respectively. Based on these considerations, it is GEI's opinion that the actual shear wave velocities are bounded by the results obtained using the empirical correlations and those of the shear wave velocity survey. The soil failure analyses were conducted for both sets of shear wave velocities.

The empirically obtained values are referred to as the stiff soil profile, and the geophysical survey results as the very stiff soil profile.

The results of the soil failure evaluation are provided in Section 3.1.4.2.

3.1.4 Evaluation of Component Fragilities and Failure Modes

The development of fragility values for equipment and structures in the Kewaunee SPRA proceeded through a three phase process: 1) component screening; 2) simplified fragility analysis; and, 3) detailed fragility analysis. This three phase consideration of the seismic ruggedness of plant equipment and structures efficiently concentrates attention on those items most significant to the overall assessment of seismic risk.

The potential for seismic interaction hazards results in specific seismic initiators within the system logic model. Thus the individual component fragilities represent the inherent seismic ruggedness of the components, independent of any seismic interaction hazards. The fragility of the interaction hazard, such as a masonry block wall, is applied to the hazard as an independent component with system logic linking the failure of the hazard to the failure of the affected components.

3.1.4.1 Component Screening

During the detailed plant walkdowns, the SRT engineers assigned a screening value to every component in the SPRA. EPRI NP-6041 supplied the framework for the screening decision making. Although the seismic margin procedure characterizes seismic ruggedness in terms of HCLPF level, the direct relationship between the HCLPF and median capacity supports the use of EPRI NP-6041 in performing SPRA component fragility screening. Followup anchorage analysis verified inclusion at either the screening level, or produced a fragility value for individual components based upon anchorage capacity.

Application of the screening guidelines to SPRA items resulted in the following categories based on peak spectral acceleration capacity:

- Screened out at the 1.2g HCLPF screening level;
- Screened out at the 0.8g HCLPF screening level, but does not meet the 1.2g criterion
- Does not meet the 0.8g HCLPF screening criterion, but the item is also in the A-46 program and meets design basis.
- Does not meet the 0.8g HCLPF screening criterion, and the item is not in the A-46 program.

Based on the results of the walkdowns, the project selected the first screening column in Table 2-3 of EPRI NP-6041 for the Kewaunee SPRA. All components meeting this screening level are screened out, and the balance are explicitly considered within the SPRA. As discussed below, a single surrogate element in the SPRA represents the aggregate effect of the screened out components. Engineering evaluations of anchorage capacities either validated the screening selection, or provided a simplified fragility value (second phase) on a component by component basis.

The Surrogate Element

The SPRA includes a single surrogate element representing the conservative capacity of the weakest component screened out. The surrogate element appears as a basic event in each system with failure leading directly to system failure. The median capacity of the surrogate element translates the EPRI NP-6041 spectral acceleration of 0.8g from a HCLPF to a median fragility expressed as PGA. The median capacity of the surrogate element computes as a direct function of the site ground spectral shape, which for the IPEEE is the median spectral shape with a 10,000 year return period provided in NUREG/CR-5250. The use of the Kewaunee ground spectral shape resulted in a median capacity of 0.64g for the surrogate element. Following the recommendations of Drs. Kennedy and Reed (Reference 19), the surrogate element has an associated combined uncertainty (β_C) of 0.3.

3.1.4.2 Fragility Analysis Results

The Kewaunee SPRA implements the concept of simplified fragility analysis as a means to bridge the gap between the summary level of the screening methodology and detailed fragility analysis. This approach improves on the use of industry generic fragilities by including plant specific analysis in the determination of component median seismic capacity values.

Simplified fragility analysis concentrates on determining the median seismic capacity taking actual plant specific conditions into consideration. All simplified fragilities use the same value for the combined variability (β_C) = 0.46. Techniques used in simplified fragility analysis include:

- Detailed anchorage analysis;
- Factoring analysis;
- USI A-46 equivalency analysis; and,
- Detailed stress analysis.

Anchorage considerations rely heavily on the availability of detailed and bounding analyses performed for components also within the USI A-46 examination program. For cases in which the USI A-46 results are not available, the SPRA capacity assures that components minimally meet the USI A-46 requirements using equivalency analysis. The SPRA treats the USI A-46 values as equivalent to a HCLPF for purposes of estimating the median capacity.

Factoring analysis converts available design analysis results to median capacity values substituting the IPEEE in-structure floor spectra for the existing design spectra as applicable. Factoring separates out the seismic component from other design loads, such as dead load and live load, following the methodology outlined in EPRI NP-6041.

The availability of IPEEE in-structure floor spectra makes the simplified fragility analysis concept a productive intermediate step for this SPRA. Component capacity values come from one of the following sources:

1. Available calculations, or reports of previous seismic analysis;
2. Detailed anchorage analysis performed for the SPRA floor spectra;
3. Generic Equipment Ruggedness Spectra (used for relays);
4. Application of the lower EPRI NP-6041 screening lane (0.8g); or,
5. Equivalency to the minimum GIP demand (A-46 components only).

Estimated Uncertainty for Simplified Fragility Analysis

The estimated fragilities couple an estimated median capacity (A_m) with an estimated logarithmic standard deviation, β_C, accounting for both randomness (β_R) and uncertainty (β_U). Examination of east coast earthquake records suggest that an adjustment to the commonly selected value for β_C would be appropriate to account for a higher degree of variability in the peaks and valleys. This examination suggests that a more appropriate value for the β_R associated with the randomness of the peaks and valleys of seismic records is 0.29. This consideration results in the selection of β_C = 0.46 for use with all but the detailed fragilities.

Mechanical and Electrical Equipment

Tables 3-3 and 3-4 provide the equipment and components considered in the SPRA. The equipment was walked down during a two week pre-walkdown in March of 1992. Final walkdowns were performed during a three week period in March and April of 1993. For the final walkdown, two SRTs participated during the first week and were comprised of Dr. R. P. Kennedy of RPK Consulting, Dr. J. D. Stevenson and W. Djordjevic of Stevenson & Associates, and P. Finnemore of WPSC. One SRT comprised of W. Djordjevic and P. Finnemore conducted the final two weeks of the walkdown.

The purpose of the walkdowns was to conduct a combined A-46 and IPEEE (screening) walkdown. As noted previously, all non A-46 items were treated as if they were A-46 items; therefore, complete SEWS exist for each of them as well.

All equipment was screened using the first column in Table 2-3 in EPRI Report NP-6041. The screening approach uses the experience gained in performing seismic margin assessments (SMAs) to screen components out of a SPRA. Meeting the caveats for these components ensures that they may be represented by a surrogate element with a median peak 5-percent damped spectral acceleration capacity of 1.2g - which is equivalent to 0.64g PGA - with a combined logarithmic standard deviation, β_C, value of 0.30.

Items that were not screened out at the 0.64g PGA value were evaluated in detail to determine their specific seismic fragility. Section 3.1.4.3 discusses detailed fragilities.

Relays

The clarifying note in Table 3.1 of NUREG-1407 classifies the Kewaunee plant as a focused-scope review for the purposes of performing relay chatter evaluation. Since the plant is included in the USI A-46 program, relays were evaluated in accordance with Section 6 of the GIP. A focused scope plant that is an A-46 plant thus needs only to conduct a "bad actor" relay review (see Reference 30 for list of bad actor relays) if such bad actors were found in the USI A-46 scope of review.

The USI A-46 review found 12 installations of the Westinghouse, Model SC relay in the 4160V switchgear. (Relays in Breakers 502 to 508 and 604 to 609.) The scope of the bad actor review was expanded to include the IPEEE equipment and no additional bad actor relays were identified.

WPS initiated Engineering Support Request (ESR) 94-005 proposing outright replacement of the 12 Westinghouse relays, or alternatively, a reworking of the system circuitry to exclude the Westinghouse relays after the emergency AC power system is initiated. As such, further consideration of the relay capacities in the SPRA model is not required. Details of the recommended solutions will be provided at a later date in the USI A-46 submittal.

Masonry Block Walls

The availability of recent detailed calculations for masonry block walls developed under the IEB 80-11 program provided a ready reference for the determination of estimated fragilities. Conservatively, the block walls were assigned a $HCLPF_{84}$ capacity equal to the design basis peak ground acceleration. This value was then factored in accordance with the guidance in Reference 18 to obtain a median capacity with an associated $\beta_C = 0.46$.

For the diesel generator day tank enclosures, which are formed with masonry block walls on three sides abutting a rear reinforced concrete wall, a $HCLPF_{84}$ value was calculated using the Conservative Deterministic Failure Margin (CDFM) methodology of EPRI NP-6041. The median capacity for this enclosure is 3.22g PGA with a $\beta_C = 0.46$.

Building Structures

The structures for the Kewaunee site considered in the SPRA are the reactor, auxiliary, and turbine buildings which are all founded on a single shallow mat foundation. The site is underlain by glacial till and lacustrine deposits as discussed in Section 3.1.3.6. These buildings are Category I structures except for the turbine building and the steel frame over the spent fuel handling area in the auxiliary building which are designated as Category III* structures. Specific areas of the turbine building mezzanine (battery rooms) and Administration building basement (diesel generator rooms) are also classified as Category I. Category III* indicates that the structure is not directly related to reactor operation or containment; however, it is designed to Class II seismic loads as prescribed by the 1967 Edition of the Uniform Building Code for Zone 1 areas (0.025g for K = 1.0).

All structures are screened using the first column in Table 2-3 in EPRI NP-6041. The screening approach uses the experience gained in performing seismic margin assessments (SMAs) to screen components out of a SPRA. Meeting the caveats for these structures ensures that the structures may be represented by a surrogate element with a median peak 5-percent damped spectral acceleration capacity of 1.2g with a combined logarithmic standard deviation, β_C, value of 0.30.

All caveats of Table 2-3 were dispositioned, including: the free standing steel containment's base mat shear keying, separations between structures, reinforcement detailing, and penetrations, including associated requisite piping flexibility. It was found that the steel structure over the spent fuel pool area and the turbine building have been designed to forces corresponding to the DBE ground response spectra (0.12g PGA). As such, they are also screened out of the SPRA.

The structures listed below are represented by the surrogate element in the SPRA with a median peak capacity of 1.2g, 5-percent damped spectral acceleration (which is equivalent to 0.64g PGA) with a composite logarithmic standard deviation, β_C value of 0.30.

- Containment vessel (including the internal structures)
- Administration building basement (diesel generator rooms)
- Shield building
- Screenhouse building
- Screenhouse tunnel
- Turbine building mezzanine (battery rooms)
- Technical Support Center (basement)

Reactor Vessel Internals and Control Rod Drive Housing and Mechanisms

The reactor vessel internals simplified fragility analysis also applies factoring analysis using the results of the original stress analysis reported in the USAR. All of the vessel internals have median capacities in excess of the surrogate element capacity. Specific components evaluated as listed in the USAR are:

Upper Core Barrel, Upper Core Plate, Rod Control Cluster Guide Tubes, Fuel Assembly Thimbles, Lower Core Barrel, Barrel Flange, Fuel Assembly Top and Bottom Nozzle Plates, Upper Support Columns, Lower Radial Support.

The control rod drive housing has a lateral seismic support (Reference 20) and may thus be screened out and assigned the surrogate element capacity according to Table 2-4 of EPRI NP-6041.

Soil Failure Analysis and Buried Piping

Soil stability and seismic displacements, both transient and permanent, along with permanent settlements were investigated for the Kewaunee site.

Due to the high factor of safety, the evaluations were conducted at 0.7g PGA using the SHAKE (Reference 21) computer program. The intake and discharge piping were also evaluated for the imposed displacements and settlements at 0.7g PGA.

Liquefaction

The results of the liquefaction stability analyses conducted for the power block and for the screenhouse structure give factors of safety equal to 9.1 and 2.0, respectively. These results indicate that a liquefaction instability failure of the critical structures at Kewaunee is very unlikely regardless of the earthquake magnitude or the value of PGA.

In addition, empirical correlations are used to assess the likelihood of significant pore pressure build-up in the soils. The results indicated that the likelihood of 100 % pore pressure development is remote, even for a peak ground acceleration of 0.7g. This is true for both the stiff and very stiff soil properties.

Transient and Permanent Horizontal Displacements and Settlements

The maximum transient horizontal displacements calculated at the ground surface for a PGA of 0.7g are 1.2 inches for the stiff soil profile and 0.16 inch for the very stiff soil profile. The differential displacement between a building and the surrounding ground can be conservatively taken to be uniformly distributed over a distance of about 25 feet from the foundation.

The permanent horizontal displacements for a PGA of 0.7g are estimated to be 0.14 inch for the stiff soil profile and about 0.5 inch for the very stiff soil profile. The differential displacement between the screenhouse and the intake crib approximately 1600 feet from the shoreline was evaluated.

The maximum calculated settlements at the ground surface range from 0.75 inch to less than 0.2 inch at 0.7g peak ground acceleration for the stiff and very stiff soil profiles, respectively.

Differential settlements can be expected within the foundation imprint of any one building and within the areas between buildings due to natural variability of the compressibility for the soil deposits. These can be taken equal to 50% of the total settlements and can be taken to occur over a distance of about 25 feet for structures on individual spread footings and for the areas between buildings. For structures founded on a continuous mat foundation, the differential displacement can be taken to occur over a distance of about 50 feet.

Differential settlements can also be expected between any one building and the ground and between adjacent buildings, such as those within the power block, due to the different thicknesses of the soil strata beneath the various structures and beneath the ground surface. Those between a building and the surrounding ground occur over a distance of only a few feet. The distance over which the differential settlements between adjacent buildings occurs is dependent on the interaction of the foundation mat with the foundation soil and can occur abruptly at construction or expansion joints between or within the buildings.

Buried Piping From Screenhouse and Intake Crib

The SPRA considers the influence of the displacements and settlements for the fragility analysis of intake and discharge piping from the screenhouse building and intake crib. At the imposed

0.7g PGA, the aforementioned steel and concrete piping do not exceed normal design allowables; thus, the buried piping is screened out of the SPRA and conservatively represented by the surrogate element.

Evaluation of Supporting Component Capacities

Piping

Piping was reviewed throughout the plant as part of the SPRA walkdown. All safety-related piping was re-evaluated using modern dynamic analysis procedures as part of the IEB 79-14 program. It was verified that no reliance on other than seismically designed piping systems was considered in the SPRA model.

Train A of the Service Water System was walked down from end to end to identify any anomalies as discussed in Section 3.1.3.3 during the first week in April, 1994. No design anomalies were found during this system walkdown.

The piping is represented by the surrogate element in the SPRA with a median peak capacity of 1.2g, 5-percent damped spectral acceleration (which is equivalent to 0.64g PGA) with a composite logarithmic standard deviation of 0.30.

Small bore piping was also reviewed during the walkdowns to consider any interaction effects that could result from such piping, for example, falling (collapsing) on equipment modeled in the SPRA. It was observed that piping supports would support more than three times an estimated dead weight and that support spacing was within two times that recommended by the ASME B.31.1 code for piping. Therefore, this issue is considered resolved and small bore piping may be considered to have the same capacity as the seismically designed large bore piping.

HVAC

Ducting in all areas of the plant was reviewed with particular attention given to containment systems, and those in the control room and battery rooms. In general, the smaller size ducting is supported by sheet metal straps secured to the ceiling by expansion anchors. Larger duct cross-sections are supported by rod trapeze hangers anchored by Phillips shells. In containment, the ducts are supported off the steel containment shell by braced cantilever brackets. The duct is circular in cross-section and attached to the brackets by large threaded hoops.

The duct is supported in accordance with Sheet Metal and Air Conditioning National Association (SMACNA) (Reference 22) spacing rules, and the anchorage vertical capacities exceed three times dead weight. The ducting is represented by the surrogate element in the SPRA with a median peak capacity of 1.2g, 5-percent damped spectral acceleration (which is equivalent to 0.64g PGA) with a composite logarithmic standard deviation of 0.30.

Electrical Raceways

The electrical raceways were walked down as part of the USI A-46 effort. All areas of the plant were surveyed and inspected against inclusion rules and caveats for raceways such as maximum spans, missing or broken hardware, and good design practices as presented in the GIP, Section 8. The results are documented in Plant Area Summary Sheets and will be presented at a later date in the USI A-46 submittal. In addition, bounding and representative supports were selected for structural and seismic evaluations called Limited Analytic Reviews. (LAR). The LAR evaluations check dead load stresses, ductility, and vertical capacity.

No anomalies were found during the plant tour of the raceway systems. The LAR evaluations resulted in no outliers. In conclusion, the electrical raceways are represented by the surrogate element in the SPRA with a median peak capacity of 1.2g, 5-percent damped spectral acceleration (which is equivalent to 0.64g PGA) with a composite logarithmic standard deviation of 0.30.

Seismically Induced Loss of Coolant Accidents

Figure 3-3 presents the relationship between increasing seismic levels, and the conditional probability of a small and medium break loss of coolant accidents. The data comes directly from NUREG/CR-4840 (Reference 24) and is included in the SPRA model.

3.1.4.3 Detailed Fragilities

Residual heat removal (RHR) heat exchanger A and B fragility is 0.63g PGA which is slightly less than the surrogate fragility. All detailed fragilities for other base mounted equipment exceed the surrogate fragility of 0.64g PGA.

Other components that require corrective action are screened at the surrogate level. These components are presented in summary on Table 3-4 and will be presented in detail at a later date in the USI A-46 submittal.

3.1.5 Analysis of Plant Systems and Sequences

The analysis of plant systems and sequences section is broken down into four sub-sections and is organized so that each sub-section discusses a step in the quantification process. The four sub-sections are as follows: seismic initiating events, seismic event trees, seismic fault trees, and seismic risk quantification and results including sensitivity analyses.

3.1.5.1 Seismically Induced Initiating Event Analysis

NUREG/CR-4840 recommends that the following events establish the basis for the seismic initiating event categories.

- Reactor Vessel Rupture
- LOCAs - Large, Medium, and Small
- Transients

In addition NUREG/CR-4840 recommends that any site specific failures that could result in an initiating event should be considered. In order to properly account for all possible site specific initiators the following analysis methodology is employed:

a. The structures and equipment that should be used to determine the plant status following a seismic event are determined.

b. Given the failure of each of the items listed in step a, the plant status following a given earthquake is defined. Failures with similar results are grouped together (i.e. components whose failures resulted in a large LOCA are grouped into one "family").

c. Fragility data representing each component "family" is developed.

d. An event tree is then developed which contains, as its first node, the Kewaunee site specific seismic hazard analysis. The remaining top event nodes represent the structure and component "families" identified in steps a and b, which were developed to describe the plant status after an earthquake.

The items chosen in step a are plant specific, including major buildings and structures, RCS equipment and supports, switchyard ceramic insulators, and other plant specific items identified during the seismic walkdown. In addition, any other plant specific components the analyst determined to be capable of causing an initiating event are considered.

The structures and equipment that are used to define the seismically induced initiating events for the Kewaunee SPRA include:

<u>Structures</u> - Containment building, turbine building, auxiliary building, screenhouse, technical support center, reactor pressure vessel supports, steam generator supports, pressurizer supports, reactor coolant pump supports, reactor coolant system piping, and secondary piping and supports.

<u>Equipment</u> - Reactor pressure vessel, steam generators, pressurizer, reactor coolant pumps, control rod drive mechanism, and switchyard ceramic insulators.

It should be noted that the above list is not exhaustive, but rather includes a list of equipment and structures that would lead to initiating events considered to be "worst case" events. For example, failure of the containment shell structure is assumed to result in total collapse of the containment in such a way as to rupture the reactor pressure vessel; hence direct core damage with loss of containment is assumed.

From the above identified items, twelve failure groups are defined. The following paragraphs are a compilation and definition of these failure groups. The equipment and/or structures, along with their individual and combined fragilities, considered in these failure nodes are listed in Tables 3-2 and 3-3.

(1) CONTAINMENT OR STEAM GENERATOR FAILURE - (CSF)

Containment structural failure is assumed to be a function of either the containment rebar or soil pressure fragilities. Failure is assumed to be structural collapse of the containment building. Damage to the RCS and the core cooling systems due to the containment collapse is hypothesized.

Steam generator (SG) integrity is assumed to be a function of the fragility of the SG and its supports. Both SGs are assumed to fail in such a way as to sever both RCS and secondary side piping (outside of containment).

Thus, either of these failures is assumed to cause direct seismically induced core damage with containment breech.

(2) REACTOR VESSEL, RCS PIPING FAILURE, OR BUILDING FAILURES - (RVB)

This failure is assumed to be a function of the fragility of the reactor pressure vessel (RPV), the RPV supports, the RCS piping, and any building failures that could render systems within the failed building unavailable. RPV or RPV support failure is assumed to lead to a vessel position that could not guarantee core cooling. RCS piping failure is assumed to be double guillotine breaks in all loops at a point that precludes emergency core cooling.

The building failures of major concern are the turbine building, auxiliary building, and the screenhouse. Failure of the screenhouse disables the SW System and thus, all of the systems that the SW System supports would consequently fail. Failure of the turbine and auxiliary buildings are assumed to fail all systems housed within these buildings. Examples of systems that are assumed to fail include the RHR System, diesel generators, AFW System, and it is assumed that this failure mode would sever all wires running through the failed buildings which would result in a loss of all power and indication.

In either case, since the core cooling function is not guaranteed, this failure is conservatively assumed to lead to direct core damage; no direct damage to the containment is assumed.

(3) OFF-SITE POWER FAILURE - (OSP)

Off-site power availability is assumed to be a function of the switchyard ceramic insulators. Failure of these insulators is assumed to lead to a complete and unrecoverable loss of off-site power. Since the ceramic insulators have a very low seismic tolerance, any seismic activity that generates a seismically induced initiating event, will also result in the failure of the ceramic insulators and thus an unrecoverable loss of off-site power. Any seismic activity that does not cause a failure of the ceramic insulators, is assumed to be of such a small magnitude that it does not generate any other seismic initiating event and is therefore encompassed by the IPE. Therefore, the only structures or components considered in this node are the ceramic insulators. It should be noted that no credit is taken or assumed for off-site power recovery and that there is a strong reliance on the on-site AC Power System and components.

(4) AC POWER (ACP)

This failure is assumed to be a function of the combined fragilities for vital AC Power System components. The components considered include cable trays, switchgear and diesel generators. Equipment related to the diesel generator operation such as fuse panels, control cabinets, sequencer panels, fuel oil day tanks and fuel oil storage tanks were also included.

(5) SERVICE WATER SYSTEM (SWS)

This failure is assumed to be a function of the combined fragilities for the SW System. The components considered include air operated valves, check valves, motor control centers, motor valves, piping and pumps.

(6) DC POWER (DCP)

This failure is assumed to be a function of the combined fragilities for vital DC Power System components. The components considered include batteries, battery chargers, battery racks, cable trays, motor control centers, distribution cabinets and fuse panels.

(7) RCS COMPONENT FAILURE - (RCF)

This failure is assumed to be a function of the pressurizer, the pressurizer supports, the reactor coolant pump and supports. The failure of any of these components is assumed to lead to a large break (Pipe ID > 6") LOCA event. Neither emergency core cooling nor containment cooling functions are compromised as a direct result of these failures.

(8) CONTROL ROD INSERTION FAILURE - (ROD)

The ability to insert the control rods is assumed to be a function of the control rod drive mechanism fragility, as well as the fragilities of the reactor core upper internals, and the lower internals i.e., core barrel and thermal shield. All of these seismically induced failures are postulated to fail in such a way as to either disable the control rod drive mechanism, or physically prevent rod insertion.

(9) MEDIUM PRIMARY PIPE BREAK - (MPB)

This category of initiating events are based on failure of reactor coolant pipes and pipe combinations leading to medium break LOCA (3" < Pipe ID < 6"). NUREG/CR-4840 presents a statistical distribution (Figure 3-5), which is used to estimate the medium break LOCA initiating event in the Kewaunee SPRA. The use of this input to the SPRA precludes the need to model a surrogate element for medium breaks.

(10) SMALL PRIMARY PIPE BREAK - (SPB)

This category of initiating events are based on failure of reactor coolant pipes and pipe combinations leading to small break LOCA (1.5" < Pipe ID < 3"). NUREG/CR-4840 presents a statistical distribution (Figure 3-5), which is used to estimate the small break LOCA initiating event in the Kewaunee SPRA. The use of this input to the SPRA precludes the need to model a surrogate element for small pipe breaks.

(11) REACTOR COOLANT PUMP SEAL LOCA (SEAL)

Loss of RCS integrity is a functional failure that could result in a loss of coolant accident LOCA. Some of these failures were considered in nodes RVB, RCF, MPB and SPB. A small break LOCA could result from the failure of the RXCP seals. This failure is assumed to be a function of the fragility of the pump seals.

In addition, the RXCPs are assumed to fail in such a way as to damage the seals on both pumps. The resultant leakage is assumed to be equivalent to that required for small LOCA classification.

(12) SECONDARY SIDE PIPE BREAK - (SSP)

The integrity of the secondary side piping is assumed to be a function of the fragility of the secondary side piping and its supports. Failure is assumed to lead to a steam line or feed line break. It is assumed that both SGs will be affected.

After the above equipment and structures are clearly defined, construction of the SPRA initiating event tree begins. There are several important rules that were followed while creating the SPRA initiating event trees. It is important that a proper hierarchy among the initiating events is developed. The order of the hierarchy is defined such that, if one initiating event occurs, the occurrence of other initiating events further down the hierarchy are of no significance in terms of plant response. Thus, for example, if a large break LOCA occurs, we are not concerned if a small break LOCA or transient occurs as the plant's response requirements are dictated by the need to mitigate the large break LOCA.

Implicit in the defined hierarchy of the set of initiating events is the requirement that the basic events that define an initiating event cannot occur in the accident sequences corresponding to initiating events lower in the hierarchy. For example, loss of offsite power can occur as a basic event in any of the LOCA sequences, but not vice versa. The hierarchy used in the development of the Kewaunee seismic PRA is presented in Table 3-5.

The SPRA initiating event tree can be found in Figure 3-4. Once the SPRA initiating event tree is completed, the individual sequences of the event tree are then converted into logic equations that represent the seismically induced initiating events. Each end state of this event tree represents an initiating event category. The following five initiating event end states are not resolved at this stage of the quantification. The sequence logic equations that represent these end states are used as the initial input into the sequence analysis as described in Section 3.1.5.2.

- SLSP - Loss of Off-site Power
- SSLB - Steam Line Break
- SSLO - Small LOCA
- SMLO - Medium LOCA
- SLLO - Large LOCA

There are six end states from the SPRA event tree that are not quantified beyond the SPRA initiating event tree. These end states are:

- CSF - Containment or steam generator catastrophic failure
- RVB - Reactor vessel or Reactor Coolant System piping or buildings catastrophic failure
- ACP - AC power catastrophic failure
- SWS - Service Water System catastrophic failure
- DCP - DC power catastrophic failure
- RDF - Control rod insertion failure

The first end state in the above list represents core damage with a failed containment. Therefore, for this end state there is a direct path for fission products to the atmosphere.

Since it is assumed, in this analysis, that any seismic activity results in an unrecoverable loss of off site power, the endstate STRS is not quantified. The end state STRS represents a transient with off-site power available.

3.1.5.2 Seismic Event Tree Analysis

Once the seismic initiators are defined, the seismic event trees are created from the applicable IPE event trees. To take advantage of the work done in the IPE, the foundation for the seismic event trees are the Level 1 IPE event trees. In order to differentiate between the normal and seismic Level 1 IPE event and fault trees, a naming convention for the seismic trees was developed. The seismic event trees are identified by a "S" prefix added to the internal event names, the seismic top event fault trees are identified by a "SX" prefix added to the internal event name, and the seismic support system fault trees are identified by a "X" suffix added to the internal event names. The "X" suffix is also used to designate internal event fault trees in which it is necessary to add seismic failures. This was due to the fault tree structure or when basic assumptions are different in the SPRA versus the IPE.

Since the seismic failure probability associated with the ceramic insulators is high, all of the event trees are constructed and quantified assuming off-site power is not available. As can be seen in Figure 3-4, the third top event node OSP in the SPRA initiating event tree is the failure of the ceramic insulators. It can also be seen that the only endstate that does not occur after the failure of the ceramic insulators is the first endstate (STRS), which represents a transient initiating event with off-site AC power. Therefore, since this initiating event requires the availability of off-site power it is not included in the seismic analysis.

The second global assumption needed during event tree construction is that the instrument air system is unavailable after a seismic event. Since a portion of the instrument air system is non-seismic class I, the entire instrument air system is assumed to fail during a seismic event. As a result of this assumption, several top event and support systems are removed from the SPRA analysis. The most visible system to be removed, at the sequence level, are the pressurizer PORVs. Therefore, the ability for primary bleed and feed is lost. The steam generator PORVs are not removed from the seismic analysis because they have a seismically qualified air supply. This supply is a seismically qualified accumulator system that is used to actuate the PORVs in a loss of instrument air scenario. Therefore, the steam generator PORVs are available during and after a seismic event to provide secondary depressurization.

The results from the seismic initiating events analysis determine that there are five events that need to be quantified for the seismic analysis. These events included large break LOCA, medium break LOCA, small break LOCA, steam line break, and loss of off-site power.

As a result of the above discussion, it is necessary to make several changes to the Level 1 IPE event trees to transform them into seismic event trees. The following is an event tree by event tree description of the changes:

- Seismic Large LOCA (SLLO)

  The seismic large LOCA (SLLO) event tree is presented on Figure 3-5. The major assumptions included in the IPE event tree are the same for the seismic event tree. As previously discussed, support systems such as AC power, DC power, and SW are removed from the system level analysis and placed into the sequence level analysis as shown on the seismic initiating event tree (Figure 3-6). For the large break LOCA event tree, two other support systems, CCW and the RPS are quantified at the sequence level. It is assumed that the failure of either CCW or RPS leads to core damage.

- Seismic Medium LOCA (SMLO)

The seismic medium LOCA (SMLO) event tree is presented on Figure 3-6. One major assumption changes as compared to the IPE event tree. Top event OM0 is removed. This node represents the Main Feedwater System, which is assumed to be lost as a result of the loss of off-site power due to the seismic initiator. Support system modeling discussed for the large LOCA event tree also applies here. An additional top event (RST) is included and represents the failure of the refueling water storage tank. Failure of RST is assumed to lead to core damage.

- Seismic Small LOCA (SSLO)

The seismic small LOCA (SSLO) event tree is presented on Figure 3-7. Three major assumptions changed as compared to the IPE event tree.

1. Top event OM0 is removed. This node represents the Main Feedwater System, which is assumed to be lost as a result of the loss of off-site power due to the seismic initiator.

2. Top event OB1 is removed. This node represents the operator action and the associated hardware necessary for primary bleed and feed. This node was removed because of the loss of instrument air, which is assumed tc fail due to the seismic initiator.

3. Top event ES1 is removed. This node represents the operator action and the associated hardware necessary to cool down and depressurize the RCS for charging flow. This node like the OB1 node is dependent on instrument air for success and is removed for the same reason.

Support system modeling including the refueling water storage tank (RST) discussed for the medium LOCA event tree also applies here.

- Seismic Loss of Off Site Power (SLSP)

The loss of off-site power on all SPRA event trees is accounted for in the quantification of the event trees through the choice of mitigating systems. Any system dependent on off-site power was eliminated from the event tree when the seismic event tree logic was constructed. The value for seismic fragility for loss of off-site power is based on values used in recent PRA evaluations based on a discussion with Dr. R. P. Kennedy (Reference 25).

The seismic loss of off site power (SLSP) event tree is presented on Figure 3-8. Two major assumptions change as compared to the IPE event tree.

1. Top event OSP is removed. This node represents the availability of on site power. This is addressed in the seismic initiating event tree node ACP (Figure 3-2).

2. Top events OB1 and HR1 are removed. The OB1 node represents the operator action and the associated hardware necessary for primary bleed and feed. This node is removed because of the loss of instrument air, which is assumed to fail due to the seismic event. Node HR1 is only addressed after the success of OB1. Since OB1 is assumed to fail, HR1 does not need to be addressed.

Support system modeling discussed for the large LOCA event tree also apply here.

- Steam Line Break (SSLB)

The seismic steam line break (SSLB) event tree is presented on Figure 3-9. Two major assumptions changed as compared to the IPE event tree.

1. Top events OM1 and OB4 are removed. These nodes are the same as OM0 and OB1 previously described and are removed for the same reasons.

2. Top event HR1 is removed. This node represents the high pressure safety injection lined up in the recirculation mode. This node is only addressed after the success of OB4. Since OB4 is assumed to fail, HR1 does not need to be addressed.

Support system modeling discussed for the large LOCA event tree also applies here.

3.1.5.3 Seismic Fault Tree Analysis

Once the seismic event trees are developed, the seismic fault trees are created. The seismic fault trees are defined by the seismic event tree top events and those support systems that require a fault tree. Note that some event tree top events do not require a seismic fault tree since the event does not model any components that could fail seismically.

The first step in creating the seismic fault trees is to identify which components are adversely affected by a seismic event. The components expected to experience seismic failures were identified during the seismic walkdown and are discussed in Section 3.1.4. The list of components includes system pumps, tanks, valves and electrical equipment. The components described in Section 3.1.4 are used to construct seismic fault trees for each system of interest.

The selection of components to be modeled is the result of a screening process by which seismically rugged structures and components are screened out. Using this methodology presents a special problem in that all failure modes cannot be accurately defined. To account for these failure modes a surrogate element is modeled. The surrogate element is discussed in Section 3.1.4.1. The surrogate element is intended to represent the conservative capacity for the weakest component screened out. A surrogate element is modeled as a basic event in each system fault tree in series with the cutsets for the top event. The results of a seismic quantification using this methodology are somewhat conservative but has the advantage of providing added flexibility to evaluate sensitivity.

The seismic fault trees are then quantified and linked to the corresponding IPE fault trees (e.g., seismic fault tree SXLR1 is linked to IPE fault tree LR1). By combining these trees together, both seismic and non-seismic failures are considered in the event tree accident sequence quantification. Thus, the IPE fault tree component random failures, human errors, and test and maintenance unavailabilities are all considered in the seismic PRA. The fault trees are quantified in a manner that ensures all seismic events modeled in the SPRA are present in the final cutsets used in sequence quantification. Table 3-6 lists the top events and support systems that require a fault tree.

The seismic fault trees are constructed based upon the following assumptions:

1. In order to remove some of the complexity involved with seismic quantification, it is assumed that analogous components in close proximity, simultaneously fail with a probability equal to that of one component. For example, if one pump in a two pump system seismically fails, its redundant partner fails without a decrease in the failure probability. This assumption, which is analogous to using a common cause β factor of 1.0, conservatively removes train redundancy while simplifying the seismic fault trees.

2. In an attempt to accurately model operator actions, it is assumed that the probability of the operator failing to perform specific actions is a function of ground motion level. Operator actions are modeled by a non-lognormal fragility curve by which the conditional probability of failure as a function of ground motion level is defined in terms of a discrete function. For low ground motions, up to and including the SSE (0.12g), the operator error rates are assumed to be the same as those used in the IPE. For ground motions greater than the SSE, the operator error rates increase in a linear manner to three times the SSE (0.36g). It is assumed that the operator error rates would increase by a factor of ten at three times the SSE. Operator actions performed at ground motions greater than 0.36g are conservatively assumed to fail (failure probability of 1.0).

3. Systems not classified as Seismic Class I are conservatively assumed to fail at any seismic activity level. That is, the failure probability is 1.0. This assumption applies to the loss of the station and instrument air system, which in turn removes all pressurizer PORVs and therefore, the possibility for bleed and feed operations. Like the Station and Instrument Air System, the Feedwater and Condensate Systems are also not Seismic Class I. Because of this assumption, it is convenient to remove top events that either rely on instrument air or whose success is dependent upon the success of a previous event that depends on instrument air.

   The SW and AFW Systems rely on instrument air. However, the AFW air operated valves fail to a predetermined position upon loss of instrument air to permit continued operation. The SW System needs instrument air for automatic backwashing of traveling screens and pump strainers. This failure is not however, a major contributor to system unavailability and these components are addressed in the seismic quantification. Thus, failure of the station and instrument air system does not directly fail the SW and AFW Systems.

4. The condensate storage tank (CST), which supplies water to the AFW pumps, is not classified as Seismic Class I and is assumed to fail at any seismic activity level. The CST is therefore removed from the AFW fault trees. With the failure of the CST, the operators

are required to transfer over to the SW System as the source of water to the AFW pumps. This operator action was modeled consistent with assumption 2 above.

5. It is assumed that failure of plant structures destroys everything within the structure confines. The electrical wires that run through structure walls are also sheared. Each structure failure is assumed to be independent of other structure failures.

6. Failure of off site power is assumed to have a duration of 24 hours. Thus, the diesel generators have a defined mission time of 24 hours.

7. Seismic failure of the pressurizer safety valves is not modeled since these valves are normally closed and it is expected that they would fail closed due to seismic activity.

3.1.5.4 Seismic Risk Quantification and Results

This section describes the procedure used to perform the seismic risk quantification and presents the results of the Kewaunee SPRA and the findings of sensitivity analyses that were performed.

Quantification Procedure

The Kewaunee SPRA is quantified using the SHIP code. SHIP was developed by Jack R. Benjamin and Associates, Inc. specifically to perform the computations required to estimate seismic risk.

Input to SHIP includes:

- discretized seismic hazard at a site (i.e., annual probability of exceeding specified levels of ground motion),

- component (e.g., structure or equipment item) fragility data defined in terms of the median capacity and logarithmic standard deviation, which are used in SHIP to define a lognormal fragility curve,

- component random failure probabilities,

- operator error probabilities, and

- seismic system logic model.

The seismic system logic model is input to SHIP in the form of:

- event tree sequence logic, and/or

- fault tree cutsets.

The seismic event tree is input for each sequence in terms of the combination of top event failures and successes. System fault trees are input to SHIP by means of the cutset files produced by the Westinghouse system analysis software, WLINK (Reference 26), which was also used in the IPE. In the Kewaunee SPRA, SHIP performs the seismic risk quantification in three steps.

1. Systems Evaluation - For each top event in the seismic event trees, SHIP evaluates the corresponding fault tree[1] to determine the conditional probability of the top event over a specified range of ground motions. To solve for the conditional probability of the top event (for each PGA interval), SHIP uses the SIGPI algorithm (Reference 27). SHIP combines individual component conditional failure probabilities, according to the system logic, to determine the system level conditional probability of failure (fragility). This evaluation is repeated for each ground motion interval to produce a fragility curve for each system or top event.

2. Event Tree Quantification - In this step, sequence level fragility curves are computed. For each accident sequence defined in the seismic event tree, SHIP combines the top event failures and successes according to the sequence logic. This evaluation is performed for each ground motion interval to produce a sequence level fragility curve. The set of sequence level fragility curves is combined to produce the plant level or core damage fragility. The plant fragility curve has the same characteristic shape as the fragility curve for a component.

3. Risk Quantification - The seismic core damage frequency (CDF) is estimated by combining the plant level fragility with the mean seismic hazard curve for the site. In addition, the frequency of each accident sequence is also estimated by combining the site hazard curve with the sequence fragility.

Figure 3-10 shows the input and output involved in a SHIP application.

Quantification Results

The Kewaunee SPRA is quantified by evaluating the 47 sequences defined in the seismic event trees (Figures 3-4 through 3-9). Table 3-7 presents the results of the Kewaunee SPRA in terms of the mean CDF, the plant median capacity and HCLPF level. The Kewaunee plant fragility provides a measure of the seismic capacity of the plant in terms of the level of earthquake loading. It is a composite measure of seismic integrity, accounting for the seismic design of structures and equipment, redundancy in the plant safety system design, and the role of operators in mitigating an accident. In the following, the SPRA fragility and CDF results are discussed.

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[1]Note in some cases a top event in the event tree may be a single basic event.

Kewaunee Fragility Results

The seismic capacity of the plant is presented in terms of the:

- plant-level fragility,
- sequence fragility curves, and
- system (or top event) fragility curves.

Plant Fragility - Figure 3-11 shows the Kewaunee plant fragility curve. The plant fragility is derived from a quantification of the seismic systems logic model, component seismic fragility data, random failure probabilities, and operator error probabilities. As indicated in Figure 3-11, the median capacity of the plant is 0.38g PGA, which is approximately a factor of three times the Safe Shutdown Earthquake level of 0.12g.

In the Kewaunee SPRA, the mean plant fragility is calculated. Although an uncertainty analysis is not performed, an estimate of the plant HCLPF can be made. Assuming the plant fragility curve is lognormally distributed, the HCLPF corresponds to the ground motion that has a 0.01 conditional probability of failure. (Note, if a full uncertainty analysis was performed, the HCLPF would correspond to the ground motion level for which there is 95 percent confidence that the probability of failure is 0.05 or lower.) Table 3-8 summarizes the plant level fragility at various failure fractions. For Kewaunee the HCLPF is 0.23g. As discussed below, this is a conservative estimate of the HCLPF, due to the fact that the plant fragility includes the effects of random (non-seismic) failures and operator actions.

The purpose of a HCLPF is to provide a measure of the seismic structural integrity of structures and equipment items. As noted in NUREG-1407, the plant HCLPF determined in a SPRA is different from the result that would be obtained in a seismic margin assessment (SMA). In the SPRA, random failures and operator actions are included in the quantification of the plant systems. The contribution of these basic events to the plant fragility is greatest at low ground motions where the conditional probabilities of failure are small (in the lower tail of the fragility curve). Therefore, with the addition of random failures and operator errors, the likelihood of failure at low ground motions is higher. As a result, the conditional probability of failure of 0.01 is reached at a ground motion that is lower than it would otherwise be if only seismic failures were considered. In order to determine a measure of the plant HCLPF based only on the seismic structural integrity of the plant, the random failures and operator actions should be removed from the model. In the Sensitivity Analysis section, the sensitivity of the plant HCLPF to the random failures and operator errors is considered.

Sequence Fragility - Figure 3-12 shows the plant fragility with the sequence level fragility curves for the dominant seismic sequences. Figure 3-13 shows the fragility curves for the remaining sequences in the seismic event trees. Based on their contribution to the plant fragility, the following seismic sequences are the primary contributors:

- Failure of the containment structure or the steam generator (SCSF)
- Failure of the emergency AC power system (SACP)
- Failure of the screenhouse, the auxiliary building, turbine building or the reactor vessel (SRVB)
- Failure of the Service Water System (SSWS)
- Loss of off-site power (SLSP01)
- Failure of emergency DC power system (SDCP)

The sequence fragility curves initially increase and then fall off with increasing ground motion. This is due to the combination of system (top event) failures and successes that are involved in a sequence. With increasing ground motion, the probability of a system not failing (success) decreases from a conditional probability of 1.0 at low ground motions to 0 at high ground motions.

System (Top Event) Fragility - The fragility of individual plant systems or top events are presented in terms of system level fragility curves. The following lists the figures and system/top event fragility curves that are presented:

Figure No.	Systems/Top Events
3-14	Charging System (SXCHG), Reactor Protection System (SXRPS), Steam Line Isolation (SXIS1) Auxiliary Feedwater System (SXAF1, SXAF3), Service Water System (SXSWS)
3-15	Low Pressure Recirculation (SXLR1, SXLR2, SXLR4), Low Pressure Injection (SXL11, SXL12,) High Pressure Injection (SXHI0, SXH12, SXH13), Component Cooling Water (SXCCW)
3-16	Accumulator Injection (SXACC), Emergency AC Power (SXACP), Cool Down and Depressurization (SXOP1, SXOP2)
3-17	Reactor Vessel and Buildings (SXRVB), Medium Pipe Break (MPB), Control Rod (SYROD), Small Pipe Break (SPB), Secondary System Pipe Break (SYSSP), DC Power (SXDCP).

The fragility curves in Figures 3-14 through 3-17 can be used to make comparisons of the seismic capacity of individual systems (top events).

Based on a review of the system fragility curves, the following systems (top events) are ranked by their contribution to the plant fragility.

- Low pressure recirculation (SXLR1, SXLR2, SXLR4)
- Low pressure safety injection (SXL12)
- Auxiliary Feedwater System (SXAF1, SXAF3)
- Charging System (SXCHG)
- Cooldown and depressurization (SXOP1, SXOP2)
- Service Water System (SXSWS)

There are a number of systems whose median capacity is determined by the capacity of the surrogate component as indicated by the group of fragility curves whose median is 0.64g.

Dominant Events - The following are the dominant basic events/component failures that contribute to seismic risk:

- Loss of off-site power,
- Surrogate component in the following systems: CSF, RVB, SWS, DCP and ACP,
- Operator Error - failure to shift AFW pumps from CST to SWS.

Kewaunee Seismic Core Damage Frequency Results - All sequences in the Kewaunee SPRA have been quantified. Table 3-9 lists in rank order, the CDF for each seismic sequence and their total. Approximately 93 percent of the CDF is determined by 6 accident sequences. Figure 3-18 shows the contribution of individual accident sequences to the plant seismic CDF. Table 3-10 summarizes for the dominant sequences the system failures and the important basic events in these sequences. A discussion of the dominant sequences is provided below.

Sequence SCSF - This sequence corresponds to failure of the containment structure or the SG. Failure of either of these components leads directly to core damage. These structures were screened using the first column in EPRI NP-6041, which corresponds approximately to a PGA HCLPF level of 0.30g. As a result, these structures are modeled by the surrogate component. The seismic CDF associated with this sequence is 3.6E-06 per year. This is conservative, since the surrogate component is a conservative representation of the capacity of components at the Kewaunee plant.

Sequence SSWS - This sequence corresponds to failure of the SW System. Failure of the SW System is dominated by failure of the intake structure, which is modeled using the surrogate component. The intake structure was screened based on a HCLPF level of 0.30g. All other components in the SW System have median capacities of 0.66g PGA or greater. The seismic risk associated with failure of the SW System is 1.99E-06 per year which is conservative, since the capacity of the SW System is dominated by the surrogate component.

Sequence SRVB - This sequence corresponds to failure of the screenhouse, the auxiliary building, turbine building or the reactor vessel. Failure of any of these components is assumed to lead directly to core damage. These structures were screened using the middle column in EPRI NP-6041, which corresponds approximately to a PGA HCLPF level of 0.30g.

As a result, these structures are modeled by the surrogate component. The seismic risk associated with this sequence is 1.74E-06 per year. This is conservative, since the surrogate component is a conservative representation of the capacity of the structures at the Kewaunee plant.

Sequence SLSP01 - This sequence corresponds to the loss of off-site power and failure of the AFW System. Failure of the AFW System is attributed to failure of the operator to shift AFW pumps from the CST to SWS and failure of the surrogate element. The seismic risk associated with this sequence is 1.35E-06 per year.

Sequence SACP - This sequence corresponds to failure of emergency AC power system, including the diesel generators, and supporting mechanical and electrical equipment. All components in the AC power system have median capacities of 1.86g PGA or greater. As a result, failure of AC power is dominated by failure of the surrogate component. The seismic risk associated with failure of the AC power system is 1.26E-06 per year. This is conservative, since the system capacity is dominated by the surrogate element.

Sequence SDCP - This sequence corresponds to failure of DC power system, including failure of the station batteries, battery chargers, cable trays and electrical support equipment. All components in the DC power system have median capacities of 1.10g PGA or greater. As a result, failure of DC power is dominated by failure of the surrogate component. The seismic risk associated with failure of the DC power system is 3.48E-07 per year. This is conservative, since the system capacity is dominated by the surrogate element.

In the Kewaunee SPRA, small, medium and large LOCA events are modeled. As part of the Kewaunee SPRA, reactor vessel failure and seismically initiated small, medium and large LOCA events are considered. The frequency of these events is:

LOCA	Frequency
Small	2.23E-07/yr
Medium	7.68E-09/yr
Large	1.77E-07/yr
Reactor Vessel	1.74E-06/yr

Contribution of Ground Motions to Plant Risk - Figure 3-19 shows the relative contribution of PGA intervals to the seismic CDF. Approximately 76 percent of the CDF is contributed by PGA values in the range 0.25g to 0.65g.

Sensitivity Analysis

This section presents sensitivity analyses that were performed to examine the importance of individual factors to the Kewaunee SPRA results. The results of the Kewaunee SPRA, described in the previous section, are referred to as the Base Case. The results of the sensitivity evaluation are summarized in Table 3-11. The following sensitivity cases are evaluated:

No.	Case	Description
1	Seismic Hazard Input	The Lawrence Livermore National Laboratory (LLNL) mean seismic hazard curves are used to estimate the seismic CDF.
2	Non-Seismic (Random) Failures and Operator Errors	The purpose of this sensitivity evaluation is: 1) assess the significance of non-seismic (random) failures and operator actions in the SPRA, 2) determine the plant HCLPF based on the seismic capacity of structures and equipment.
3	Non-Seismic (Random) Failures	The purpose of this sensitivity evaluation is to assess the significance of random failures in the SPRA independent of other failures.
4	Operator Errors	The purpose of this sensitivity evaluation is to assess the significance of operator actions in the SPRA independent of other failures.
5	Surrogate Component	The contribution of the surrogate component to plant risk is determined by increasing the median capacity of the surrogate to 2.0g.
6	Operator action - Shift AFW from CST to SWS	The contribution of this operator action to plant risk is determined by assuming that this action is successful.
7	Seismic failure of SW System screens	The contribution of this component to plant risk is determined by increasing the medium capacity for the SW System screens to 3.0g.

Case 1: Seismic Hazard Input - To consider the sensitivity of the risk results to the seismic hazard input, the mean LLNL PGA hazard curves are used. The purpose of this evaluation is to assess the sensitivity of the ranking of the seismic sequences and their relative importance to plant risk. Since the LLNL mean PGA hazard curves are higher, as compared to the EPRI hazard, the seismic CDF based on the LLNL hazard is higher (1.15E-05). However, since the shape of the EPRI and LLNL mean PGA hazard results are not significantly different, the ranking of seismic sequences and the importance of individual components is the same.

There is one major difference between the EPRI and the LLNL hazard curves that needs to be addressed. The CDF estimate using the EPRI curve (Figure 3-1) considers ground motion up to 1.6g PGA whereas the LLNL hazard input (Figure 3-2) examines ground motion up to 1.0g PGA. The error induced by not specifically examining the effect of ground motion greater than 1.0g PGA would result in an estimated CDF of 1.32E-05, a 15% increase.

Case 2: Random (Non-Seismic) Failures and Operator Errors - This case examines the contribution of non-seismic failures and operator errors to the assessment of the seismic structural integrity of the plant and the CDF. To do this the random failure and operator error probabilities are set to 0.0. By eliminating these basic events, an estimate of the plant HCLPF based on the seismic integrity of structures and equipment is determined.

Figure 3-20 shows the plant level fragility curves for the Base Case and this case. In this case the plant HCLPF has increased to 0.26g, a relatively small change. This is attributed to the fact that random failures and operator errors contribute very little to the plant level fragility at ground motions less than 0.30g. There is a 10% change in the plant median capacity, from 0.38 to 0.43g PGA. This sensitivity evaluation also resulted in a reordering of sequences. The CDF for sequences SLSP01 and SACP decreases by 81% and 31%, respectively.

The CDF based on seismic failures is 9.27E-06. From this it is concluded that the random failures and operator error events in the SPRA make a 16% contribution to the plant CDF. The major contributor is the operator action to align SW System to the AFW pumps.

Case 3: Random (Non-Seismic) Failures - In this case two calculations are made to examine the contribution of non-seismic failures. The random failure probabilities are increased and then reduced by an order of magnitude (an overall change of a factor of 100). In the first calculation, when the random failure probabilities are increased by an order of magnitude, the CDF has increased by approximately a factor of 2 and the plant median capacity has decreased to 0.35g from 0.38g. By increasing the random failure probabilities by an order of magnitude, the HCLPF hs decreased to 0.10g. With the increase in the random failure probabilities, the HCLPF probability level of 0.01 is reached at a much lower ground motion level.

In the second calculation in this case in which the random failure probabilities are reduced by an order of magnitude, there is very little difference when compared to the base case. The CDF is reduced to 1.06E-05 from 1.10E-05 and the median capacity is unchanged. As a result of the lower random failure probabilities, the HCLPF increases slightly to 0.25g.

Case 4: Operator Errors - This case examines the contribution of operator errors to the seismic CDF. Two sets of calculations are performed in which the operator error probabilities are increased and then reduced by an order of magnitude (an overall change of a factor of 100). This adjustment applies to ground motions less than 0.36g. The results are nearly the same in both cases. When the operator error probabilities are decreased by one order of magnitude, the CDF, plant median capacity and HCLPF have not changed from the base case. Increasing the operator error probabilities by one order of magnitude results in a slightly higher CDF (5%), and a HCLPF that has reduced to 0.20g. The results of these sensitivity evaluations indicate the seismic risk results are relatively insensitive to the operator error probabilities.

Case 5: Surrogate Component - As described in Section 3.1.4, the surrogate component is used to model components in the plant that have been screened out. The parameters of the surrogate component, the median capacity and logarithmic standard deviation, are conservatively defined to represent those components not explicitly included in the SPRA. This sensitivity analysis considers the dependence of the SPRA results on the parameters of the surrogate components.

The median capacity of 0.64g PGA for the surrogate component is based on the use of the middle column in the EPRI seismic margins report, NP-6041. Using the middle column, components are screened out that have a HCLPF greater than approximately 0.30g PGA.

In this sensitivity analysis, the median capacity of the surrogate is set to 2.0g, making the surrogate a much stronger component and thus less likely to fail. Figure 3-21 shows a comparison of the Base Case plant level fragility and the plant fragility assuming the higher capacity for the surrogate component.

Given the higher median capacity for the surrogate, the CDF is 9.66E-06. However, the plant median capacity has remained at 0.38g. As described in Section 3.1.4, the parameters of the surrogate component are conservative. In addition, the use of the surrogate component on a system specific basis is an additional conservatism. The results of the Base Case and the sensitivity evaluation provide a bound on the SPRA results that accounts for the approach used to screen components.

Case 6: Operator Action to Transfer AFW from the CST to the SW System - In the event that the CST fails, the operator must transfer to the SW System as the source of water for the AFW pumps. To examine the importance of this operator action, the corresponding error probability is set to 0.0. The CDF is 9.95E-06 which represents a 9.5% reduction. There is a 10% change in the plant median capacity, from 0.38 to 0.42g PGA and the HCLPF is unchanged.

Case 7: SW System Screens - In the base case, failure of the SW System contributes 18% to the total seismic CDF In this case, the median capacity of the SW System screens is increased to 3.0g. The CDF is 1.07E-05 which represents a 3% reduction and the plant median capacity remains at 0.38g PGA The relatively small change in the CDF suggests that failure of the SW System may occur as a result of failure of one or more other components.

Summary

This section summarizes the results of the Kewaunee SPRA. The seismic CDF is 1.10E-05 which is considerably less than the internal events CDF (8.74E-05). The median capacity of the plant is 0.38g PGA, which is greater than a factor three times the SSE. The HCLPF for Kewaunee is 0.26g PGA or approximately two times the SSE. The HCLPF is based on an evaluation of seismic systems model for seismic failures only.

Based on sensitivity evaluations and a review of the SPRA results, the following conclusions are made:

1. There does not exist a single failure mode for Kewaunee that dominates the seismic CDF.

2. Failure of a surrogate component, which is a conservative measure of the capacity of components that were screened out, is for many systems the important mode of failure. Since the surrogate does not specifically model the failure of a particular component, this observation is a further reinforcement of the conclusion that there does not exist any component specific failure modes that dominate the seismic CDF.

3. Operator actions are not a major contributor to the seismic CDF or plant capacity.

4. Loss of off-site power is an important contributor to the seismic risk.

5. As a group random failures and operator actions are an important part of the seismic CDF. In a relative sense, variation in the random failure probabilities produced the largest change in the seismic CDF (a range corresponding to a factor of 2.5).

3.1.6 Analysis of Containment Performance

As stated in NUREG-1407, the purpose of this evaluation is to identify vulnerabilities that involve early failure of containment functions. The following sub-sections present the details of the evaluations that were performed including the results.

3.1.6.1 Basis for Analysis Scope

The scope of this analysis is based upon a review of the Level 2 analysis that was performed for the IPE as well as the specific issues presented in Section 3.2.6 of NUREG-1407.

3.1.6.2 Containment Structure and Components

The plant walkdown included a review of all penetrations and hatches, all isolation valves, the free standing steel containment, and the piping and conduit flexibility between the concrete containment wall and steel shell as well as either side of wall and shell and found everything to be within the screening criteria. Thus, all were assigned the surrogate element value for the analysis of containment performance.

3.1.6.3 Containment Safeguard Systems

A. Containment Safeguard Systems Included in the SPRA

The systems evaluated in the IPE and the IPEEE that are classified as containment safeguard systems (CSG) are described as follows.

1. Internal Containment Spray

The Internal Containment Spray (ICS) System is designed to spray cool water into the containment atmosphere following a design basis accident (DBA). The spray provides sufficient heat removal capability to maintain the post-accident containment pressure below its design value. In addition, the spray is effective in scrubbing fission products from the containment atmosphere. Sodium Hydroxide (NaOH) is added to the spray solution for pH adjustment. The resulting alkaline pH of the spray fluid enhances its ability to scavenge iodine fission products from the containment atmosphere and precludes the possibility of stress corrosion cracking of the stainless steel components that are exposed to the containment sump fluid.

2. Containment Air Cooling

The Containment Air Cooling (CAC) System is a subsystem of the Reactor Building Ventilation (RBV) System. The RBV System consists of several subsystems that operate together to cool and circulate containment air during all modes of plant operation; provide containment purge and vent capabilities; provide containment vacuum protection and post-LOCA hydrogen control.

During normal plant operation the CAC System provides the majority of air cooling and circulation in order to maintain containment air temperatures below 120°F. Post-LOCA, the CAC System is designed to remove sufficient heat from the containment vessel, following the initial pressure transient, to keep containment pressure from exceeding the design value of 46 psig at 268°F (100% relative humidity).

3. Containment Isolation

The design of Kewaunee includes three barriers to prevent the release of fission products to the environment: the fuel rod cladding, the Reactor Coolant System, and the containment. The capability of the containment to provide third barrier protection in the event of an accident in which one or both of the other two barriers is not maintained requires that containment integrity be established and maintained so as to limit the leakage of fission products to a low value.

Any flow of fluids that may contain fission products from the containment following an accident may be from one of the two sources: leakage from the containment structure and leakage through containment penetrations. The function of the Containment Isolation (CI) System is to provide for the automatic isolation of all fluid lines not required for accident mitigation.

B. Systems Analysis

The containment safeguard systems were included in the plant walkdowns described in Section 3.1.2.3. The components associated with these systems were evaluated and specific fragilities assigned as described in Section 3.2.4.

Seismic fault trees for the containment safeguard systems are developed and quantified in the same manner as described in Section 3.1.5. These fault trees are as follows.

SXICS -	Internal Containment Spray
SXFCH -	Containment Air Cooling
SXCI -	Containment Isolation

C. Analysis of Specific Issues

Backup Air Systems

The containment, containment penetrations and the containment safeguard systems do not rely on any backup air systems for proper operation. Also a key assumption in the SPRA, which is presented in Section 3.1.5.3, is that the Station and Instrument Air System (IAS), which is not classified as Seismic Class I, is assumed to fail at any seismic level. Loss of IAS causes all air operated valves in the CSG systems to fail in the safe position.

Relay Chatter

A relay evaluation was performed based on the guidance provided in NUREG-1407. The evaluation included relays associated with the actuation signals for the CSG systems and the emergency core cooling systems. The results of this evaluation are presented in Section 3.1.4.

3.1.6.4 Quantification and Results

This section describes the procedure used to perform the seismic Level 2 risk quantification and presents the results of the Kewaunee Level 2 SPRA.

Quantification Procedure

The same procedure is used for Level 1 and Level 2 analysis. The Level 1 procedure is described in Section 3.1.5.4. Of the 47 sequences defined in the seismic event trees (Figures 3-4 through 3-9), those with frequencies greater than 1.0E-07 are quantified in the Level 2 portion. These sequences represent over 96% of the total seismic core damage frequency.

For each of these sequences, the availability of the containment safeguards systems discussed in Section 3.1.6.3 is determined. In addition, availability of low pressure recirculation is also considered. In some cases, such as loss of AC power, one or more containment safeguards systems is considered unavailable due to failure of support systems. For the containment structure failure sequence (SCSF), containment safeguards are not considered at all. All other sequences are divided into several sequences based on containment safeguards system availability. This is the same method that was employed in the IPE and is presented in Section 4 of the IPE submittal (Reference 2).

Since each sequence corresponds to an IPE sequence, the containment analysis developed for the IPE is assumed to be applicable to the seismic analysis. The only exception to this is the SCSF sequence, which is considered a worst case containment failure, i.e., greater than 10% of volatile fission products released. The frequencies of each containment event tree end state are presented in Table 3-12. As in the IPE, a 48 hour mission time is used.

Quantification Results

System Fragilities - The fragility of the individual containment systems are presented in terms of system level fragility curves. The CSG systems, Containment Isolation (SXCI), Containment Air Cooling (SXFCH) and Internal Containment Spray (SXICS) are presented on Figure 3-22.

Table 3-13 shows the frequency of each release category due to seismic events. The seismic containment failure frequency is 6.24E-06 which is about the same as the containment failure frequency due to internal events (8.03E-06). The seismic containment failure frequency is 57% of the seismic core damage frequency. The containment failure median capacity and HCLPF for Kewaunee are 0.51 and 0.30g PGA, respectively. The median capacity is about four times the SSE and the HCLPF is two and a half times the SSE. The HCLPF is based on an evaluation of the seismic containment systems model that considers both random and seismic failures. No additional Level 2 vulnerabilities were discovered.

3.1.6.5 Containment Analysis Summary

The results of the evaluations performed indicated that the containment as well as the systems designed to ensure containment integrity are seismically sound and no vulnerabilities could be identified.

3.2 USI A-45, GI-131, and Other Seismic Safety Issues

GI-131 Flux Mapping Cart

This generic issue deals with mobile flux mapping carts designed by Westinghouse Electric Corporation that could roll off their guide rails during a seismic event. This issue is not directly applicable to Kewaunee. Although the cart is a Westinghouse design, it is not a moveable table as discussed in GI-131. The so-called ten path flux mapping frame at Kewaunee is stationary. As part of the issue, the lateral seismic capability of the ten path frame was investigated to ensure it could withstand seismic forces without lateral restraints. The dynamic analysis shows that the frame would easily sustain seismic forces without the aid of lateral restraints. This issue is thus considered resolved.

An overhead I-beam rail supporting a chain-fall is located directly above the 10-path assembly. The I-beam is cantilevered by a hinged support from a concrete wall. Administrative controls were implemented to restrain the hoist at the fixed end of the crane rail when not in use. This eliminates any interaction hazards to the ten path assembly.

Charleston Earthquake Issue

The NRC states in Generic Letter 88-20, supplement 4 that the Charleston Earthquake issue is subsumed in the IPEEE. WPSC has performed a seismic PRA for Kewaunee using the current LLNL and EPRI seismic hazard estimates. WPSC considers that it has fulfilled the requirements for this issue.

USI A-45 Shutdown Decay Heat Removal Requirements

Generic Letter 88-20, discusses resolution of USI A-45, "Shutdown Decay Heat Removal Requirements." The IPE submittal for Kewaunee (Reference 2) indicates that a complete evaluation of the decay heat removal capability as well as the possible resolution of the USI A-45 will be considered as part of the IPE for external events. This section outlines the analysis of the Kewaunee decay heat removal (DHR) capability, as required by the generic letter.

The Kewaunee IPE and IPEEE are an integrated look at core damage risk from all internal and external events including loss of decay heat removal. These examinations use a systematic approach to evaluate plant systems and components looking for vulnerabilities to severe accidents. Inherent to this approach is an evaluation of the potential for loss of decay heat removal capability.

NUREG-1289, "Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," Section 1.1, lists two criteria that must be met by systems used to remove decay heat. These criteria are (1) to maintain sufficient water inventory in the RCS to ensue adequate cooling of the fuel and (2) to provide the means for transferring decay heat from the RCS to an ultimate heat sink.

Systems Available for Decay Heat Removal

There are four possible methods by which decay heat can be removed from the reactor core:

- Secondary cooling through the steam generators with main feedwater or auxiliary feedwater providing the steam generator makeup.
- Bleed and feed cooling using the high-head SI pumps and pressurizer PORVs.
- ECCS injection and recirculation as provided by the SI and RHR Systems.
- Shutdown cooling mode of RHR operation after the RCS has been cooled down and depressurized to RHR conditions.

1. Secondary Cooling

Heat removal through the steam generators is the primary and preferred method of removing decay heat until the RHR System is placed in service. Effective heat removal using the steam generators requires circulation of primary coolant through the core with energy removal from the steam generators by use of steam relief to the condenser or atmosphere and steam generator makeup. Steam relief is not generally modeled for Kewaunee because of the many diverse means of steam removal. Following a reactor trip, steam is relieved to the condenser through four air-operated relief valves or to the atmosphere through four air-operated valves. If the MSIVs should fail closed, steam relief is possible through an air-operated PORV for each steam generator or through five safety valves on each steam generator, all of which are upstream of the MSIVs. In the event of loss of air, DC control power or instrument power, steam relief is assured through the five safety valves for each steam generator as they are not dependent on any support systems. With these many and diverse means of steam relief, it is assumed that the main reason for loss of steam generator cooling would be through loss of makeup capability. Two means of makeup to the steam generators were modeled: auxiliary feedwater (AFW) and main feedwater (MFW). A description of both of these systems follows:

Auxiliary Feedwater System

The AFW System consists of three completely redundant trains, each capable of feeding both steam generators. One train consists of a turbine-driven pump and the other two trains each consist of a motor-driven pump. Any one of the three pumps can supply adequate flow to meet the decay heat removal needs following any transient event. The normal water supply to the AFW System is the condensate storage tanks. These are backed up by the Service Water (SW) System. Valves in the AFW flow paths that have the potential to reduce AFW flow have their positions administratively controlled to prevent inadvertent valve misalignments after system maintenance.

Failure of the AFW pumps to start and run is minimized by ensuring the reliability of the pump driver. The off-site power supplies to the motor-driven pumps are backed up by emergency diesel generators to ensure reliability. The steam supply to the turbine-driven pump is from the Main Steam System upstream of the MSIVs. The motor valves on the steam supply lines are normally open and have indication in the control room that would alert the operators if the valves were misaligned. The steam supply valve that isolates steam from the turbine-driven AFW pump is a motor-operated valve powered from the station battery and is not dependent on AC power.

Kewaunee performed a reliability study using PRA techniques to identify potential improvements to the AFW System. The study, completed in October 1987, had several recommendations that were implemented and increased the reliability of the system. This included an increased awareness of the importance of the AFW System resulting in additional improvements to maximize system availability.

The following is a discussion of the major contributors to the AFW System reliability as derived from the IPE fault trees.

Initiating Event	AFW Failure Probability
Transients and LOCAs	2.7E-04
Tube rupture/steam line break	4.4E-03
Station blackout	1.1E-01

The failure probability of the AFW System varies for the spectrum of initiating events. Its support systems include only AC and DC power with SW providing a backup suction source in the event the condensate storage tanks are depleted. In the event that a station blackout occurs causing loss of the motor-driven AFW pumps, the turbine-driven pump is the only means for feedwater addition to the steam generators. The turbine-driven AFW pump is not dependent on AC power, as the steam admission valve to the pump requires DC power.

Important Hardware Failures:

- Random failure of turbine-driven AFW pump to run.
- Random failure of motor-driven AFW pumps to run.

Random failures of the turbine-driven auxiliary feedwater pump to run appear in a large portion of the results. Even through it is not dependent on AC power, the turbine-driven pump is not as reliable as the motor-driven pumps. Common cause events between the AFW pumps are not large contributors because of the diversity of the AFW pump drivers, one turbine-driven and two motor-driven pumps, and the redundancy provided by three pumps, all of which must fail to disable the AFW System.

Valve failures are not large contributors to the AFW System failure as the only valve required to change state is the steam admission valve to the turbine-driven AFW pump. All other valves required for system operation are in their required positions during normal operations or fail in their required positions on loss of support system.

Important Operator Actions:

In a station blackout or loss of instrument air, it is necessary to manually isolate the line connecting the condensate storage tanks and the condenser in order to prevent the draining of the condensate storage tanks. This is a very conservative assumption because a recent study shows that this draining is a very slow process, and the operator has ample warning before the tanks are drained.

Main Feedwater System

The MFW System is the primary source of makeup to the steam generator during normal operation. Following a reactor trip, feedwater is automatically isolated to the steam generators in order to prevent a rapid RCS cooldown. Feedwater can be easily recovered from the control room for initiating events that do not generate an SI signal by opening the feedwater regulating bypass valves. If an SI signal has been generated, MFW restoration is somewhat more complicated in that the MFW pumps must be restarted, the SI, containment isolation and feedwater isolation signals must be reset, the MFW containment isolation valves must be opened and the MFW bypass valves must then be opened.

The MFW System consists of two redundant pumps that can supply feedwater to both steam generators. Each pump is capable of supplying all of the necessary flow to remove decay heat after any transient event. The pump suction is supplied from the Condensate System. The discharge of the pumps is cross-connected and then flows through two parallel feedwater heater trains. The flow is again combined before it is split to flow through the feedwater regulating valves to the steam generators.

The following is a discussion of the major contributors to the MFW System reliability as derived from the IPE fault tree.

Initiating Event	MFW Failure Probability
Transients	2.8E-03
LOCAs	6.1E-03
Tube rupture	1.5E-02
Steam line break	1.5E-02

Since MFW is not a safeguards system, the pumps are not powered from a diesel backed bus and therefore are unavailable during a loss of off-site power. Loss of service water causes loss of lube oil cooling to the MFW and condensate pumps causing eventual failure of the pumps. Since the feedwater regulating and bypass valves are air-operated fail closed valves, loss of DC power and loss of instrument air both cause closure of all of the feedwater valves, failing feedwater. Local operation of the MFW valves is proceduralized for loss of heat sink. The different MFW System failure probabilities for initiating events that do fail feedwater and those initiating events that caused an SI signal to be generated is the difference in recoveries that the control room operators must perform to recover MFW as explained above.

Important hardware failures:

- Common cause failure of MFW System.
- Random failure of feedwater isolation valve.

Important operator actions:

The important operator action is the restoration of MFW for events in which it is isolated as a result of the initiating event but is still available. The actions associated with MFW restoration for those initiating events that do and do not generate an SI signal have been described above. Restoration of MFW is a relatively simple proceduralized action that is performed in the control room.

2. Bleed and Feed

Transients resulting in reactor trip employ secondary cooling as the primary mechanism for core heat removal. For accident scenarios in which secondary cooling cannot be established or maintained, decay heat is absorbed by the primary system causing RCS temperature and pressure to rise. In these accidents, the emergency procedures direct the operator to initiate bleed and feed cooling.

To perform decay heat removal via bleed and feed, the operators inject cool water to the RCS with the SI System and relieve steam from the RCS through the pressurizer PORVs. In this cooling mode, primary coolant is released through the PORVs into the pressurizer relief tank (PRT) resulting in RCS pressure reduction and decay heat removal. An evaluation was performed that indicates that only one PORV is required for successful bleed and feed operations at Kewaunee. SI injection in this mode maintains adequate RCS inventory as well as providing decay heat removal. A short description of the pressurizer PORVs and SI System follows.

The pressurizer PORVs are air-operated fail-closed valves that are used to prevent overpressure in the RCS. For bleed and feed operation, they are manually opened from the control room to allow flow from the pressurizer to the PRT.

Since the PORVs are air operated valves, they are dependent on instrument air and DC power to operate. The instrument air supply to containment passes through one air operated fail open containment isolation valve. A failure of this valve or the associated control circuit will cause the valve to close and isolate instrument air to containment, which results in the loss of continuous bleed and feed capability. The PORVs have air accumulators to allow limited valve operation in the event of a loss of instrument air, but for conservatism it is assumed that they do not have sufficient capacity to allow valve operation for the duration of the bleed and feed operation. Therefore, the PORVs are assumed to fail on a loss of instrument air event.

Safety Injection System

The SI System is used to inject water from the refueling water storage tank (RWST) into the RCS when the RCS pressure is greater than the shutoff head of the RHR pumps. The SI System consists of two redundant trains containing a pump and its associated valves.

The SI pump motors are powered from safeguards buses, which are backed up by diesel generators for reliability. The motor-operated valves that must operate to align the system for injection are also powered from safeguards power supplies. Motor-operated valves in the injection lines to the RCS cold legs are normally open with the breaker locked open to prevent inadvertent valve misalignment. The SI pump discharge is cross-connected such that either pump can supply cold leg injection or reactor vessel injection.

The SI pumps draw a suction off the boric acid tank (BAT) for the first few minutes of the injection phase of an event and then switch to the RWST when the lo-lo level alarm is reached on the BAT. When the RWST low level alarm setpoint is reached, the operators are instructed to begin the transfer from injection to recirculation mode.

The following is a discussion of the major contributors to bleed and feed reliability as derived from the IPE fault trees.

Initiating Event	Bleed and Feed Failure Probability
Loss of off-site power	2.7E-02
Transients	6.1E-03
Loss of one DC bus	6.3E-02
Steam line break	6.3E-03
Small LOCA	3.7E-03

As shown above, the availability of bleed and feed cooling is highly dependent on the initiating event. During normal transient events, bleed and feed is relatively reliable and its operation is principally dependent on operator action to initiate bleed and feed. For events involving loss of DC power or instrument air, the pressurizer PORVS fail closed, failing bleed and feed.

Loss of component cooling water (CCW) is assumed to fail the SI System as the SI pumps rely on CCW for lube oil cooling. Loss of SW has the same effect as loss of CCW as loss of SW results in loss of CCW.

Important hardware failures:

- Common cause failure of high pressure injection.
- Common cause failure of BAT interlock.

Important operator actions:

Hardware failures are insignificant contributors to the bleed and feed failure probability as most of the failure probability is from human error. The operator actions to initiate bleed and feed differ as to whether an SI signal has been generated by the initiating event. If an SI signal has not been generated, the operator must manually start the SI pumps and open the pressurizer PORVs, while if an SI signal has been generated, he must only verify an SI pump is running and then open the pressurizer PORVs.

The most important operator actions for bleed and feed are manually stopping the reactor coolant pumps (RXCPs) and manually establishing bleed and feed. Manually stopping the RXCPs is modeled based on the very conservative assumption that the heat input from the RXCPs is sufficient to speed up drying of the steam generators to the point at which there is insufficient time for manual initiation of bleed and feed. Manually establishing bleed and feed consists of opening the PORVs, and if they are not running already, starting the SI pumps.

3. ECCS Injection and Recirculation

During LOCA events, decay heat is removed from the RCS by the Emergency Core Coolant Systems (ECCS). The two systems from the ECCS that provide RCS inventory control are the SI and RHR Systems. Each system has two modes of operation. During the initial phase of injection, both systems inject into the RCS from the RWST. The SI pumps are high head low capacity pumps that inject into the RCS when the RCS pressure remains above the shutoff head of the RHR pumps. The RHR pumps are low head high capacity pumps that are used to prevent core damage for a design basis LOCA.

When low level in the RWST is reached, the operators are instructed to switch to the recirculation mode of ECCS. In recirculation, the RHR pump suction is shifted to the containment sump. Water is drawn from the containment, cooled in the RHR heat exchangers and discharged either back into the RCS or to the suction of the SI pumps depending on whether RCS pressure is above or below the shutoff head of the RHR pumps.

For high head recirculation, when RCS pressure is above RHR pump shutoff head, the RHR pump discharge is directed to the SI pump suction. The SI pumps then pump the water back into the RCS at high pressure. For low head recirculation, when the RCS pressure is below the RHR pump shutoff head, the RHR pumps discharge directly into the reactor vessel. Since the SI System is discussed under bleed and feed, it is not discussed here.

High Head Injection and Recirculation

The following is a discussion of the major contributors to high head injection and recirculation system reliability.

High Head Injection

The reliability of high head injection does not vary for the spectrum of initiating events considered in the IPE.

Initiating Event	HPSI Failure Probability
Interfacing system LOCA	1.29E-03
Other events	3.0E-03

Important hardware failures:

- Common cause failure of system.
- Common cause failure of BAT interlock.

The two largest contributors to the high head SI System failure probabilities ar. the suction valves from the BAT and RWST failing to open due to common cause. At Kewaunee, the SI pumps first draw suction from the BAT and then switch over to the RWST on low BAT level. If the BAT or RWST suction valves fail to open, a small 2-inch passive suction line from the RWST provides some limited protection for the SI pumps.

Important operator actions:

The only operator action is associated with system restoration after test and maintenance. This action contributes less then 3 percent to system reliability.

High Head Recirculation

The reliability of high head recirculation does not vary appreciably except for the loss of a DC bus, in which one complete train of safeguard systems fail.

Initiating Event	HPR Failure Probability
Loss of a DC bus	1.0E-01
Other events	6.0E-03

Loss of CCW is assumed to fail high head recirculation as the SI pumps rely on CCW for lube oil cooling and the RHR heat exchangers use CCW for their heat sink. Loss of SW has the same effect as loss of CCW as loss of SW results in loss of CCW.

Important hardware failures:

- Common case failure of high head recirculation.
- Common case failure of low head recirculation.
- Common case failure of low head injection.
- Plugging of the containment sump strainers.
- Random failure of CCW, RHR, and SI motor valves required during recirculation.

Important operator actions:

The single most important operator action is the failure of the operator to stop the RHR pumps when RCS pressure is above the RHR pump head. It is assumed that failure of this action within 30 minutes results in the failure of the RHR pumps.

Low Head Injection and Recirculation

The low head SI (RHR) System is used to inject water from the RWST or the containment sump into the RCS when the RCS pressure is low. The RHR System consists of two redundant trains each containing a pump and heat exchanger. The heat load from the heat exchangers is transferred to the CCW System. Each RHR train is capable of providing the necessary injection flow to prevent core damage for a design basis LOCA.

The RHR pump motors are powered from safeguards buses which are backed up by diesel generators for reliability. The motor-operated valves that operate to align the system for recirculation are also powered from safeguards power supplies. The RHR flow control valves downstream of the heat exchangers are air-operated and are dependent on instrument air, but they fail open on a loss of air and therefore do not fail their associated train. Since the RHR pumps are located in pits in the auxiliary building, it is assumed that room cooling is required for successful RHR pump operation.

Low Head Injection

The reliability of low head injection does not vary appreciably for the spectrum of events considered based on the IPE fault trees.

Initiating Events	LPSI Failure Probability
Large LOCA	6.9E-04
Small and medium LOCA	4.3E-04

Important hardware failures:

- Common cause failure of both RHR pumps.
- Common cause failure of actuation signals.

The largest contributors to RHR injection failure are common cause failures of the two RHR pumps to start and run. Since the only support systems that the RHR System requires during the injection mode of operation is AC and DC power, support system failures are not large contributors to system failure Valve failures do not contribute to system failure as there are no valves required to change state for successful RHR injection as the RHR vessel injection valves are open because of pressure locking/thermal binding concerns.

Important operator actions:

There are no significant operator actions associated with low head injection.

Low Head Recirculation

The reliability of low head recirculation used for small, medium, and large LOCA events is 2.8E-03.

Important hardware failures:

- Common cause failure of both RHR pumps.
- Plugging of the containment sump screen.
- Random and common cause failure of the CCW valves for the RHR heat exchangers.
- Random and common cause failure of the containment sump valves

The largest hardware failures that contributes to low head recirculation failure is the common cause failure to open of the CCW supply valves to the RHR heat exchangers and the containment sump valves. Failure of the CCW valves causes loss of heat sink for the RHR heat exchangers, which is assumed to cause failure of recirculation as heat cannot be removed from the RCS. Failure of the sump valves prevents flow from being established from the sump to the RHR System and thus failing recirculation. Area cooling failures contribute to low head recirculation failure as it has been assumed that the RHR pumps require room cooling during the recirculation mode of operation.

Important operator actions:

The single most important operator action is the failure of the operator to stop the RHR pumps for small and medium LOCA events where RCS pressure remains above RHR discharge pressure. It is assumed that failure of this action results in the failure of the RHR pumps due to excessive time in minimum flow conditions.

4. Residual Heat Removal Shutdown Cooling

The RHR System is discussed above in the injection and recirculation modes of operation. This section describes the RHR shutdown cooling mode of operation In this mode of operation, the RHR pumps draw suction from the A and B RCS hot legs and discharge the coolant through the RHR heat exchangers and back to B RCS cold leg. The heat load of the coolant is transferred to the CCW System from the RHR heat exchangers. This mode of RHR operation can only be entered after the RCS has been cooled and depressurized to 350°F and 425 psig.

The initiating events for which RHR is modeled, namely small LOCA and SGTR, have no affect on the RHR System, therefore only hardware failures and human error contribute to the system failure probability of 9.4E-03.

Important hardware failures:

- Random failure of RHR return valve to open.
- RHR System common cause failures.

The largest hardware contribution to RHR failure is the failure of the single RHR loop return valve to open. Since this is the single return valve, failure of this valve to open fails both trains of RHR. It was conservatively assumed that RHR could not be established through the low head injection valves to the reactor vessel.

Important operator actions:

The only significant operator action is the stopping of the RHR pumps that was discussed previously. This event contributes approximately 4.5 percent to system reliability.

Containment Cooling Considerations

There are four fan coil units (FCUs) inside the containment that draw air from around the unit and pass it through cooling coils to cool the air and condense any steam in the air, returning the condensate to a containment sump. They then discharge the cooled air to various areas in the containment. The fan motors are powered from a safeguards power supply that is backed up by an emergency diesel generator. The normal cooling medium for the FCU cooling coils is from the safeguards SW System.

The Internal Containment Spray (ICS) System consists of two pumps that are able to draw a suction from the RWST and a caustic standpipe to deliver a borated water-sodium hydroxide mixture to spray ring headers located in the containment dome. The ICS System operation consists of two phases: an injection and a recirculation phase. During the injection phase, the pumps draw a suction from the RWST and the caustic standpipe and deliver the mixture to spray rings in the containment dome. During the recirculation phase of operation, water is supplied to the suction of the ICS pumps from the RHR pumps, which draw their suction from the containment sump.

In all of the Kewaunee containment event trees, success or failure of recirculation is addressed before FCU or ICS success or failure. Credit is not taken for the FCUs removing decay heat from containment and condensing the water to return it to the containment sump. In the case of recirculation, failure of the RHR heat exchanger is assumed to result in failure of recirculation even though the RHR pumps could recirculate the water through containment where the heat could be removed by the FCUs. If recirculation fails, it is assumed that core damage occurs.

External Event Considerations

This section evaluates the effect of the external events considered in the IPEEE on the DHR capability at Kewaunee.

Seismic

Important SPRA Assumptions

The following are the major assumptions used in developing the SPRA models as discussed in Section 3.1 that have an impact on the DHR capability at Kewaunee.

- Offsite power is lost at low ground motion.
- The condensate storage tanks (CST) fail at low ground motion.
- The Instument Air System fails at low ground motion.

Secondary Cooling

Auxiliary Feedwater

In the seismic analysis AFW success is highly dependent on the diesel generators for support due to the loss of offsite power.

The CSTs are the primary supply of water to the AFW pumps. The CSTs are assumed to fail, requiring the operator to switch to the SW supply. This operator action plays a more important role in AFW success. This is, however, a simple task accomplished from the control room.

Main Feedwater System

The MFW System is not available due to the loss of offsite power to backup the AFW system in providing secondary cooling. The buses that provide power to the feedwater pumps are not backed up by diesel generators. Therefore, no credit is taken for MFW in the SPRA.

Bleed and Feed

In the seismic analysis, the Instrument Air System is assumed to fail at low ground motions. As discussed previously, bleed and feed requires the pressurizer PORVs, which are dependent on instrument air. Therefore, no credit is taken in the SPRA for bleed and feed. This is a conservative assumption in that a seismically qualified air supply would be available. It is uncertain, however, whether there would be sufficient capacity for the duration of the bleed and feed operation

ECCS Injection and Recirculation

The only significant finding associated with the ECCS Systems (SI and RHR) is the seismic capacity of the RHR heat exchangers (0.63g PGA), which is below the value for the surrogate (0.64g PGA). A sensitivity evaluation for this component provided results that were the same as the base case.

Fire

Kewaunee meets all the requirements of 10 CFR 50, Appendix R, other than exemptions approved by the NRC. This includes assuring that no fire can prevent operators from safely shutting down the plant. This includes the decay heat removal function.

For successful decay heat removal, it is necessary to have sufficient RCS inventory and, since steam generators are used to remove decay heat, sufficient steam generator inventory and relief capability. After cooldown to cold shutdown, it is necessary to have another means of long-term decay heat removal. Each of these is specifically addressed in the WPSC response to Appendix R.

RCS Inventory

The dedicated means of providing RCS inventory is the C charging pump. The alternate means is the B SI pump.

Steam Generator Relief

The primary means of steam relief is the A steam generator PORV. The alternate means is the B steam generator PORV.

Steam Generator Inventory

The dedicated means of providing steam generator inventory is the A AFW pump. The alternate means is the B AFW pump.

Long-Term Decay Heat Removal

The dedicated means of long-term heat removal is RHR train A. The alternate means is RHR train B.

Other decay heat removal options (MFW, bleed and feed, recirculation) are not credited for Appendix R purposes. They may be available however, depending on the location of the fire. Section 4 of this report provides a detailed description of which of these additional systems is available in which fire scenarios.

Support systems are also divided by train. In general, the same support systems are needed for fire events as for internal events. The A train supports dedicated train equipment and the B train supports alternate train equipment.

Other

A review of the analysis for the other external event indicated that none of the analyzed events had any impact on the DHR capability at Kewaunee.

Conclusions

In NUREG-1289, the two DHR requirements listed are:

a. Maintain sufficient water inventory in the Reactor Coolant System to ensure adequate cooling of the fuel.

b. Provide the means for transferring heat from the Reactor Coolant System to an ultimate heat sink.

With this definition in mind, loss of DHR becomes synonymous with core damage as there are no core damage sequences that do not involve loss of either one or both of the two requirements listed above. As identified above, there are many redundant and diverse means for DHR at Kewaunee. Several of the DHR systems and operator actions have to fail in combination to have an impact on the DHR capability at Kewaunee. With the performance of the IPE/IPEEE and the resultant overall CDF being acceptably low, WPSC considers that it has fulfilled the requirements of USI A-45.

USI A-17 Systems Interactions

The walkdowns explicitly considered USI A-17 interactions. Therefore USI A-17 is subsumed in the USI A-46 program. The seismic, fire, and flooding examinations for this IPEEE report incorporate the walkdown findings for USI A-17 related items. This will be presented in detail at a later date in the USI A-46 submittal.

USI A-40 Seismic Design Criteria, a Short Term Program

The one remaining element of USI A-40 concerns the evaluation of tanks. The IPEEE seismic evaluation does not explicitly evaluate large tanks. The USI A-46 assessments, however, do evaluate tanks for the concerns raised in USI A-40. Evaluation techniques incorporated the considerations established for the Seismic Margins Program thereby resolving the analytical concerns raised in USI A-40. The evaluation results will be presented at a later date in the USI A-46 submittal.

USI A-46 Verification of Seismic Adequacy of Equipment

The IPEEE project team performed the seismic PRA jointly with the USI A-46 evaluations. The selection of SPRA systems and components sought to retain commonality with the USI A-46 SSEL to the extent practical. Seismic walkdown teams gathered data for both evaluations simultaneously. The evaluation results are presented in Section 3.1.4 and Table 3-4 of this report. A more detailed report will be presented at a later date in the USI A-46 submittal.

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Table 3-1

Mean Seismic Hazard Curves for the Kewaunee Site

EPRI HAZARD CURVE	
Peak Ground Acceleration (g)	**Mean Annual Probability of Exceedence**
0.01	2.10E-03
0.11	1.10E-04
0.22	3.8E-05
0.50	5.90E-06
0.85	9.60E-07
1.12	2.70E-07
1.59	6.20E-08

LLNL HAZARD CURVE (NUREG-1488)	
Peak Ground Acceleration (g)	**Mean Annual Probability of Exceedence**
0.05	3.04E-04
0.07	1.77E-04
0.15	6.42E-05
0.25	2.75E-05
0.30	1.98E-05
0.40	1.14E-05
0.50	7.21E-06
0.66	4.04E-06
0.81	2.47E-06
1.02	1.41E-06

Table 3-2
SPRA Equipment/Component Fragilities

Component Description	Median PGA	β_c
Off-site Power	0.35	0.55
Reactor Internals	5.22	0.30
Diesel Generator Day Tank Block Walls	3.22	0.46
Circulating Water Intake and Discharge Piping	1.05	0.46
120/208V DISTRIBUTION PANEL	1.51	0.46
120/208V DISTRIBUTION PANEL	1.51	0.46
4160V SWGR	3.57	0.46
AC DISTR. CAB BRB-105	1.10	0.46
AC DISTR. CAB BRB-113(EXT)	3.45	0.46
AC DISTRIB CAB BRA-113 EXT	3.60	0.46
AC FUSE PANEL SD-100	3.20	0.46
ACCUMULATOR-ACCUMULATOR 1A	1.44	0.46
ACCUMULATOR-ACCUMULATOR 1B	1.44	0.46
BATTERY-SAFEGUARDS STATION BATTERY A	2.74	0.46
BATTERY-SAFEGUARDS STATION BATTERY B	2.74	0.46
BORIC ACID TANK 1A	1.16	0.46
BORIC ACID TANK 1B	1.16	0.46
BUS 1-51	3.47	0.46
BUS 1-52	3.47	0.46
BUS 1-61	3.47	0.46
BUS 1-62	3.47	0.46
BRA127 - 120/128VAC DISTR. CAB.	3.45	0.46
BRA113 118VAC DISTR. CAB.	1.51	0.46
BRA114 118VAC DISTR. CAB.	3.45	0.46
BRB102 125VDC MAIN DISTR. CAB.	1.10	0.46
BRB104 125VDC DISTR. CAB.	1.10	0.46
BRB113 118VAC DISTR. CAB.	3.45	0.46
BRB114 118VAC DISTR. CAB.	3.45	0.46
CABINET-DR102-LOGIC PANEL 1A 4KV	3.60	0.46
CABINET-DR103-LOGIC PANEL 1A 480V	3.60	0.46
CABINET-DR104-SEQ LOADING 1A PANEL	3.60	0.46
CABINET DR105-SEQ LOADING 1A PANEL	3.60	0.46
CABINET-DR106-SEQ LOADING 1A PANEL	3.60	0.46
CABINET-DR112-LOGIC PANEL 1B 4KV	3.60	0.46
CABINET-DR113-LOGIC PANEL 1B 480 V	3.60	0.46
CABINET-DR114 SEQ LOADING 1B PANEL	3.60	0.46
CABINET-DR115-SEQ LOADING 1B PANEL	3.60	0.46
CABINET-DR116-SEQ LOADING 1B PANEL	3.60	0.46
CHARGER-BATTERY CHARGER BRA108 125 VDC	3.45	0.46
CHARGER-BATTERY CHARGER BRB108 125 VDC	3.45	0.46

Table 3-2
SPRA Equipment/Component Fragilities

Component Description	Median PGA	β_c
CONTAINMENT FAN COIL UNIT 1A	2.85	0.46
CONTAINMENT FAN COIL UNIT 1B	2.85	0.46
CONTAINMENT FAN COIL UNIT 1C	2.85	0.46
CONTAINMENT FAN COIL UNIT 1D	2.85	0.46
COOLER-PUMP 1A GLAND SEAL COOLER	1.50	0.46
COOLER-PUMP 1B GLAND SEAL COOLER	1.50	0.46
DAMPER-OUTSIDE AIR INLET DAMPER TO DG ROOM 1A	1.46	0.46
DAMPER-OUTSIDE AIR INLET DAMPER TO DG ROOM 1B	1.46	0.46
DC FUSE PANEL SD-101	3.20	0.46
DG FUEL OIL DAY TANK 1A1	1.86	0.46
DG FUEL OIL DAY TANK 1A2	1.86	0.46
DG FUEL OIL DAY TANK 1B1	1.86	0.46
DG FUEL OIL DAY TANK 1B2	1.86	0.46
DIESEL AUX RELAY PANEL	1.57	0.46
DIESEL AUX RELAY PANEL	1.57	0.46
DIESEL GENERATOR 1A	3.67	0.46
DIESEL GENERATOR 1B	3.67	0.46
EDG-EXCITER CAB DR-101	2.22	0.46
EDG-EXCITER CAB DR-111	2.22	0.46
BRA101N FUSE CAB. (NEG)	3.45	0.46
BRA101P FUSE CAB. (POS)	3.45	0.46
BRA102 125VDC MAIN DISTR. CAB.	1.10	0.46
BRA104 125VDC DISTR. CAB.	1.10	0.46
BRB101N FUSE CAB. (NEG)	3.45	0.46
BRB101P FUSE CAB. (POS)	3.45	0.46
MIN. INTERRUPT. BUS 120/208 VAC DISTR. CAB.	1.10	0.46
FAN COIL UNIT-AUX BLDG BSMT FAN COIL 1C	1.51	0.46
FAN COIL UNIT-AU[illegible]DG BSMT FAN COIL UNIT 1D	3.47	0.46
FAN COIL UNIT-AU[illegible]RY BLDG BSMT 1A	1.45	0.46
FAN COIL UNIT-AUXILIARY BLDG BSMT 1B	1.45	0.46
FAN COIL UNIT-RESIDUAL HEAT PUMP PIT 1A	1.43	0.46
FAN COIL UNIT-RESIDUAL HEAT PUMP PIT 1B	1.43	0.46
HEAT EXCHANGER-RESIDUAL HEAT EXCHANGER 1A	0.63	0.46
HEAT EXCHANGER-RESIDUAL HEAT EXCHANGER 1B	0.63	0.46
INSTRUMENT BUS SUB-DIST CABINET IBSD-2	3.46	0.46
INSTRUMENT BUS SUB-DIST CABINET IBSD-4	3.46	0.46
INSTRUMENT BUS TRANSFORMER	3.41	0.46
INSTRUMENT BUS TRANSFORMER	3.41	0.46
INVERTER BRB-111 (INSTRUMENT BUS II)	3.45	0.46
INVERTER-BRA111 (INSTRUMENT BUS I)	3.45	0.46

Table 3-2
SPRA Equipment/Component Fragilities

Component Description	Median PGA	β_c
INVERTER-BRA112 (INSTRUMENT BUS IV)	3.45	0.46
INVERTER-BRB112 (INSTRUMENT BUS III)	3.45	0.46
MCC 1-52F (EXT)	3.48	0.46
MCC 1-52A	2.84	0.46
MCC 1-52B	1.87	0.46
MCC 1-52B (EXT)	3.34	0.46
MCC 1-52D	1.59	0.46
MCC 1-62C	2.67	0.46
MCC 1-62D	1.59	0.46
MCC 1-3352	3.34	0.46
MCC 1-62G	2.95	0.46
MCC 1-5262	2.84	0.46
MCC 1-52C	2.67	0.46
MCC 1-52E	2.94	0.46
MCC 1-52F	3.00	0.46
MCC 1-62A	2.84	0.46
MCC 1-62B	2.93	0.46
MCC 1-62BE	3.34	0.46
MCC 1-62E	1.87	0.46
MCC 1-62H	3.38	0.46
MCC 1-62J	1.08	0.46
Motor Starter - AFW10A/MV32027 AFW Train A X-over Valve	1.54	0.46
Motor Starter - AFW10B/MV32028 AFW Train B X-over Valve	1.54	0.46
Motor Starter - Turb Dvn AFW Pump Aux Lube Oil Pump	1.54	0.46
Motor Starter - BT3A/MV32078 SG A Blowdown Isol Valve A2	1.58	0.46
Motor Starter - BT3B/MV32080 SG B Blowdown Isol Valve B2	1.58	0.46
Motor Starter - MS102/32040 T/D AFW Pump Main Steam Isol	1.54	0.46
Motor Starter - SW502/MV32031 SW to Turb Dvn Aux FWP	1.54	0.46
PUMP-AUXILIARY FEEDWATER PUMP 1A	1.55	0.46
PUMP-AUXILIARY FEEDWATER PUMP 1B	1.55	0.46
PUMP-CHARGING PUMP 1A	1.51	0.46
PUMP-CHARGING PUMP 1B	1.51	0.46
PUMP-CHARGING PUMP 1C	1.51	0.46
PUMP-COMPONENT COOLING PUMP 1A	1.49	0.46
PUMP-COMPONENT COOLING PUMP 1B	1.49	0.46
PUMP-CONTAINMENT SPRAY PUMP 1A	1.51	0.46
PUMP-CONTAINMENT SPRAY PUMP 1B	1.51	0.46
PUMP-TURB DRIVEN-AUXILIARY FEEDWATER PUMP 1C	1.57	0.46
REACTOR TRIP BREAKERS RTA AND RTB - RD106	3.48	0.46
Relay Rack - RR121-RR125 Reactor Protection Train 'B' (1C165)	3.46	0.46

Table 3-2
SPRA Equipment/Component Fragilities

Component Description	Median PGA	β_c
Relay Rack RR126 - Engineered Safeguard Train B	3.46	0.46
Relay Rack - RR127 - Engineered Safeguard Train B	3.46	0.46
Relay Rack - RR128 - Engineered Safeguard Train B	3.46	0.46
Relay Rack - RR129 - Engineered Safeguard Train A	3.46	0.46
Relay Rack - RR130-RR134 Reactor Protection Train 'A'	3.46	0.46
Relay Rack - RR142 - Aux Relay Rack A	3.46	0.46
Relay Rack - RR143 - Aux Relay Rack Train A	3.46	0.46
Relay Rack - RR144 - Aux Relay Rack Train B	3.46	0.46
Relay Rack - RR147 - Aux Relay Rack B	3.46	0.46
Relay Rack - RR170-RR171 - Sub Distr AC Fuse Panel	1.11	0.46
Relay Rack - RR172 - Sub Distr AC Fuse Panel	1.11	0.46
Relay Rack - RR173 - Sub Distr AC Fuse Panel Normal	1.11	0.46
Relay Rack - RR174 - Sub Distr DC Fuse Panel Normal	1.11	0.46
Relay Rack - RR175 - Sub Distr AC Fuse Panel Train 'B'	1.11	0.46
Relay Rack - RR176 - Sub Distr DC Fuse Panel Train 'B'	1.11	0.46
RELAY-ELECT-BRA107 AUTOMATIC TRANSFER SWITCH	3.45	0.46
RELAY-ELECT-BRB107 AUTOMATIC TRANSFER SWITCH	3.45	0.46
RHR Pump 1A	1.51	0.46
RHR Pump 1B	1.51	0.46
RR101 - RR109/1E-0200-0208	3.46	0.46
SI PUMP 1A	1.51	0.46
SI PUMP 1B	1.51	0.46
SW PUMP 1A1	1.45	0.46
SW PUMP 1A2	1.45	0.46
SW PUMP 1B1	1.45	0.46
SW PUMP 1B2	1.45	0.46
SWITCHGEAR 4160V SWITCHGEAR BUS 6	3.57	0.46
Tank - Volume Control Tank	1.43	0.46
Transducer panel	1.57	0.46
Transducer panel	1.57	0.46
Transducer panel	1.57	0.46
Transducer panel	3.60	0.46
TRANSFORMER-STATION SERVICE TRANSFORMER 51	3.57	0.46
TRANSFORMER-STATION SERVICE TRANSFORMER 52	3.57	0.46
TRANSFORMER-STATION SERVICE TRANSFORMER 61	3.57	0.46
TRANSFORMER-STATION SERVICE TRANSFORMER 62	3.57	0.46
TURBINE BLDG FAN COIL 1B	1.42	0.46
TURBINE BLDG. FAN COIL 1A	1.42	0.46

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
145-661/SW-AUX FW PMP 1A AUX LUBE OIL PMP CONT PS
155-031/SWITCH-RESIDUAL HT PMP PIT FAN COIL UNIT 1A TS
155-032/SWITCH-RESIDUAL HT PMP PIT FAN COIL UNIT 1B TS
155-101/ SWITCH-AUX BLDG BSMT FAN COIL UNIT 1A TS
155-102/ SWITCH-AUX BLDG BSMT FAN COIL UNIT 1B TS
155-111/SWITCH-AUX BLDG MEZZ FAN COIL UNIT 1A TS
155-112/SWITCH-AUX BLDG MEZZ FAN COIL UNIT 1B TS
155-211/SWITCH-BATTERY RM FAN COIL UNIT 1A DSCH AIR TS
155-311/ SWITCH-AUX BLDG BSMT FAN COIL UNIT 1C TS
155-312/ SWITCH-AUX BLDG BSMT FAN COIL UNIT 1D TS
Accumulator - Service Water Turb Bldg Hdr 1A CV Accumulator
Accumulator - Service Water Turb Bldg Hdr 1B CV Accumulator
ACT- PRZR POWER RELIEF CV 1B SOL VLV
Actuator - ACC3A/MD32397 CRPA Recirc Damper A
Actuator - ACC3B/MD32371 CRPA Recirc Damper B
Actuator - ACC4/MD32374 Control Rm A/C Normal Recirc Dmpr
Actuator - Reactor Clnt Pmp 1A Seal Water Isol SV
Actuator - Reactor Clnt Pmp 1B Seal Water Isol SV
Actuator - Seal Wtr Return by - Pass Block CV
ACTUATOR SW TO CCW
ACTUATOR-ACC1A/MV32367 CONT ROOM FRESH AIR INLET DAMPER A
ACTUATOR-ACC1B/MV32368 CONT ROOM FRESH AIR INLET DAMPER B
ACTUATOR-ACC5/CD34007 NON-ACCIDENT FRESH AIR DAMPER
ACTUATOR-AFW PMP 1A DISCH CV SV
ACTUATOR-AFW PMP 1A FAN COIL UNIT COOLING VLV SV
ACTUATOR-AFW PUMP 1A DSCHG
ACTUATOR-AUX FW PMP 1A OIL COOLER COOLING WTR INLET SV
ACTUATOR-AUX FW PMP 1B OIL COOLER COOLING WTR INLET SV
ACTUATOR-BATT RM FAN COIL UNIT 1B SV
ACTUATOR-BATT RM FAN COOL UNIT 1A SV
ACTUATOR-CC610A/CV31127 RXCP A THERMAL BARR CC RETURN
ACTUATOR-CC610A/CV31127 RXCP A THERMAL BARR CC RETURN
ACTUATOR-CC610B/CV31128 RXCP B THERMAL BARR CC RETURN
ACTUATOR-CHG LINE TO AUX SPRAY/CVC15/33039
ACTUATOR-CHG LINE TO REGEN HX SV
ACTUATOR-CHG. LINE TO COLD LEG LOOP B STOP SV
ACTUATOR-COND MAKE-UP CV
ACTUATOR-D/G RM 1A INLET DMPR SV 1A1
ACTUATOR-D/G RM 1A INLET DMPR SV 1A2
ACTUATOR-D/G RM 1A OUTLET DMPR SV 1A1

Table 3-3 SPRA Equipment/Component List-Screened at Surrogate Level
Component Description
ACTUATOR-D/G RM 1A OUTLET DMPR SV 1A1
ACTUATOR-D/G RM 1B INLET AND RECIRC DMPR SV 1B1
ACTUATOR-D/G RM 1B INLET AND RECIRC DMPR SV 1B2
ACTUATOR-DG 1A OIL CLR WTR OUTL CV
ACTUATOR-DG RM 1B OUTLET DAMPER SV 1B1
ACTUATOR-DG RM 1B OUTLET DAMPER SV 1B2
ACTUATOR-DG1B OIL CLR WTR OUTLET SV
ACTUATOR-DIESEL GEN 1A OIL CLR WTR OUT SV
ACTUATOR-EXCESS LETDOWN HX INLET SV
ACTUATOR-FAN COIL UNIT 1B
ACTUATOR-HEADER 1A SHROUD CLG COIL A/B BYPASS SV
ACTUATOR-HEADER 1B SHROUD CLG COIL A/B BYPASS SV
ACTUATOR-HEADER 1C SHROUD CLG COIL C/D BYPASS SV
ACTUATOR-HEADER 1D SHROUD CLG COIL C/D BYPASS SV
ACTUATOR-HS2203A/MV32143 CONT ROOM AC UNIT A 3-WAY MIXING
ACTUATOR-HS2203B/MV32144 CONT ROOM AC UNIT B 3-WAY MIXING
ACTUATOR-LETDOWN & SEAL WTR BY-PASS BLOCK 1B SV
ACTUATOR-LETDOWN AND SEAL WATER
ACTUATOR-LTDN AND SEAL WTR BYPASS BLOCK SV
ACTUATOR-LTDN AND SL WTR BYPASS BLOCK 1A SV
ACTUATOR-LTDN AND SL WTR BYPASS DIVERT 1A SV
ACTUATOR-MN STM HDR 1A ISOL VLV AIR RELEASE SV 1A6
ACTUATOR-MN STM HDR 1A ISOL VLV AIR RELEASE SV 1A7
ACTUATOR-MN STM HDR 1A ISOL VLV AIR SPLY SV 1A1
ACTUATOR-MN STM HDR 1A ISOL VLV AIR SPLY SV 1A2
ACTUATOR-MN STM HDR 1B ISOL VLV AIR RELEASE SV 1B6
ACTUATOR-MN STM HDR 1B ISOL VLV AIR RELEASE SV 1B7
ACTUATOR-MN STM HDR 1B ISOL VLV AIR SPLY SV 1B2
ACTUATOR-MN STM HDT 1B ISOL VLV AIR RELEASE SV 1B1
ACTUATOR-PRZR PWR RLF PRESS SV 1A
ACTUATOR-PRZR PWR RLF PRESS SV 1B
ACTUATOR-RCTR HEAD VENT TO CNTNMT SV 49
ACTUATOR-RCTR HEAD VENT TRAIN A SV 33A
ACTUATOR-RCTR HEAD VENT TRAIN A SV 45A
ACTUATOR-RCTR HEAD VENT TRAIN B SV 33B
ACTUATOR-RCTR HEAD VENT TRAIN B SV 45B
ACTUATOR-RCTR HEAD VENT TO PZR RELIEF TNK SV 46
ACTUATOR-REAC CLNT CLD LEG LOOP B LTDN CV 1A
ACTUATOR-REAC CLNT COLD LEG LOOP B LETDOWN SV 1B
ACTUATOR-RELAY RM ISOL INLET DMPR

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
ACTUATOR-RELAY RM ISOL OUTLET DMPR
ACTUATOR-RELAY ROOM SPLY AND EXH DMPR SLND
ACTUATOR-SCRNHSE EXH FAN 1A DISCH DMPR TRAIN A SV
ACTUATOR-SCRNHSE EXH FAN 1B DISCH DMPR TRAIN B SV
ACTUATOR-SW AND RX BLDG 1A
ACTUATOR-SW PUMP HDR ISOL SV 1A
ACTUATOR-SW PUMP HDR ISOL SV 1B
ACTUATOR-SW TURB BLDG HDR 1A SV
ACTUATOR-SW TURB BLDG HDR 1B SV
ACTUATOR-SW1006C/CV31746 AUX BLDG BSMT FN COIL UNIT C INLT
ACTUATOR-SW1006D/CV31747 AUX BLDG BSMT FN COIL UNIT 1D INLT
ACTUATOR-SW1041A/CV31129 CR A/C CDSR A SW RTRN
ACTUATOR-SW1041B/CV31130 CR A/C CDSR B SW RTRN
ACTUATOR-SW1211A/SV33303 RHR PUMP PIT FAN COIL UNIT A INLET
ACTUATOR-SW1211B/SV33304 RHR PMP PIT FAN COIL UNIT B INLET
ACTUATOR-SW1221A/SV33307 AUX BLDG MEZZ FAN CL UNIT A INLET
ACTUATOR-SW1221B/SV33308 AUX BLDG MEZZ FAN COIL UNIT B INLET
ACTUATOR-SW1261/SV33778 CC PMP B FAN COIL COOLING WTR
ACTUATOR-SW802A/SV33313 AUX BLDG BSMT FAN COIL UNIT A INLET
ACTUATOR-SW802B/SV33302 AUX BLDG BSMT FN CL UNIT B INLET
ACTUATOR-TURB BLDG FAN COIL UNIT 1A CLG WTR SV
ACTUATOR-TURB BLDG FAN COIL UNIT 1B CLG WTR SV
ACTUATOR-TURB DRIVEN AFW PMP OIL COOLER SV
All Cable Tray and Conduit Raceways
All Piping System
CABINET-CR108 N.I.S. RACK NO.4
CABINET-CR109 N.I.S. RACK NO.3
CABINET-CR110 N.I.S. RACK NO.2
CABINET-CR111 N.I.S. RACK NO.1
CABINET-CR112 I.T.S. RACK NO. 4
CABINET-CR113 I.T.S. RACK NO. 3
CABINET-CR114 I.T.S. RACK NO. 2
CABINET-CR115 I.T.S. RACK NO. 1
CABINET-CR116 I.T.S. RACK NO. 2
CABINET-CR117 I.T.S. RACK NO. 1
CABINET-CR118 I.T.S. RACK NO.
CCW SURGE TANK
Control Room Ventilation Panel
Controller - Control Rm A/C Unit 1A Cooling Wtr TC
Controller - Control Rm A/C Unit 1B Cooling Wtr TC

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
CONTROLLER-BRC ACID TNK 1A HEATER TC
CONTROLLER-BRC ACID TNK 1A HEATER TC
CONTROLLER-BRC ACID TNK 1B HEATER TC
CONTROLLER-BRC ACID TNK 1B HEATER TC
CONTROLLER-COMP CLG PUMPS 1A/1B DSCH PC
CR107 MECHANICAL VERT PNL B
CV-CNTMT CLG SW RETURN HEADER 1C MV
DAMPER TAV 63A
DAMPER-CNTMT FAN COIL 1C EMERG DSCH DMPR
DAMPER-DG ROOM RECIRC SUPPLY FAN 1A DAMPER
DAMPER-DG ROOM RECIRC SUPPLY FAN 1B DAMPER
DAMPER-RBV150A/CD34131 CNTMT FAN COIL UNITB EMERG DISCH DMPR
DAMPER-RBV150B/CD34131 CNTMT FAN COIL 1B EMERG DISCH DMPR
DAMPER-RBV150D/CD34133 CNTMT FAN COIL UNIT D EMERG DISCH DMP
Detector - Neutron Flux Monitoring Detector Assg, Channel 2
Detector - Neutron Flux Monitoring Detector Assg, Channel 2
DG FUEL OIL STORAGE TANK 1A
DG FUEL OIL STORAGE TANK 1B
DG FUEL OIL TRANSFER PUMP 1A
DG FUEL OIL TRANSFER PUMP 1B
DIESEL GI [illegible] B FIRE DAMPER TAV60B
DIESEL GEN [illegible] 1B FIRE DAMPER TAV61A
Elec Equip - FR104 Steam Exclusion Protection Rack
Elec Equip - FR105 Steam Exclusion Protection Rack
Elec Equip - FR106 Steam Exclusion Protection Rack
Fan - Control RM Post Accid Recirc Fan 1A
Fan - Control RM Post Accid Recirc Fan 1B
FCU - Component Cooling Pump 1B
HEAT EXCHANGER-COMPONENT COOLING HEAT EXCHANGER 1A
HEAT EXCHANGER-COMPONENT COOLING HEAT EXCHANGER 1B
INSTRUMENT-AFW PUMP 1B DISCHARGE PRESSURE
INSTRUMENT-AFW PUMP 1B DISCHARGE PRESSURE (P-21024)
INSTRUMENT-MN STM HDR 1A CONTROLLED RLF SV
INSTRUMENT-MN STM HDR 1B CONTROLLED RLF SV
ISOL V-AUX BLDG SW HEADER B ISOLATION
Isolation Devices - Neutron Flux Monitoring Optical Isolator
Junction Box - Neutron Flux Monitoring Junction Brk Assg.
Junction Box - Neutron Flux Monitoring Junction Brk Assg.
JUNCTION BOX FOR STAION & INSTRUMENT AIR COMP. 1A TIMING RELAY
LD-2 CVC LETDOWN ISO (LCV-427)

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
LD-3 CVC LETDOWN ISO (LCV-428)
LD-300 CVC EXCESS LETDOWN/33172
MS HDR 1A CONTROLLED RELIEF CV SD3A ACCUMULATOR
MS HDR 1B CONTROLLED RELIEF CV SD3B ACCUMULATOR
PANEL-CARDOX CONT PNL DGA
PANEL-CARDOX CONT PNL DGB
POSITIONER - 31305
Positioner for 31103
Positioner for 31170
Positioner for 34072
Positioner for 34073
PRZR LEVEL TRANS 24031
PRZR RELIEF TANK
PUMP-AUX FEEDWATER PUMP 1A AUX LUBE OIL PUMP
PUMP-AUXILIARY FEEDWATER PUMP 1B AUXILIARY LUBE OIL PUMP
PUMP-CONTROL ROOM A/C CHILLER PUMP 1A
PUMP-CONTROL ROOM A/C CHILLER PUMP 1B
PUMP-TDAFW PMP AUX LUBE OIL PMP
Regulator - 1A for 31103
Regulator - 1A for 31103
REGULATOR-IA FOR 31109
REGULATOR-IA FOR 31110
REGULATOR-IA FOR 33875
REGULATOR-IA FOR 33876
REGULATOR-IA FOR 33877
REGULATOR-IA FOR 33878
REGULATOR-IA FOR 34072
REGULATOR-IA FOR 34073
REGULATOR-SCRN HSE EXH FAN 1A DISCH DMPR
Relay Rack - RR186 - ICCMS Train B
Relay Rack - RR187 - ICCMS Train A
RELIEF VLV-SAFETY TO ATMOS-STEAM GEN 1A
RELIEF VLV-SAFETY TO ATMOS-STEAM GEN 1A
RELIEF VLV-SAFETY TO ATMOS-STEAM GEN 1A
RELIEF VLV-SAFETY TO ATMOS-STEAM GEN 1A
RELIEF VLV-SAFETY TO ATMOS-STEAM GEN 1A
RELIEF VLV-SEAL WTR RETURN LINE TO PRESS. RLF TANK
RTD-REACTOR COOLANT LOOP A COLD LEG RTD
RTD-REACTOR COOLANT LOOP A HOT LEG RTD
RTD-REACTOR COOLANT LOOP B COLD LEG RTD

Table 3-3 SPRA Equipment/Component List-Screened at Surrogate Level
Component Description
RTD-REACTOR COOLANT PMP 1A THERM BARRIER RTD
RTD-REACTOR COOLANT PMP 1B THERM BARRIER RTD
SB1B PRESS. TRANS 21097
SCREEN-TRAVELING WTR SCRN 1A1
SCREEN-TRAVELING WTR SCRN 1B2
SCREENHOUSE EXH. FAN 1A
SIGNAL CNVTR-CHARGING LINE TO REGEN HT EXGR CV SOL VLV
Signal Converter - Neutron Flux Monitoring Wide Range Amplifie
Signal Converter - Neutron Flux Monitoring Wide Range Amplifie
SURGE TK LEVEL TRANSMITTER 24041
SV FOR LD2
SV: ACTUATOR-CHARGING LINE AUX SPRAY TO PRZR STOP SV
SW PMP HEADER 1A ISO CV
SW STRAINER 1A1
SW STRAINER 1A2
SW STRAINER 1B1
SW STRAINER 1B2
SW-AUX FW PMP 1A LUBE OIL PERMISSIVE PS
SW-AUX FW PMP 1B AUX LUBE OIL PMP CONT PS
SW-AUX FW PMP 1B LUBE OIL PERMISSIVE PS
SW-TD FW PMP AUX LUBE OIL PMP CONT PS
Switch - AFW PMP 1A Lube Oil Perm PS
Switch - AFW PMP 1B Lube Oil Perm PS
Switch - Battery Rm FCU 1B discharge Air TS
Switch - MN Stm Hdr 1A Controlled Relief PS
Switch - MN Stm Hdr 1B Controlled Relief PS
Switch - TDAFW Pmp Lube Oil Press Perm PS
Switch - TDAFW PMP Lube Oil Press Perm PS
Switch- Reac Clnt Pmp 1B Clg Wtr Rtrn FC
SWITCH-AFW PUMP 1A FAN COIL TS
SWITCH-REAC CLNT PMP 1A CLG WTR RTRN FC
SWITCH-TURB BLDG FAN COIL UNIT 1A TS
SWITCH-TURB BLDG FAN COIL UNIT 1B TS
TANK-REFUELING WATER STORAGE TANK
TERM BOX FOR CNTMT COOLING TRN A LOCKOUT RELAY
TERM BOX FOR CNTMT COOLING TRN B LOCKOUT RELAY
TERM BOX-TB1279 TB AUX RELAY BOX FOR SWITCHGEAR BUS 1-3
TERM BOX-TB1371-TB FOR SV33261, SV33270 FUTURE
Terminal Box - TB1282 - Aux Relay Box for Switchgear Bus 1-61
TRANSMITTER - CC PUMP DISCH PRESS XMTR

Table 3-3 SPRA Equipment/Component List-Screened at Surrogate Level
Component Description
TRANSMITTER - RXCP A SEAL WATER HI RANGE FLOW XMTR
TRANSMITTER - RXCP A SEAL WATER LO RANGE FLOW XMTR
TRANSMITTER - RXCP B SEAL WATER HI RANGE FLOW XMTR
TRANSMITTER - RXCP B SEAL WATER LO RANGE XMTR
TRANSMITTER - SG 1A LVL IND XMTR
Transmitter - SG 1A LVL Ind XMTR
TRANSMITTER - SG 1B LVL IND XMTR
Transmitter - SG 1B LVL Ind XMTR
Transmitter - Vol Cont Tank Ref Line P Xmtr
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1A L XMTR
TRANSMITTER-BORIC ACID TANK 1B L XMTR
TRANSMITTER-CHG PMPS DSCH TO REGEN HT EXGR F XMTR
TRANSMITTER-CNTMT PRESSURE NO 1A1 DIFF P XMTR
TRANSMITTER-CNTMT PRESSURE NO 1A2 DIFF P XMTR
TRANSMITTER-CNTMT PRESSURE NO 1A3 DIFF P XMTR
TRANSMITTER-CNTMT PRESSURE NO 1B1 DIFF P XMTR
TRANSMITTER-CNTMT PRESSURE NO 1B2 DIFF P XMTR
TRANSMITTER-CNTMT PRESSURE NO 1B3 DIFF P XMTR
TRANSMITTER-PRESS. TRANSMITTER P-21006
TRANSMITTER-PRESSURE TRANSMITTER P-21005 SW TRAIN A DISCH PRES
TRANSMITTER-PRESSURE TRANSMITTER PT-922 (INJECTION LINE A)
TRANSMITTER-PRESSURE TRANSMITTER PT-923 (INJECTION LINE B)
TRANSMITTER-PRESSURE TRANSMITTER SGIA PT-460
TRANSMITTER-PRESSURE TRANSMITTER SGIA PT-468
TRANSMITTER-PRESSURE TRANSMITTER SGIA PT-482
TRANSMITTER-PRESSURE TRANSMITTER SGIB PT-479
TRANSMITTER-PRESSURE TRANSMITTER SGIB PT-483
TRANSMITTER-PRESSURE TRANSMITTER WIDE RANGE RCS LOOP A (PT-419
TRANSMITTER-PRESSURE TRANSMITTER WIDE RANGE RCS LOOP A (PT-420
TRANSMITTER-PRESSURIZER RELIEF TANK PRESSURE TRANSMITTER PT-44
TRANSMITTER-PRZR LEVEL XMTR
TRANSMITTER-PRZR LEVEL XMTR 1A
TRANSMITTER-PRZR PRESSURE XMTR 1A
TRANSMITTER-PRZR PRESSURE XMTR 1B

Table 3-3 SPRA Equipment/Component List-Screened at Surrogate Level
Component Description
TRANSMITTER-PRZR PRESSURE XMTR 1C
TRANSMITTER-PRZR RELIEF TANK LVL XMTR
TRANSMITTER-RESID HT REML PMP 1A DSCH P XMTR
TRANSMITTER-RESID HT REML PMP 1B DSCH P XMTR
TRANSMITTER-RWST LVL XMTR
TRANSMITTER-RWST LVL XMTR
TRANSMITTER-RXCP A NO. 1 SEAL INS. FLOW XMTR
TRANSMITTER-RXCP B NO. 1 SEAL INS. FLOW XMTR
TRANSMITTER-SFTY INJ PMP 1A DISH F XMTR
TRANSMITTER-STM GEN 1A L XMTR
TRANSMITTER-STM GEN 1A L XMTR
TRANSMITTER-STM GEN 1A L XMTR (LT-461)
TRANSMITTER-STM GEN 1B L XMTR
TRANSMITTER-STM GEN 1B L XMTR
TRANSMITTER-STM GEN 1B L XMTR (LT-471)
TRANSMITTER-TEMP XMTR COLD LEG RCS LOOP (TE-450B)
TRANSMITTER-VOL CONTROL TANK LVL
TRANSMITTER-VOL CONTROL TANK LVL
Turbine Oil Storage Tank
VALVE- MTR OPER-COMPONENT COOLING HX 1A INLET
VALVE- SG A MSIV BYPASS
VALVE-CHECK-MAIN STEAM ISOLATION VALVE ASSEMBLY-GEN 1A
VALVE-CHECK-MAIN STEAM ISOLATION VALVE ASSEMBLY-GEN 1B
VALVE-CONTROL-AUX FEEDWATER PUMP 1B FLOW CONTROL VALVE
VALVE-CONTROL-CC610A/CV31127 RXCP A THERMAL BARRIER
VALVE-CONTROL-CC610B/CV31128 RXCP B THERMAL BARR COMP CLG RT
VALVE-CONTROL-CNTMT CLG SW RETURN HEADER 1A MV
VALVE-CONTROL-CNTMT CLG SW RETURN HEADER 1D MV
VALVE-CONTROL-CONDENSATE EMERGENCY MAKE-UP CONTROL STATION
VALVE-CONTROL-CONTAINMENT SUMP TO DH NO.1
VALVE-CONTROL-CONTAINMENT SUMP TO DH NO.1
VALVE-CONTROL-HEADE 1C SHROUD COOLING COIL C/D BYPASS
VALVE-CONTROL-HEADER 1A SHROUD COOLING COIL A/B BYPASS
VALVE-CONTROL-HEADER 1B SHROUD COOLING COIL A/B BYPASS
VALVE-CONTROL-HEADER 1D SHROUD COOLING COIL C/D BYPASS
VALVE-CONTROL-ICS201/CV31273 ICS RECIRCULATION TO RWST
VALVE-CONTROL-ICS202/CV31272 ICS RECIRCULATION TO RWST
VALVE-CONTROL-LOCA100B/CV31725 POST LOCA H2 TO RECOMBINER B
VALVE-CONTROL-LOCA201B/CV31727 POST LOCA HYD RECMBR B TO CNT
VALVE-CONTROL-MAIN STM CONTROLED RELIEF VALVE STM HDR 1B

Table 3-3 SPRA Equipment/Component List-Screened at Surrogate Level
Component Description
VALVE-CONTROL-MAIN STM CONTROLLED RELIEF VALVE STM HDR 1A
VALVE-CONTROL-PRESSURIZER TO PRESSURIZER RELIEF TANK
VALVE CONTROL-PRESSURIZER TO PRESSURIZER RELIEF TANK
VALVE-CONTROL-REAC COOLANT TO DH NO.1
VALVE-CONTROL-REAC COOLANT TO DH NO.1
VALVE-CONTROL-REACTOR COOLANT PUMP 1A SEAL WATER OUTLET
VALVE-CONTROL-REACTOR COOLANT PUMP 1B SEAL WATER OUTLET
VALVE-CONTROL-REGEN. HEAT EXCH. CHARGING LINE
VALVE-CONTROL-RHR8A/CV31114 RHR HEAT EXGR A FLOW CV
VALVE-CONTROL-RHR8B/CV31115 RHR HEAT EXGR B FLOW CV
VALVE-CONTROL-SA7003B/MV32148 HYDROGEN DILUTION TO CONTAINMN
VALVE-CONTROL-SEAL WTR INJECTION BYPASS BLOCK CV
VALVE-CONTROL-SERVICE WATER FROM DIESEL GEN, OIL COOLER
VALVE-CONTROL-SERVICE WATER HEADER ISOLATION
VALVE-CONTROL-SERVICE WATER TO TURBINE BUILDING
VALVE-CONTROL-SERVICE WATER TO TURBINE BUILDING
VALVE-CONTROL-SI101A/CV31247 SI PUMP MAKEUP TO ACCUMUL A
VALVE-CONTROL-SI101B/CV31248 SI PUMP MAKEUP TO ACCUM B
VALVE-MTR OPER-AFW10B/MV32028 AFW TRAIN B CROSSOVER VALVE
VALVE-MTR OPER-AUX FEEDWATER PUMP 1A
VALVE-MTR OPER-AUX FEEDWATER PUMP 1B
VALVE-MTR OPER-AUX FW PUMP 1A DISCH X-OVER MV
VALVE-MTR OPER-BORIC ACID FILTER (STA. 1)
VALVE-MTR OPER-BT2A/MV32077 S/9 A BLOWDOWN ISOL VALVE A1
VALVE-MTR OPER-BT2B/MV32079 S/G B SLOWDOWN ISOL VALVE B1
VALVE-MTR OPER-BT3A/MV32078 S/G A BLOWDOWN ISOL VALVE A2
VALVE-MTR OPER-BT3B/MV32080 S/G B BLOWDOWN ISOL VALVE B2
VALVE-MTR OPER-CC400A/MV 32119 COMP COOLING TO RHR HT EXCH A
VALVE-MTR OPER-CC400B/MV32120 COMP COOLING TO RHR HT EXCH B
VALVE-MTR OPER-CC600/MV32088 CC TO RXCPS AND EXCHS LD HX
VALVE-MTR OPER-CC601A/MV32084 COMP COOLING TO RXCP A
VALVE-MTR OPER-CC601B/MV32085 COMP COOLING TO RXCP B
VALVE-MTR OPER-CC612A/MV32086 RXCP A COMP CLG RETURN ISOL
VALVE-MTR OPER-CC612B/MV32087 RXCP B CC RETURN ISOL
VALVE-MTR OPER-CC6A/MV 32121 COMP COOLING HT EXCH A OUTLET
VALVE-MTR OPER-CC6B/MV32122 COMP COOLING HT EXCH B OUTLET
VALVE-MTR OPER-CVC211/MV32124 RXCP SEAL WTR RETURN ISOLATION
VALVE-MTR OPER-ICS2A/MV32406 CNTMT SPRAY PMP A SUCT FRM RWST
VALVE-MTR OPER-ICS2B/MV32407 CNTMT SPRAY PMP B SUCT FRM RWST
VALVE-MTR OPER-ICS5A/MV32066 CNTMT SPRAY PMP A DISCH ISOL

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
VALVE-MTR OPER-ICS5B/MV32068 CNTMT SPRAY PMP B DISCH ISOL
VALVE-MTR OPER-ICS6A/MV32067 CNTMT SPRAY PMP A DISCH ISOL
VALVE-MTR OPER-ICS6B/MV32069 CNTMT SPRAY PMP B DISCH ISOL
VALVE-MTR OPER-MS100A/MV32038 S/G A STM SPLY TO T/D AFW PUMP
VALVE-MTR OPER-MS100B/MV32039 S/G STM SUPPLY TO T/D AFW PMP
VALVE-MTR OPER-MS102/MV32040 T/D AFW PUMP MAIN STM ISOL
VALVE-MTR OPER-MS2B/32008 S/G MSIV BYPASS VALVE
VALVE-MTR OPER-PEN. NO. 14 N EL. 616'-11 15/16 INCH
VALVE-MTR OPER-PR1A/MV32089 PRESSURIZER PORV BLOCK VALVE
VALVE-MTR OPER-PR1B/MV32090 PRESSURIZER PORV BLOCK VALVE
VALVE-MTR OPER-R.W. INLET AT CHARGING PUMP
VALVE-MTR OPER-RHR11/MV32118 RHR DISCHARGE TO RCS LOOP B
VALVE-MTR OPER-RHR1A/MV32116 RCS LOOP A SUPPLY TO RHR PUMPS
VALVE-MTR OPER-RHR1B/MV32132 RCS LOOP B SUPPLY TO RHR PUMPS
VALVE-MTR OPER-RHR2A/MV32117 RCS LOOP A SUPPLY TO RHR PUMPS
VALVE-MTR OPER-RHR2B/MV32133 RCS LOOP B SUPPLY TO RHR PUMPS
VALVE-MTR OPER-RHR300A/MV32134 RHR PMP A SUPPLY TO SI PMP A
VALVE-MTR OPER-RHR300B/MV32135 RHR PMPB SUPPLY TO SI PMP B
VALVE-MTR OPER-RHR400A/MV32125 RHR PMP A SUPPLY TO ICS PMP A
VALVE-MTR OPER-RHR400B/MV32126 RHR PMP B SUPPLY TO ICS PMP B
VALVE-MTR OPER-SI208/MV32131 SI RECIRCULATION TO RWST
VALVE-MTR OPER-SI209/MV32130 SI RECIRCULATION TO RWST
VALVE-MTR OPER-SI2A/MV32104 BORIC ACID TANK OUTLET ISOLATION
VALVE-MTR OPER-SI2B/MV32105 BORIC ACID TANK OUTLET ISOLATION
VALVE-MTR OPER-SI300A/MV32111 RWST SUPPLY TO RHR PUMP A
VALVE-MTR OPER-SI300B/MV32112 RWST SUPPLY TO RHR PUMP B
VALVE-MTR OPER-SI302B/MV32101 RHR PUMP B INJN TO REAC VESSEL
VALVE-MTR OPER-SI350A/MV32102 CNTMT SUMP B SPLY TO RHR PMP A
VALVE-MTR OPER-SI350B/MV32103 CNTMT SUMP B SPLY TO RHR PMP B
VALVE-MTR OPER-SI351A/MV32113 CNTMT SUMP B SPLY TO RHR PMP A
VALVE-MTR OPER-SI4A/MV32109 RWST SUPPLY SI PUMPS
VALVE-MTR OPER-SI4B/MV32110 RWST SUPPLY TO SI PUMPS
VALVE-MTR OPER-SI5A/MV32107 SI PUMP A SUCTION ISOLATION
VALVE-MTR OPER-SI5B/MV32108 SI PUMP B SUCTION ISOLATION
VALVE-MTR OPER-TURBINE DRIVEN AUX. FEEDWATER PUMP
VALVE-MTR OPER-VOLUME CONTROL TANK - OUTLET
VALVE-RELIEF-CHARGING PUMP 1A DISCH RELIEF
VALVE-RELIEF-CHARGING PUMP 1B DISCH RELIEF
VALVE-RELIEF-CHARGING PUMP 1C DISCH RELIEF
VALVE-RELIEF-PRESSURIZER SAFETY RELIEF TO PRSZR RELIEF TANK

Table 3-3
SPRA Equipment/Component List-Screened at Surrogate Level

Component Description
VALVE-RELIEF-PRESSURIZER SAFETY RELIEF TO PRSZR RLF TNK
VALVE-RELIEF-SAFETY TO ATMOS-STEAM GEN 1B
VALVE-RELIEF-SAFETY TO ATMOS-STEAM GEN 1B
VALVE-RELIEF-SAFETY TO ATMOS-STEAM GEN 1B
VALVE-RELIEF-SAFETY TO ATMOS-STEAM GEN 1B
VALVE-RELIEF-SAFETY TO ATMOS-STEAM GEN 1B
VALVE-SI302A/MV32100 RHR PUMP A INJECTION TO REACTOR VESSEL
VALVE-SI351B/MV32114 CNTMT SUMP B SUPPLY TO RHR PUMP B
VALVE-SOLENOID-DG RM 1A DMPR CONTROL SV 1A3
VALVE-SOLENOID-DG RM 1A DMPR CONTROL SV 1A4
VALVE-SOLENOID-DG RM 1B DMPR CONTROL SV 1B3
VALVE-SOLENOID-DG RM 1B DMPR CONTROL SV 1B4
XMITTER-AUX FW TO STM GEN 1A F XMTR
XMITTER-AUX FW TO STM GEN 1B F XMTR

Table 3-4
EQUIPMENT OUTLIERS
IPEEE WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
Motor Control Centers MCC52F & MCC52FEXT.	Adjacent MCC's not bolted together, which may pose an interaction hazard based on relay chatter concerns.	Cabinet displacements during a design basis seismic event were determined. The evaluation concluded that the cabinets will not impact.
Diesel Generator Excitation & Control Cabinets DR101 & DR111.	Several fasteners on cast-in-place anchors were found missing. An overhead emergency light posed an interaction hazard to DR101.	Missing fasteners were replaced during the 1992 refueling outage and restraint installed on emergency light during 1993 outage.
Station Service Transformers 51, 52, 61 & 62.	Transformer cabinets were found anchored to the floor with friction clips, which are considered undesirable according to A-46 walkdown guidelines.	A design change was initiated to have the transformer cabinet bases welded to embedded floor channels. Transformers 51 and 52 were modified during the 1994 refueling outage. Transformers 61 and 62 are scheduled for modification during a later refueling outage.
Relay Racks RR186 & RR187.	The relay racks are not bolted to adjacent panels, which may pose an interaction hazard based on relay chatter concerns.	An engineering support request was initiated to have the racks bolted together to eliminate the concern. A schedule for completion has not been determined.
Reactor Trip Breaker Cabinet RD106.	Several anchor bolts which connect cabinet to embedded channel were found missing.	A design change was initiated to have the cabinet sections welded to the embedded channel in lieu of installing bolts. Work was completed during the 1993 refueling outage.

Table 3-4
EQUIPMENT OUTLIERS
IPEEE WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
SI Pump B Suction Isolation Valve SI5B.	One leg of a Unistrut frame is within 1/2" of the valve motor, which may present an interaction hazard.	IE Bulletin 79-14 pipe stress evaluation determined that displacement of the pipe and valve is approximately 1/8". Issue considered resolved.
Main Steam Header 1A Controlled Relief Valve SD3A.	Valve actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
SI Pump Makeup Valve SI101B to Accumulator.	Actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
Aux Feedwater Pump Lube Oil Pressure Switches 16016, 16019 & 16085.	All three switches identified as Mercoids, which are considered outliers for the A-46 program.	A design modification was previously initiated to have the switches replaced for other reasons. Modification completed in 1993.
Flux Mapping Transfer Cart (GI-131 Issue).	Two concerns identified; (1) lateral restraints for the 10-path assembly frame were never installed, and (2) chain hoist on overhead rail identified as a possible interaction hazard to 10-path assembly.	It was determined by analysis that lateral restraints are not required to support 10-path assembly under seismic loads. Administrative controls were implemented to restrain hoist at the fixed end of crane rail and required to be functional when not in use.
Overhead Fluorescent Lights.	Generic problem throughout safety-related areas of the plant. S-hooks on the chains supporting the lights are not closed, presenting a possible interaction hazard to equipment below.	A plant walkdown was conducted during the 1994 refueling outage to pinch the S-hooks closed.

Table 3-4
EQUIPMENT OUTLIERS
IPEEE WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
Emergency Lights.	Some of the lights were found to not have seismic restraints installed, presenting a possible interaction hazard to equipment below.	Lights and battery units strapped to supports as required during 1994 refueling outage.
480V Switchgear Bus 62.	An empty spare breaker cabinet was used for parts storage, presenting a possible interaction hazard on the basis of relay chatter concerns.	Maintenance department notified of problem. Spare parts were removed and all other spare breaker cabinets were inspected for similar problems during 1993 refueling outage.
Control Room Ceiling.	Aluminum ceiling diffuser panels were considered as a possible hazard to operators if the diffusers were to dislodge from T-bar supports.	An engineering support request was initiated to have the diffuser panels tie-wrapped to the T-bar supports. A schedule for completion has not been determined.
Control Room Vertical Panel C.	Rear doors on panel could not be latched shut due to interference with cables which extend from rear of cabinet. Unlatched doors present possible interaction hazard on basis of relay chatter.	It was determined that the cables were temporarily in place to support radiation monitoring modifications. Doors could not impact with cabinet because of cable interference. Operations department agreed to latch doors shut following completion of work during 1994 refueling outage.
All Equipment	Possible interaction hazards due to loose or unrestrained portable equipment.	Plant procedure GNP 1.31.1 drafted to provide guidelines for control of portable equipment. Full implementation occurred June 1, 1994.

TABLE 3-5
COMPONENTS AND STRUCTURES USED TO DETERMINE PLANT STATUS FOLLOWING A SEISMIC EVENT

Containment Building
Reactor Pressure Vessel
Reactor Pressure Vessel Supports
Switchyard Ceramic Insulators
Auxiliary Building
Turbine Building
Screenhouse
Steam Generators
Steam Generator Supports
Pressurizer
Pressurizer Supports
Reactor Coolant Pumps
Reactor Coolant Pump Supports
Control Rod Drive Mechanism
Reactor Core Upper Internals
Reactor Coolant System Piping
Reactor Coolant Pump Seals
Secondary Side Piping and Supports

TABLE 3-6
DESCRIPTION OF SEISMIC EVENT TREE NODES

EVENT TREE NODE	DESCRIPTION
FRONTLINE SYSTEMS	
SXACC	Safety Injection Accumulators
SXLI1, SXLI2	Low Pressure Injection System
SXLR1, SXLR2, SXLR3	Low Pressure Recirculation System
SXHI0, SXHI2, SXHI3	High Pressure Injection System
SXAF1, SXAF3	Auxiliary Feedwater System
SXCHG	Chemical and Volume Control System
SXIS1	Main Stream Isolation
SXOP1, SXOP2	Operator Cooldown and Depressurize RCS
SUPPORT SYSTEMS	
SXRPS	Engineered Safeguards Actuation System Including 120V AC Instrument Power
SXCCW	Component Cooling Water System
SXSWS**	Service Water System
SXDCP**	125V DC Electrical System
SXACP**	4160V AC Electrical System, 480V AC Electrical System, Emergency Diesel Generators

** Fault trees used in the seismic initiating events tree Figure 3-4

Table 3-7

Kewaunee SPRA Results

Result	Value
Seismic Core Damage Frequency	1.10E-05
Plant Median Capacity (PGA(g))	0.38g
Plant High Confidence of a Low Probability of Failure (HCLPF) (PGA(g))	0.23

Table 3-8

Summary of the Plant Level Fragility

Conditional Probability of Failure	PGA (g)
0.05	0.293
0.15	0.344
0.25	0.364
0.50	0.377
0.75	0.452
0.85	0.512
0.95	0.611
HCLPF[1]	0.228

[1] Assuming the plant fragility curve is lognormal, the HCLPF corresponds to a 0.011% probability of failure

Table 3-9

Contribution of Individual Accident Sequences Towards the Total Plant Risk

Seismic Sequence[1]	Mean Annual Core Damage Frequency	Percent
SCSF	3.60E-06	32.70
SSWS	1.99E-06	18.08
SRVB	1.74E-06	15.76
SLSP01	1.35E-06	12.25
SACP	1.26E-06	11.40
SDCP	3.48E-07	3.16
SRDF	2.09E-07	1.90
SSLB06	1.08E-07	0.98
SLLO04	9.34E-08	0.85
SLLO19	7.65E-08	0.69
SSLO09	4.90E-08	0.44
SLLO01	4.04E-08	0.37
SLLO02	[illegible]E-08	0.24
SSLO11	[illegible]08	0.21
SLLO03	[illegible]4E-08	0.16
SSLO12	1.58E-08	0.14
SSLO14	1.46E-08	0.13
SSLO01	1.35E-08	0.12
SSLO02	9.17E-09	0.08
SSLO04	8.15E-09	0.07
SSLO10	6.77E-09	0.06
SSLO20	5.03E-09	0.05
SMLO09	4.70E-09	0.04
SLSP02	3.25E-09	0.03
SSLB05	2.91E-09	0.03

Table 3-9 (cont.)

Contribution of Individual Accident Sequences Towards the Total Plant Risk

Seismic Sequence[1]	Mean Annual Core Damage Frequency	Percent
SMLO01	1.29E-09	0.01
SSLB04	1.08E-09	0.01
SMLO02	8.68E-10	0.01
SMLO05	6.35E-10	0.01
SSLO13	4.03E-10	0.00
SSLB01	3.40E-10	0.00
SSLO03	2.34E-10	0.00
SSLB02	2.28E-10	0.00
SSLB03	2.28E-10	0.00
SSLO08	1.77E-10	0.00
SSLO18	1.53E-10	0.00
SMLO04	1.48E-10	0.00
SSLO07	6.57E-11	0.00
SSLO17	5.90E-11	0.00
SSLO06	4.44E-11	0.00
SSLO05	4.38E-11	0.00
SSLO16	4.08E-11	0.00
SSLO15	4.07E-11	0.00
SMLO03	2.22E-11	0.00
SMLO08	1.30E-11	0.00
SMLO06	4.20E-12	0.00
SMLO07	3.24E-12	0.00
Total	1.10E-05	100.00

1 See the seismic event trees in Figures 3-4 through 3-9.

Table 3-10
Dominant Sequences

Sequence	Description	Important Components
SCSF	Failure of the Containment or Steam Generator. These components have been screened out and are modeled by the surrogate element.	Surrogate Component
SSWS	Failure of the service water system occurs due to failure of the Intake Structure valves (air or motor operated, check valves, pumps, etc.)	Intake Structure (Surrogate Component). All other components have a median capacity of 0.95g or greater.
SRVB	Failure of the Screenhouse, Auxiliary Building, Turbine Building, or Reactor Vessel. These structures have been screened out and are modeled by the surrogate element.	Surrogate Component
SLSP01	Loss of off-site power, and the Auxiliary Feedwater System.	Operator Error Surrogate Component
SACP	Failure of the emergency AC power system occurs, which includes the diesel generators and supporting equipment.	Surrogate Component All components have a median capacity of 1.05g and higher.
SDCP	DC power system fails which includes the station batteries, diesel generators, cable trays, etc.	Surrogate Component All components have a median capacity of 1.05g and higher.

Table 3-11

Summary of Kewaunee SPRA Sensitivity Analysis Results

Case	Description	Core Damage Frequency	Plant Median (g)	HCLPF (g)
Base Case	EPRI Seismic Hazard Curve Used to Estimate CDF	1.10E-05	0.38	0.23
1	LLNL (NUREG-1488)[1] Seismic Hazard Curve Used to Estimate CDF	1.15E-05	0.38	0.23
2	Random Failure and Operator Action Probabilities Set to 0.0	9.27E-06	0.43	0.26
3	Random Failure Probabilites Increased By an Order of Magnitude	2.35E-05	0.35	0.10[2]
	Random Failure Probabilities Reduced By an Order of Magnitude	1.06E-05	0.38	0.25
4	Operator Action Probabilities Increased By an Order of Magnitude	1.15E-05	0.38	0.20
	Operator Action Probabilities Reduced By an Order of Magnitude	1.10E-05	0.38	0.25
5	Surrogate Component Capacity Set to 2.0g	9.66E-06	0.38	0.23
6	Operator action - Shift AFW from CST to SWS probability set to 0.0	9.95E-06	0.42	0.23
7	SW System screen capacity set to 3.0g	1.07E-05	0.38	0.24

[1] Plant model is the same as the base case.

[2] See text for a discussion of this HCLPF estimate.

Table 3-12

CET End States and Frequencies

CET End State	End State Frequency	Release Catagory
HAAAA	1.68E-07	S
HAAFA	3.96E-09	S
HAFAA	8.76E-09	S
HAFFA	1.68E-10	S
HFFAA	1.16E-06	S
HFFFA	3.03E-06	A
HAAAF	4.86E-09	G
HAFAF	2.51E-10	G
HFFAF	1.42E-07	G
HFFFF	7.48E-07	G
LFFFF	5.34E-06	U

Notes:

1. End States are given the following identifier:

   PLIFC Where:

   P = H for high pressure (>400 Psia) reactor vessel failure,
   L for low pressure vessel failure.

   L = A if low pressure recirculation is available, F if it has failed.

   I = A if containment spray is available, F if it has failed.

   F = A if containment fan coil units are available, F if they have failed.

   C = A if containment is isolated, F if isolation has failed.

2. Release catagories are defined in Table 3-13.

TABLE 3-13

KEWAUNEE NUCLEAR PLANT SPRA
AIRBORNE RELEASE CATEGORIES AND PROBABILITIES

Release Category	Definition	Frequency	Conditional Probability[1,2]
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful)	1.35E-06	0.13
U	Containment failure prior to vessel failure with noble gases and more than 10% of the volatiles released (containment isolation impaired)	5.34E-06	0.50
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired)	8.96E-07	0.08
A	No containment failure within 48 hr mission time, but failure could eventually occur without accident management action; noble gases and less than 0.01% volatiles released	3.03E-06	0.29

NOTES:

1. Conditional probability of release category given core damage.
2. Core damage frequency for Level 2 = 1.06×10^{-5}/yr.

FIGURE 3-1

EPRI Seismic Hazard Curve
for the Kewaunee Nuclear Power Plant Site

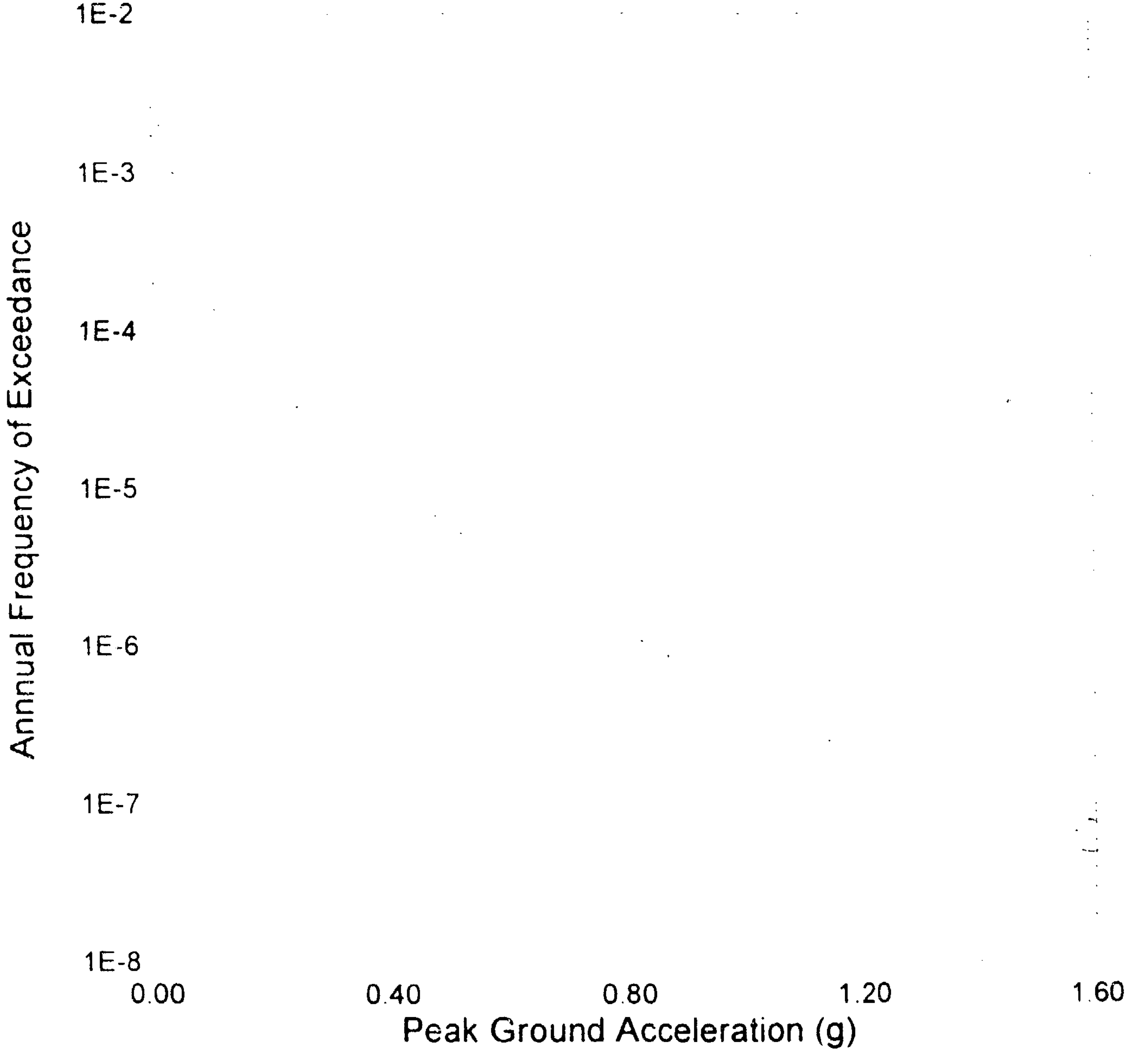


FIGURE 3-2

LLNL (NUREG-1488) Seismic Hazard Curve
for the Kewaunee Nuclear Power Plant Site

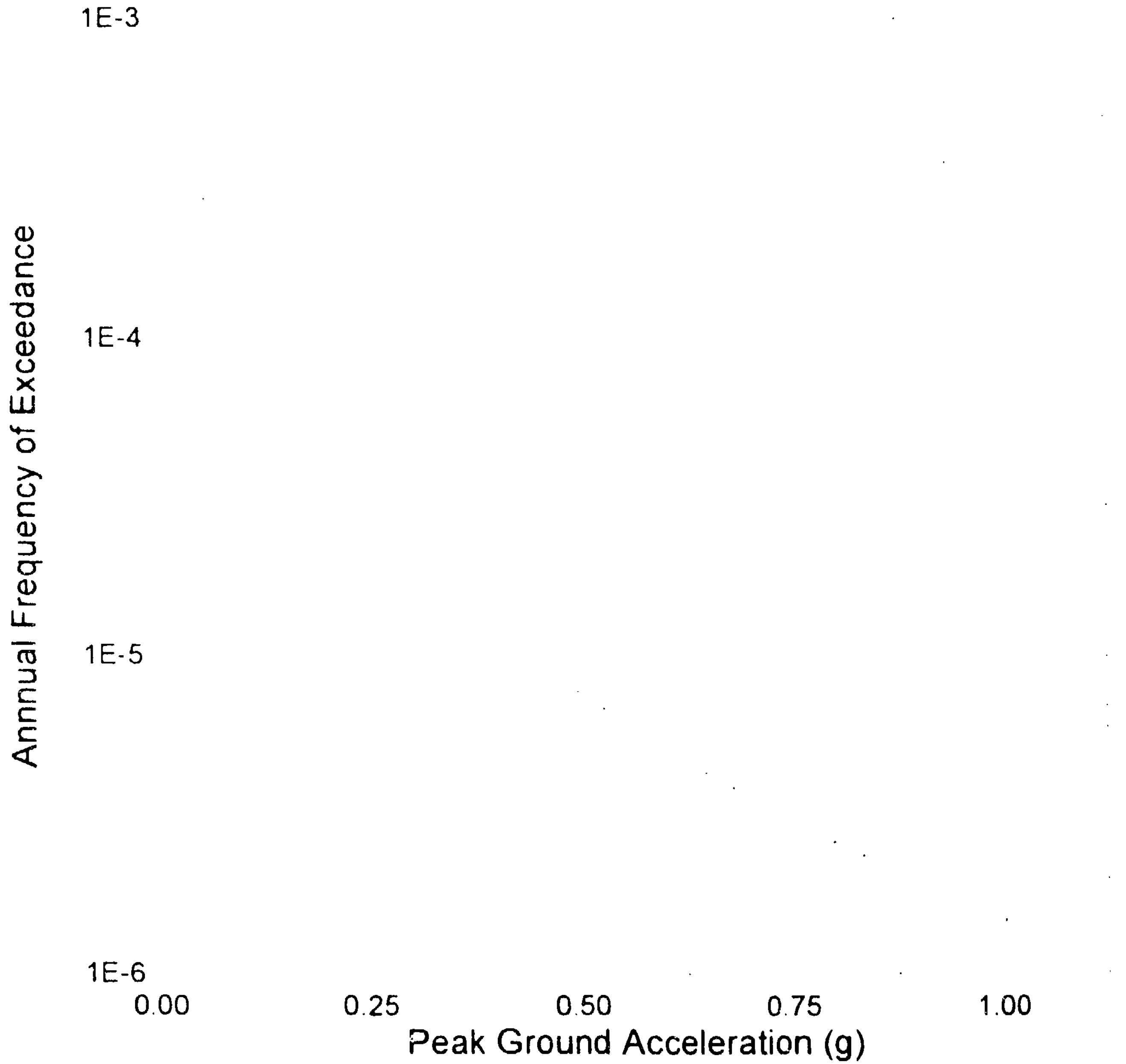


Figure 3-3
Small and Medium B eak LOCA
(Figure 3.6 - NUREG/CR-4840)

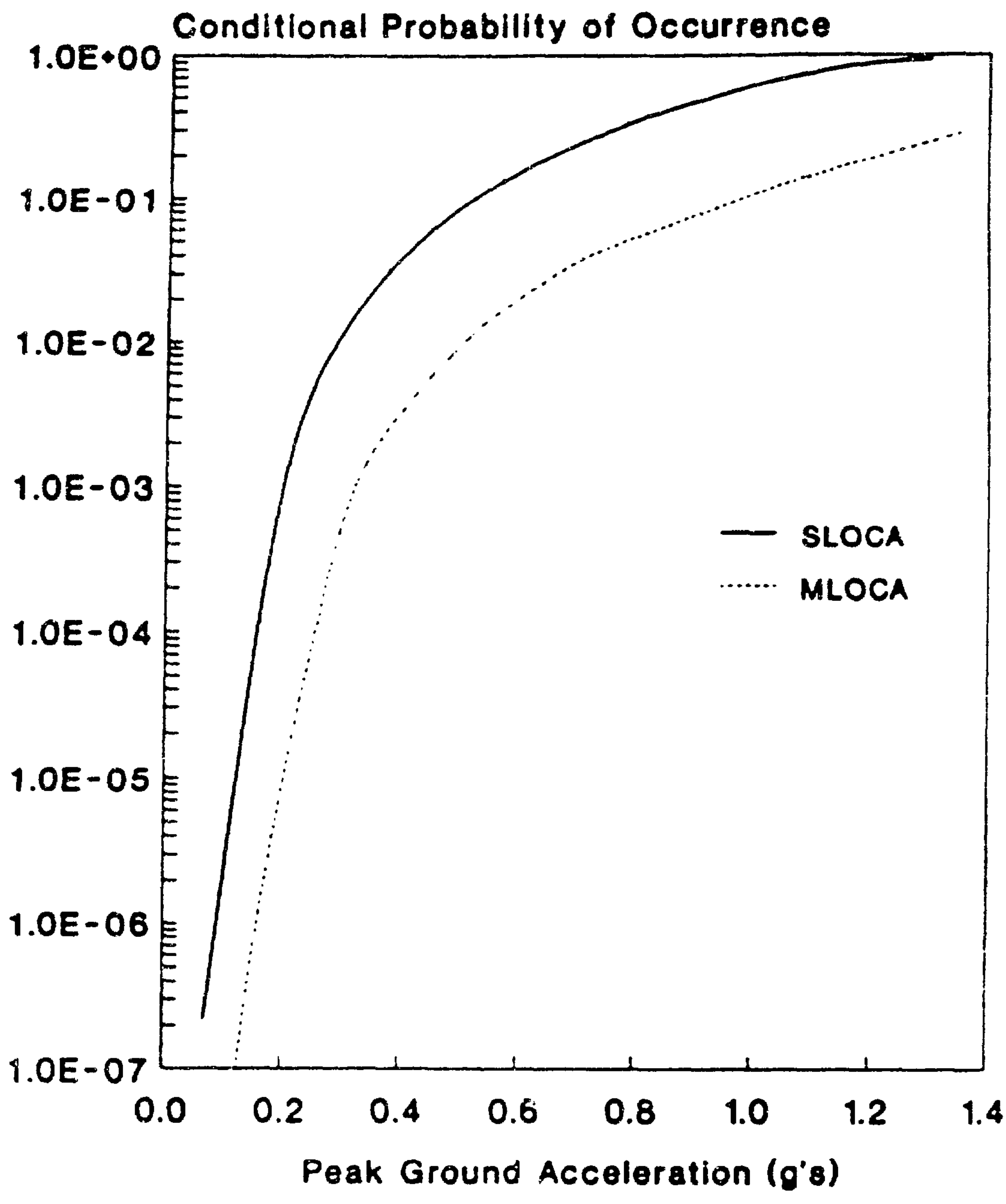
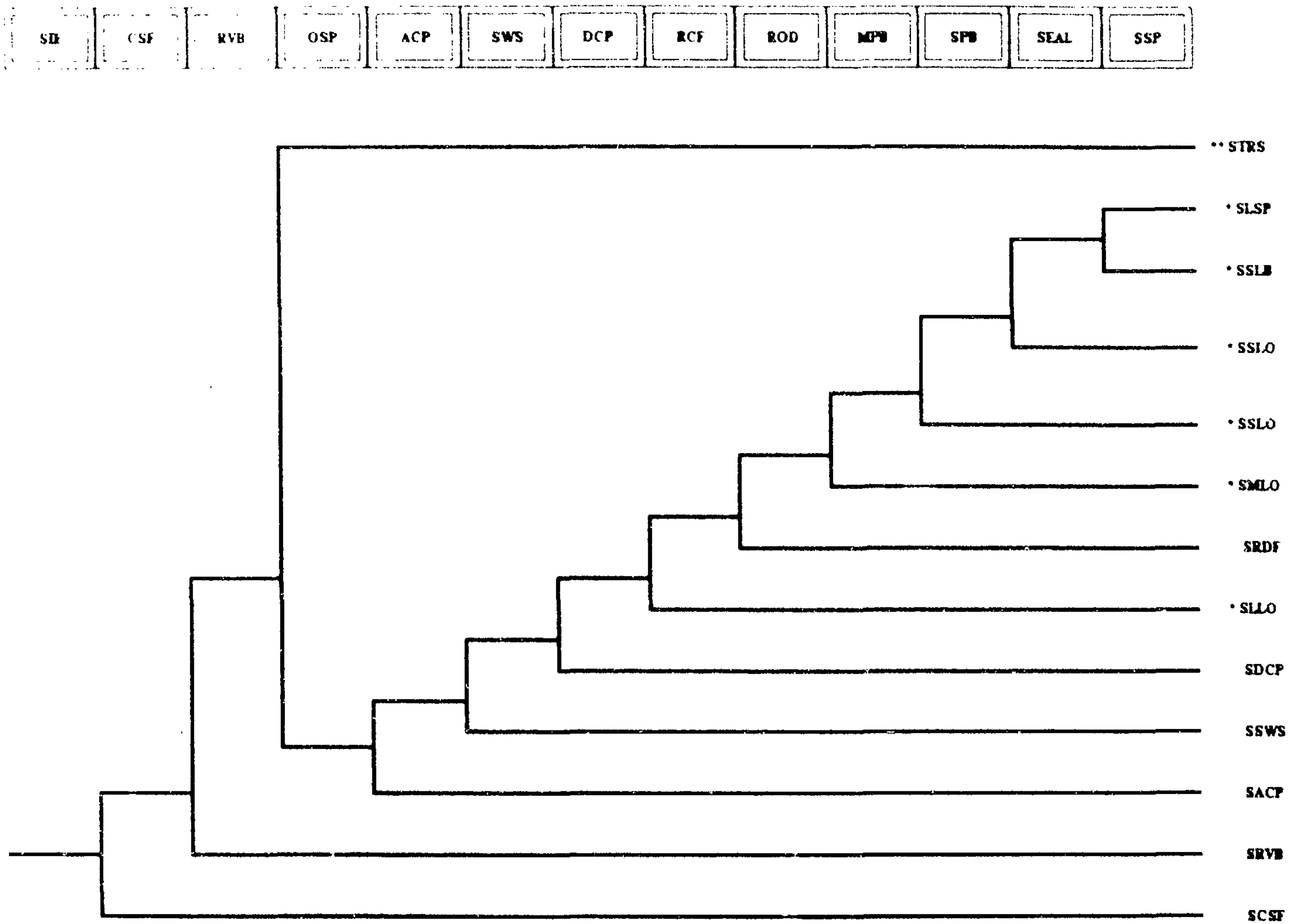


FIGURE 3-4 SEISMIC INITIATING EVENT TREE



** With Off-Site Power Not Explicitly Modeled

* Sequence Continues On Associated Seismic Event Tree

FIGURE 3-5 SEISMIC LARGE LOCA EVENT TREE

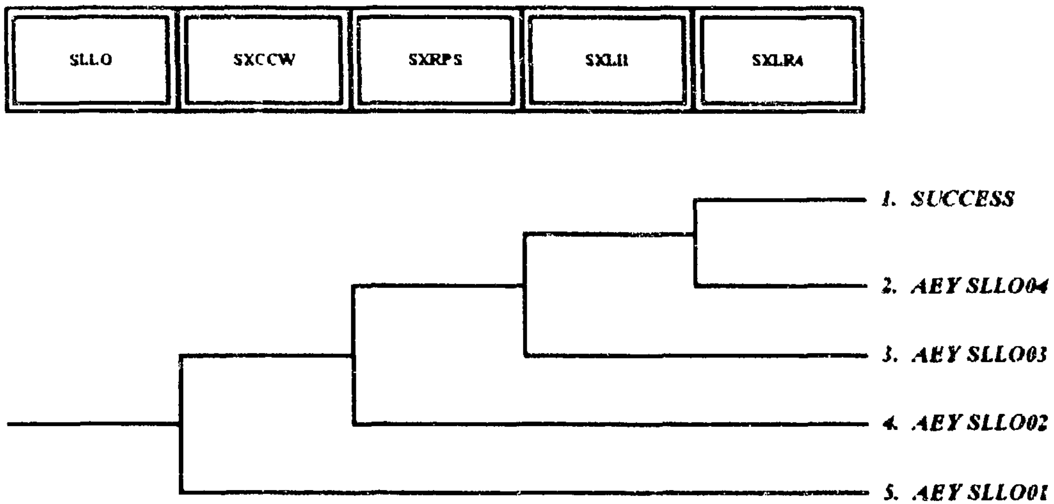


FIGURE 3-4 SEISMIC MEDIUM LOCA EVENT TREE

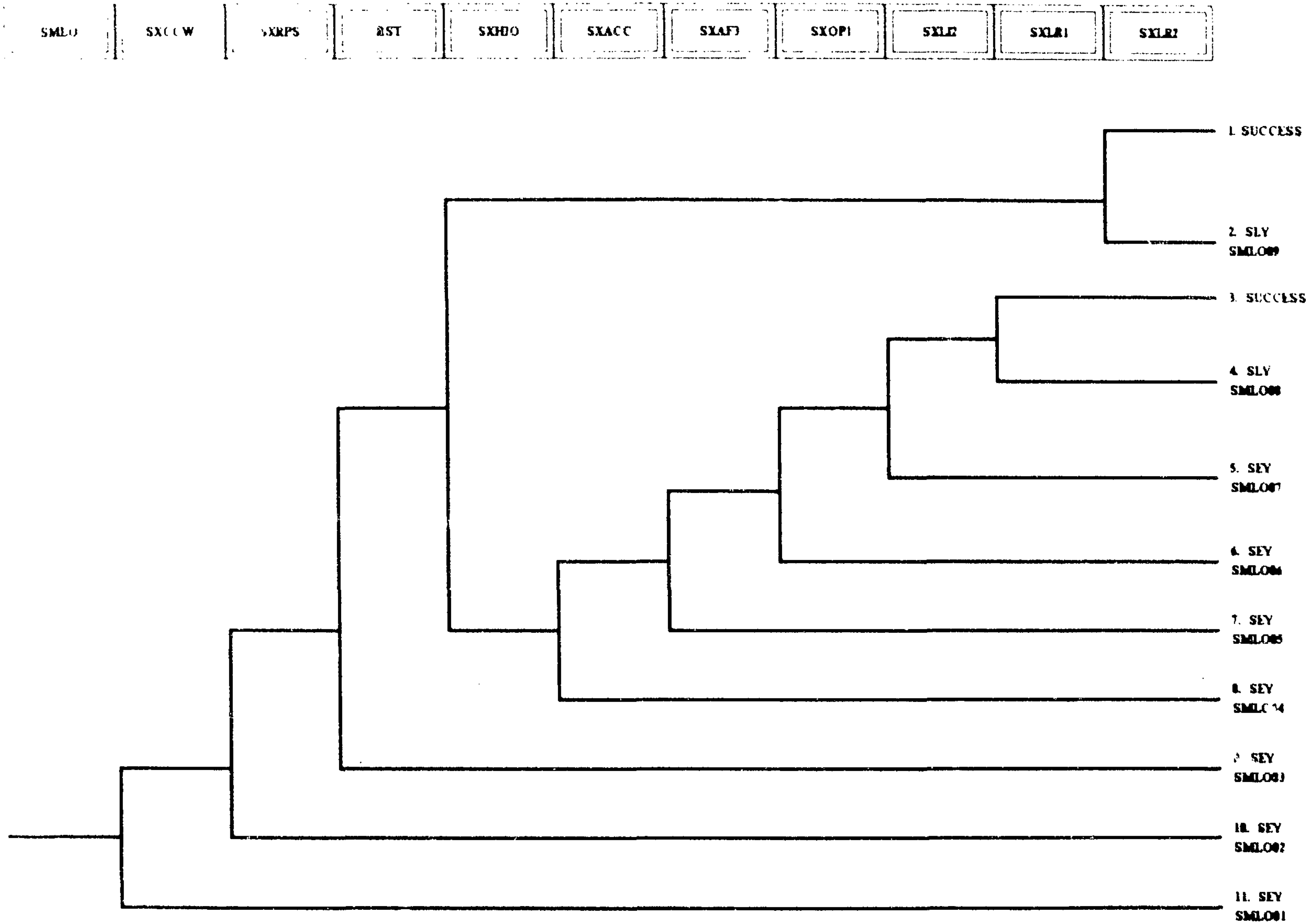


FIGURE 3-7 SEISMIC SMALL LOCA EVENT TREE

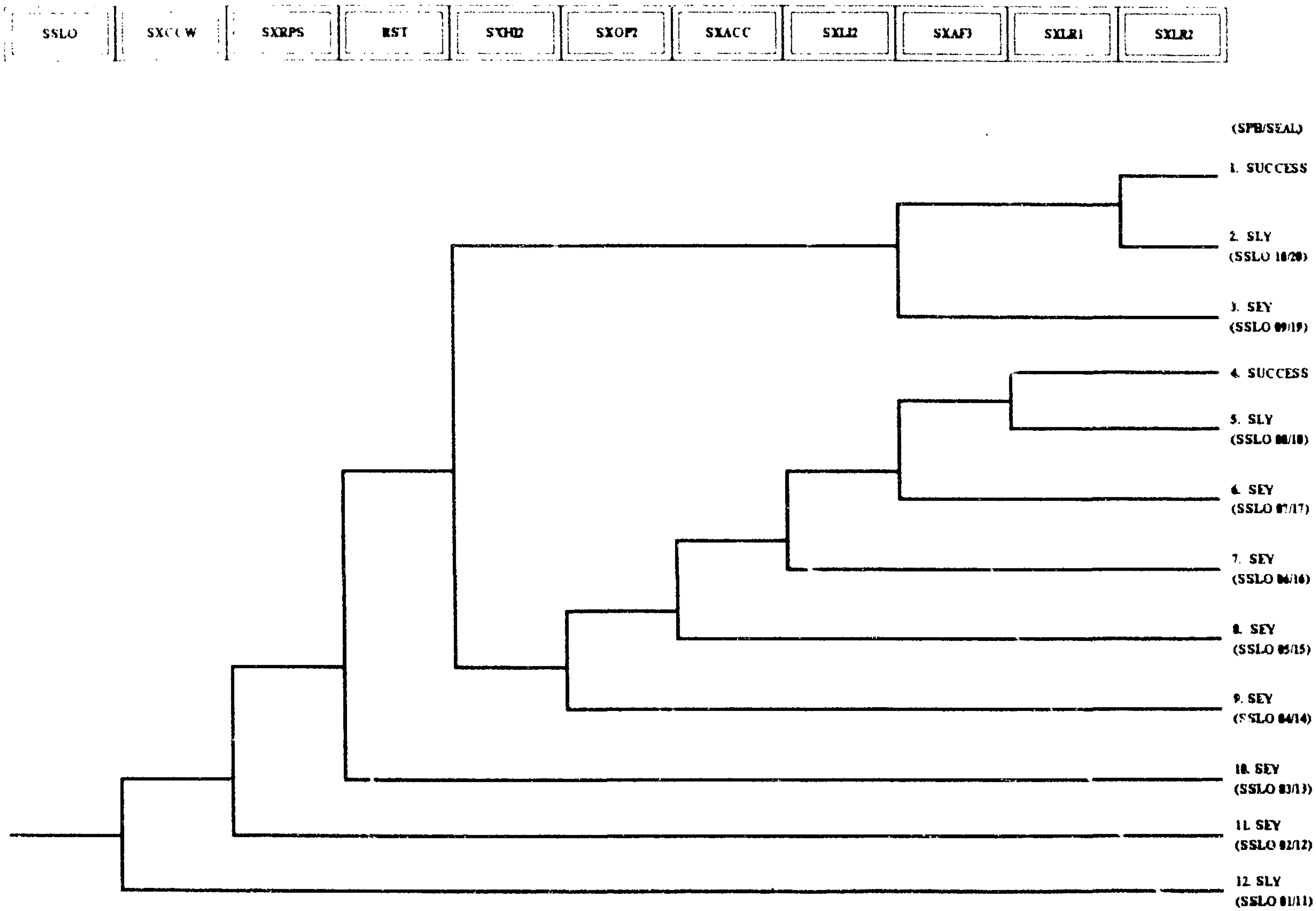


FIGURE 3-8 SEISMIC LOSS OF OFF SITE POWER EVENT TREE

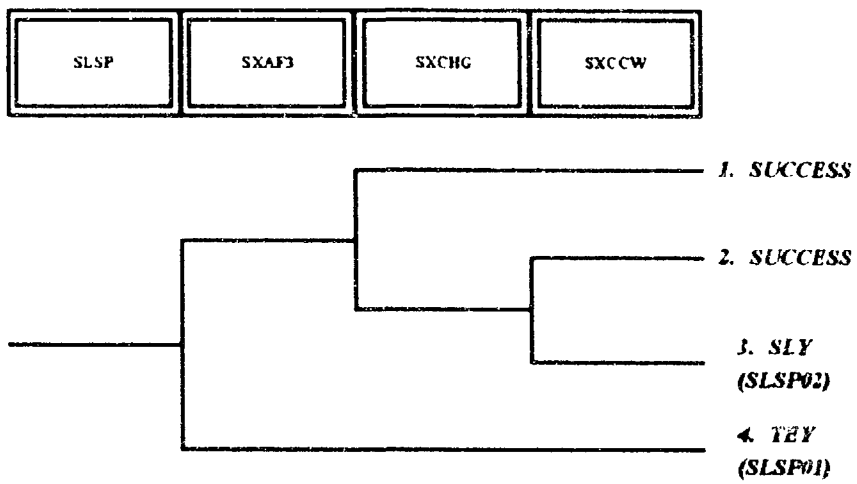


FIGURE 3-9 SEISMIC STEAM LINE BREAK EVENT TREE

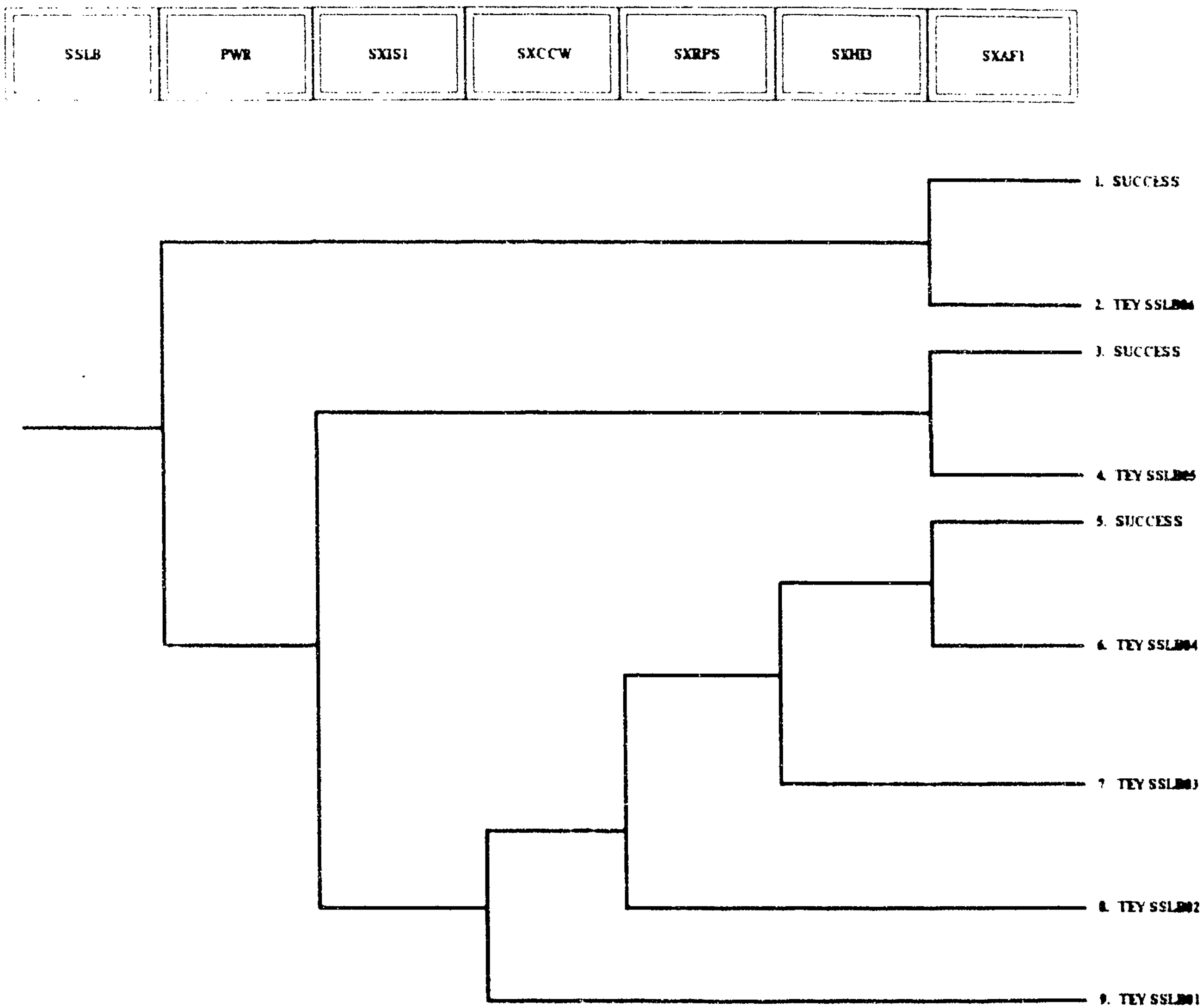
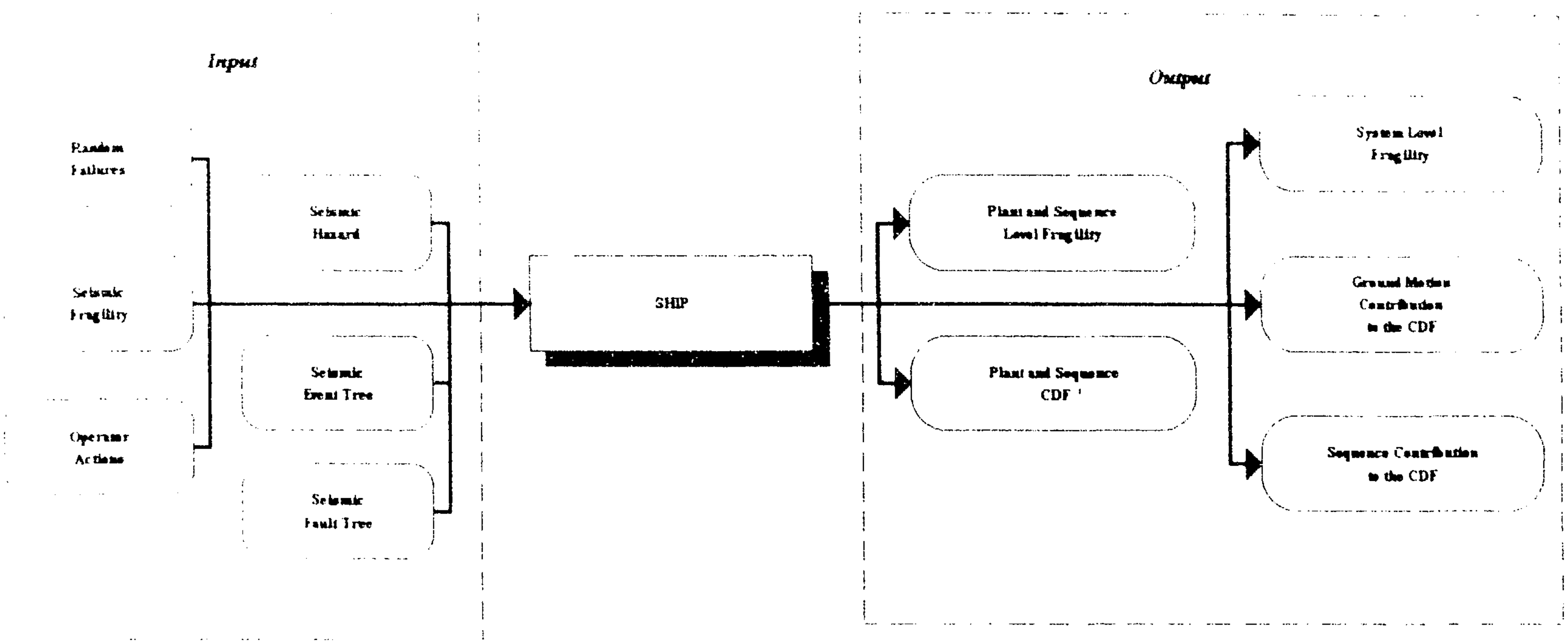


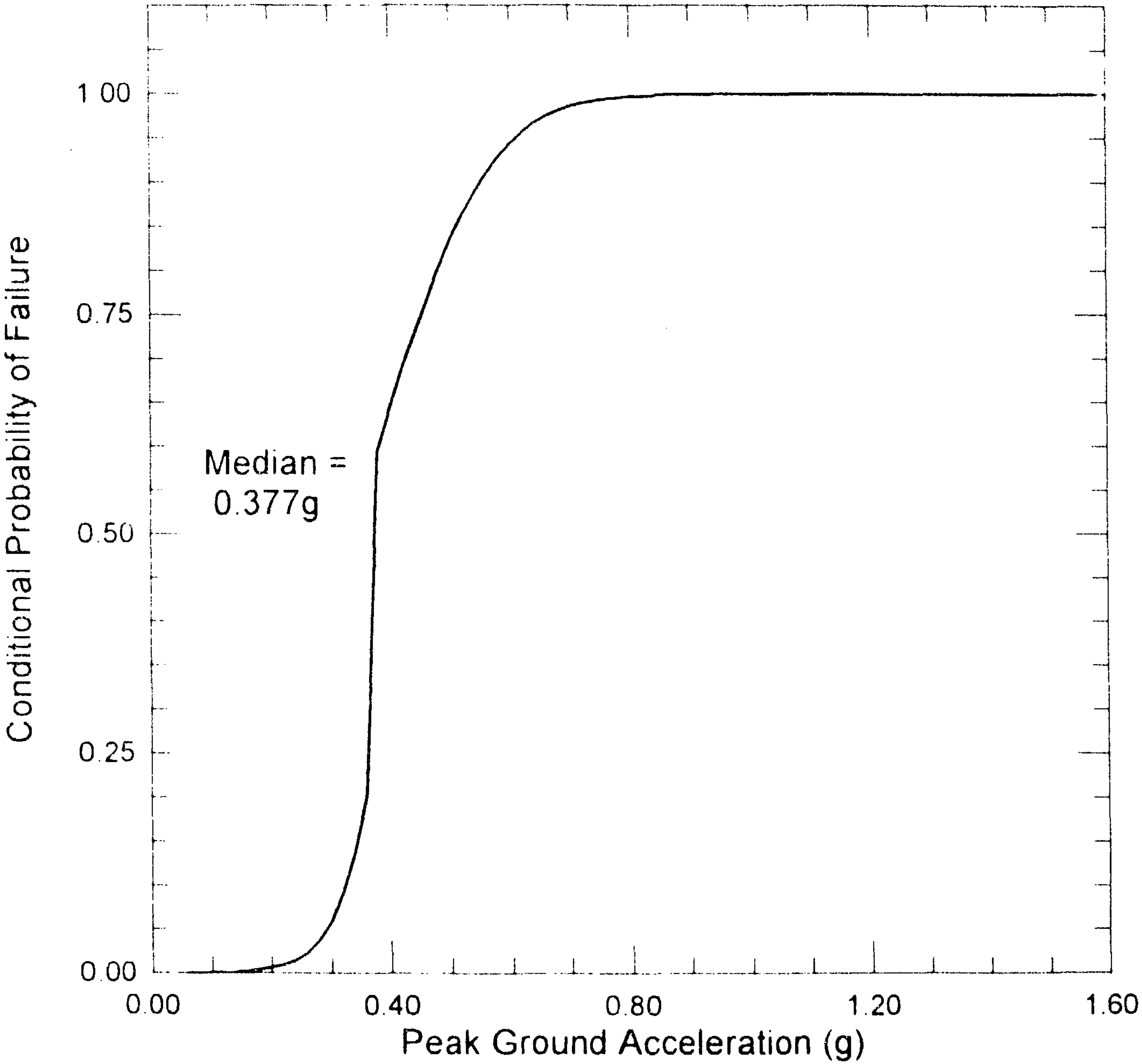
Figure 3-10

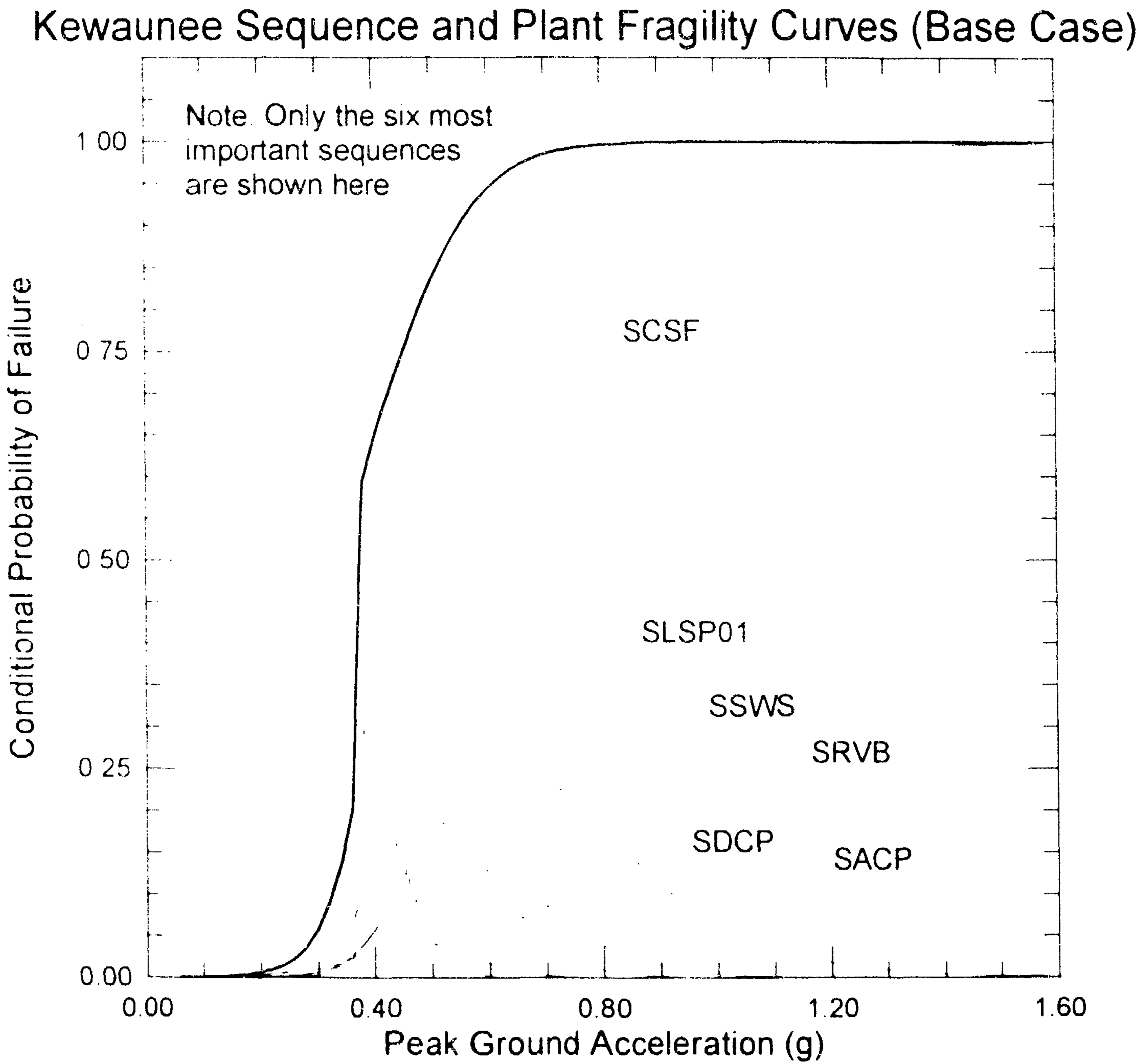
SHIP FLOW CHART

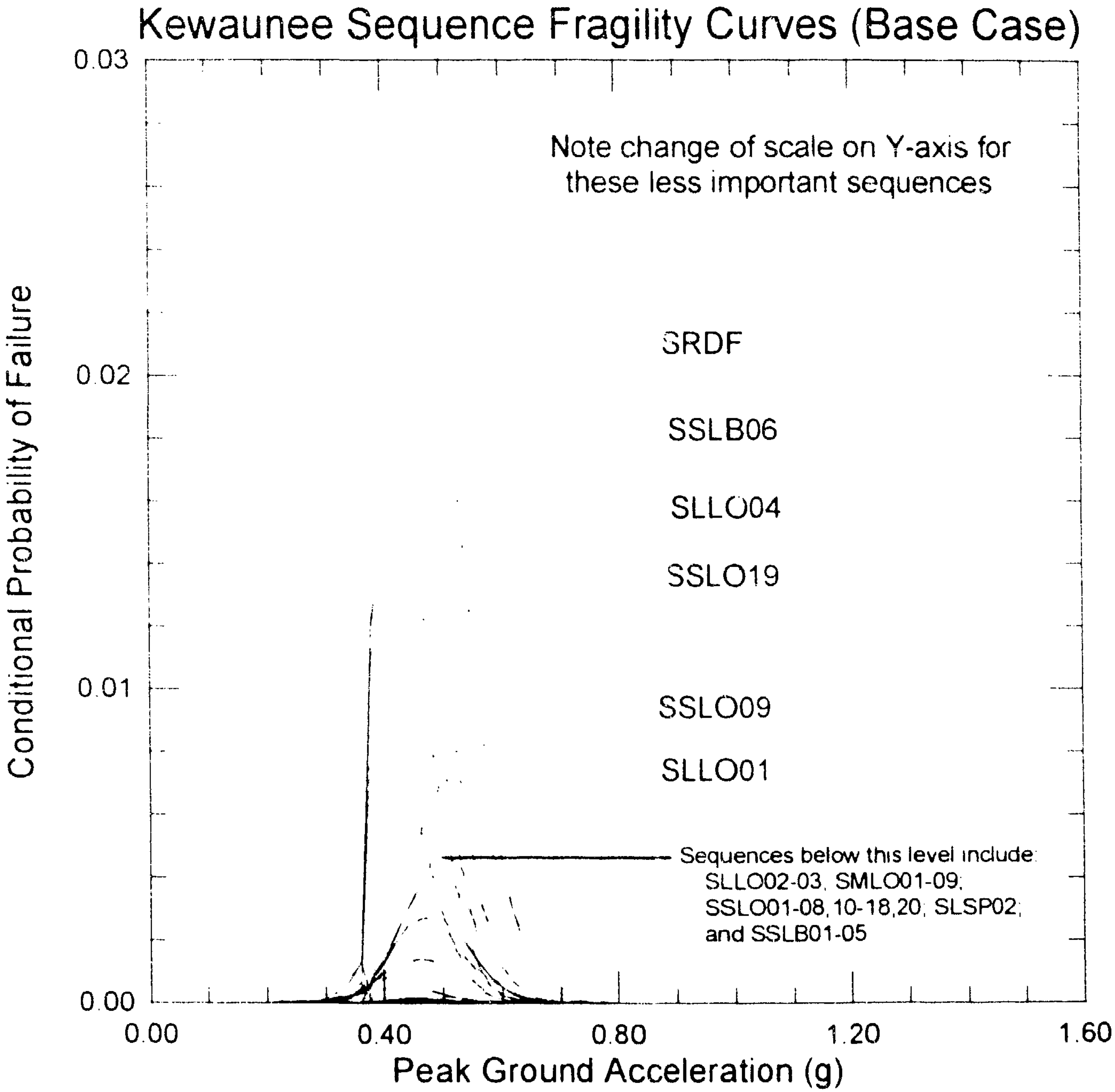


* CDF Core Damage Frequency

Kewaunee Plant Fragility Curve (Base Case)

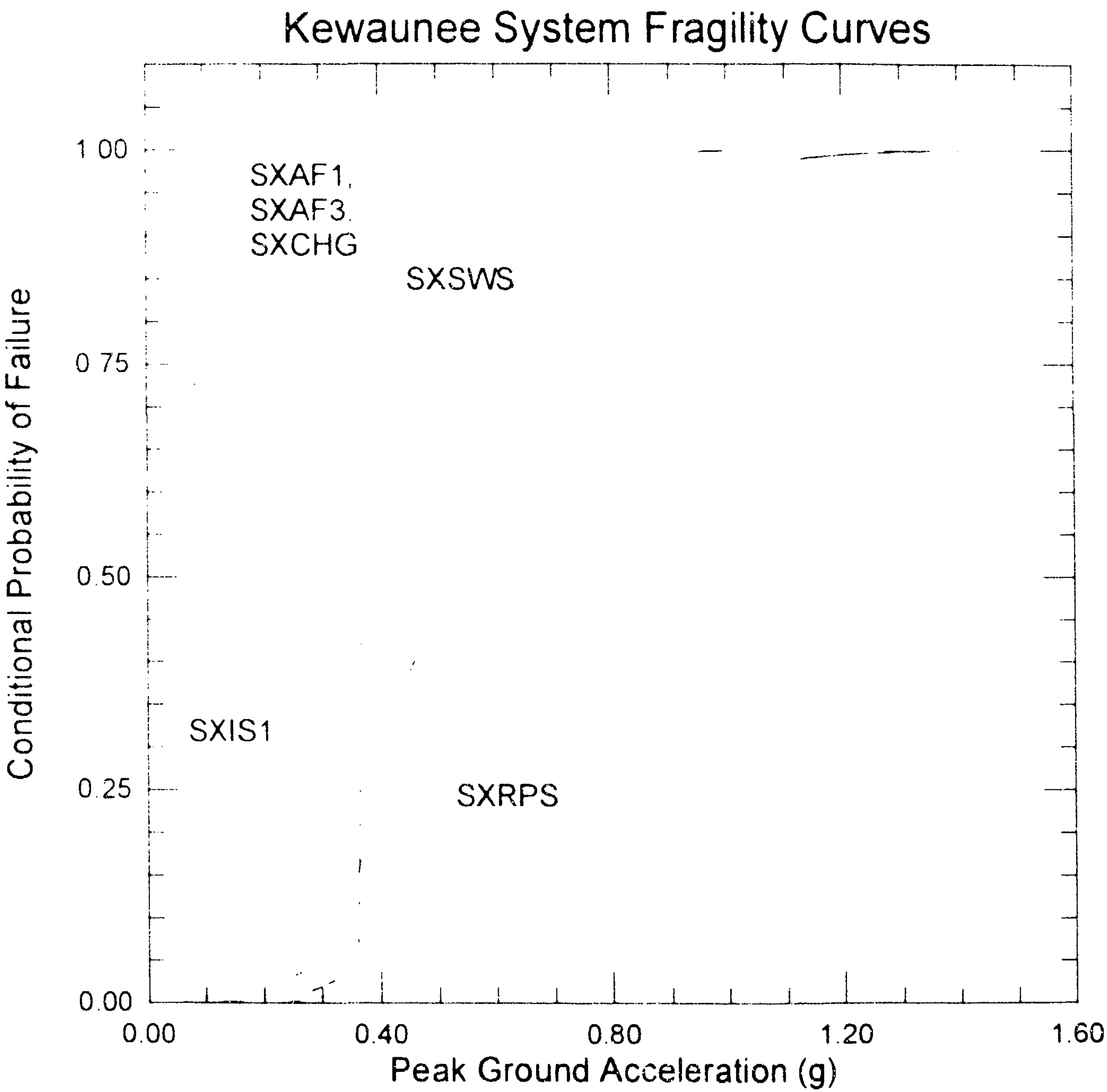



Kewaunee Sequence and Plant Fragility Curves (Base Case)
Note. Only the six most important sequences are shown here
SCSF
SLSP01
SSWS
SRVB
SDCP
SACP
Conditional Probability of Failure
1 00
0 75
0 50
0 25
0.00
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)



Kewaunee Sequence Fragility Curves (Base Case)
0.03
0.02
0.01
0.00
Conditional Probability of Failure
Note change of scale on Y-axis for these less important sequences
SRDF
SSLB06
SLLO04
SSLO19
SSLO09
SLLO01
Sequences below this level include: SLLO02-03, SMLO01-09; SSLO01-08,10-18,20; SLSP02; and SSLB01-05
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)

FIGURE 3-114


Kewaunee System Fragility Curves
1.00
0.75
0.50
0.25
0.00
SXAF1,
SXAF3,
SXCHG
SXSWS
SXIS1
SXRPS
Conditional Probability of Failure
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)

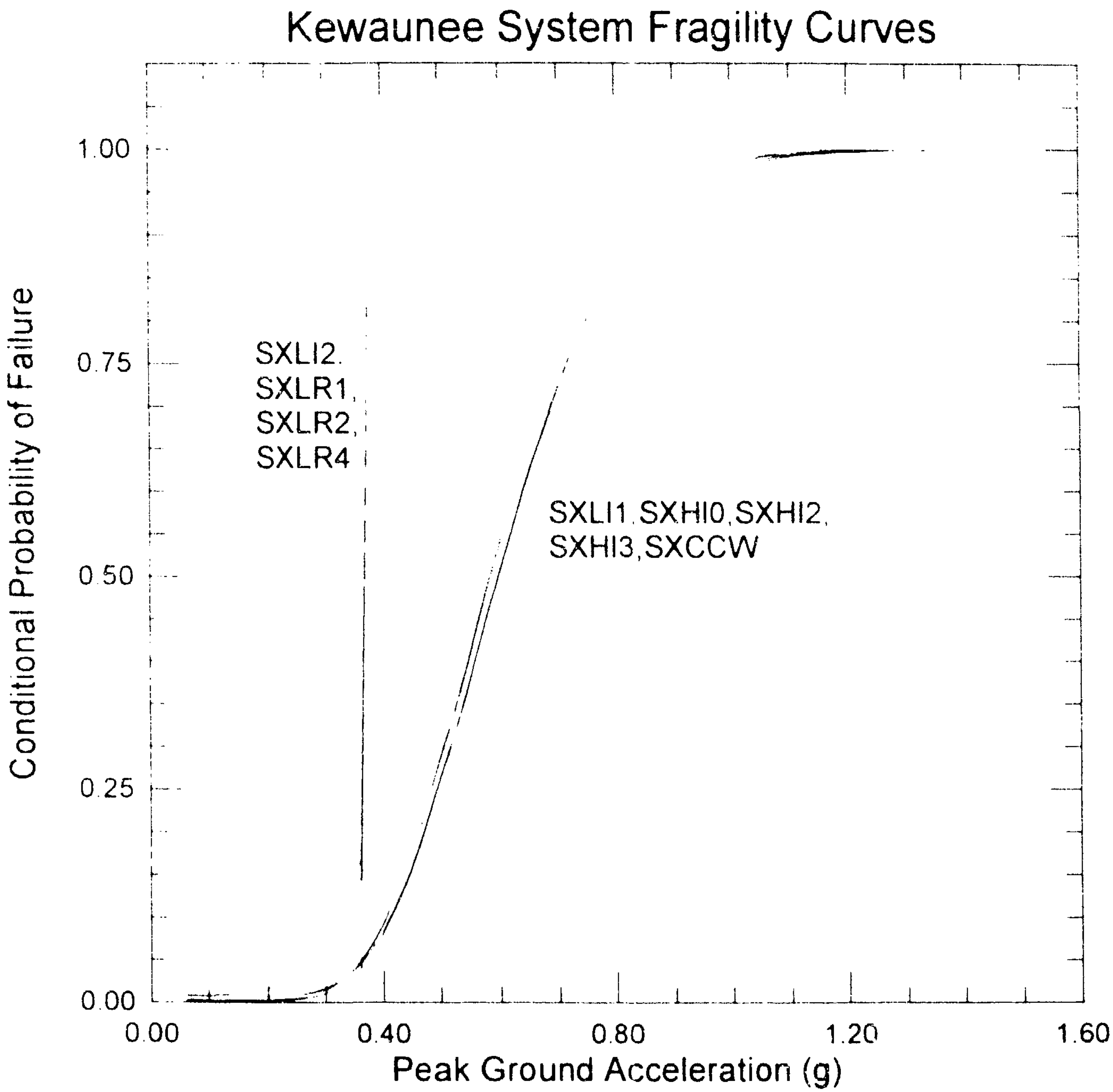
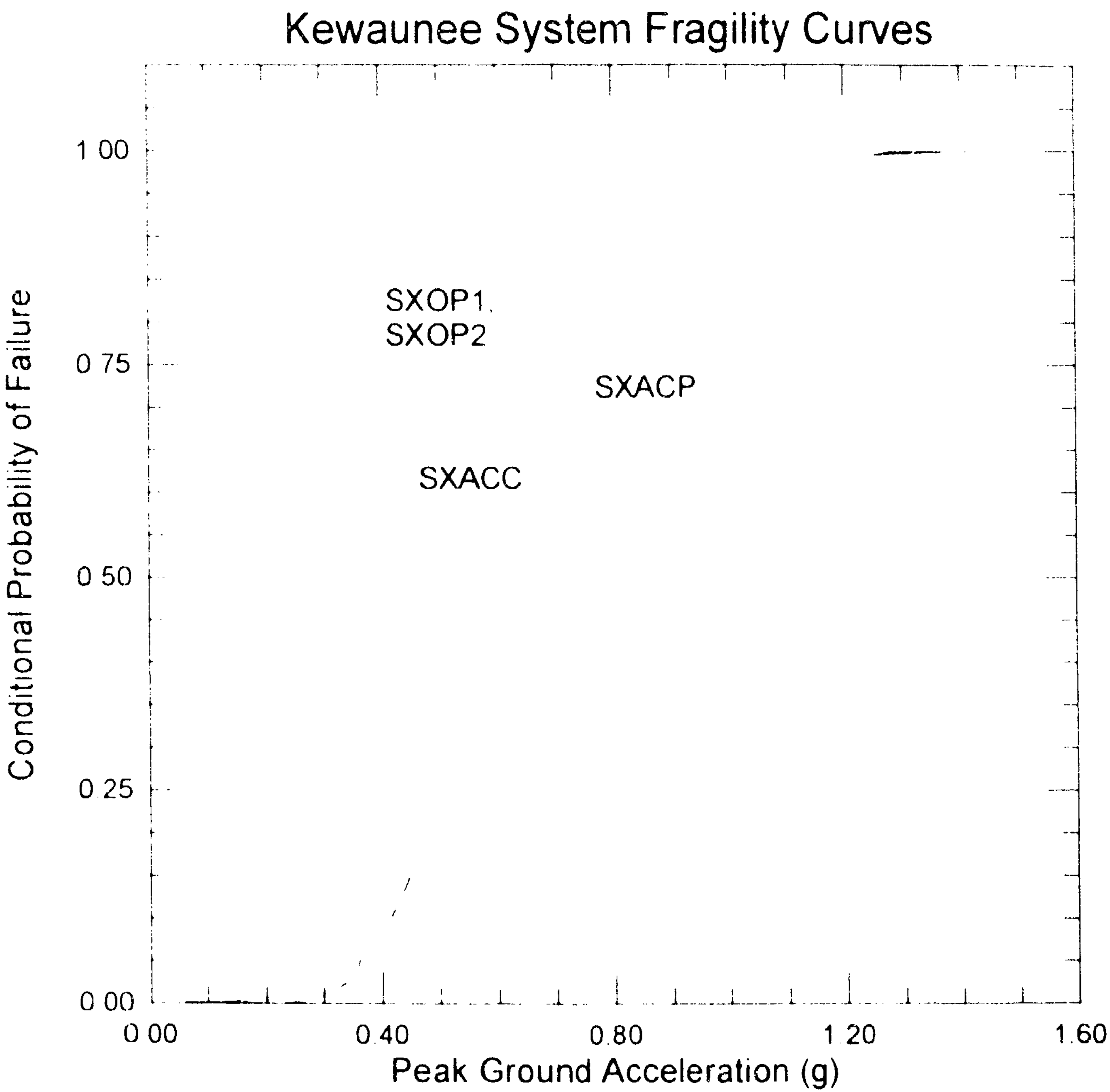
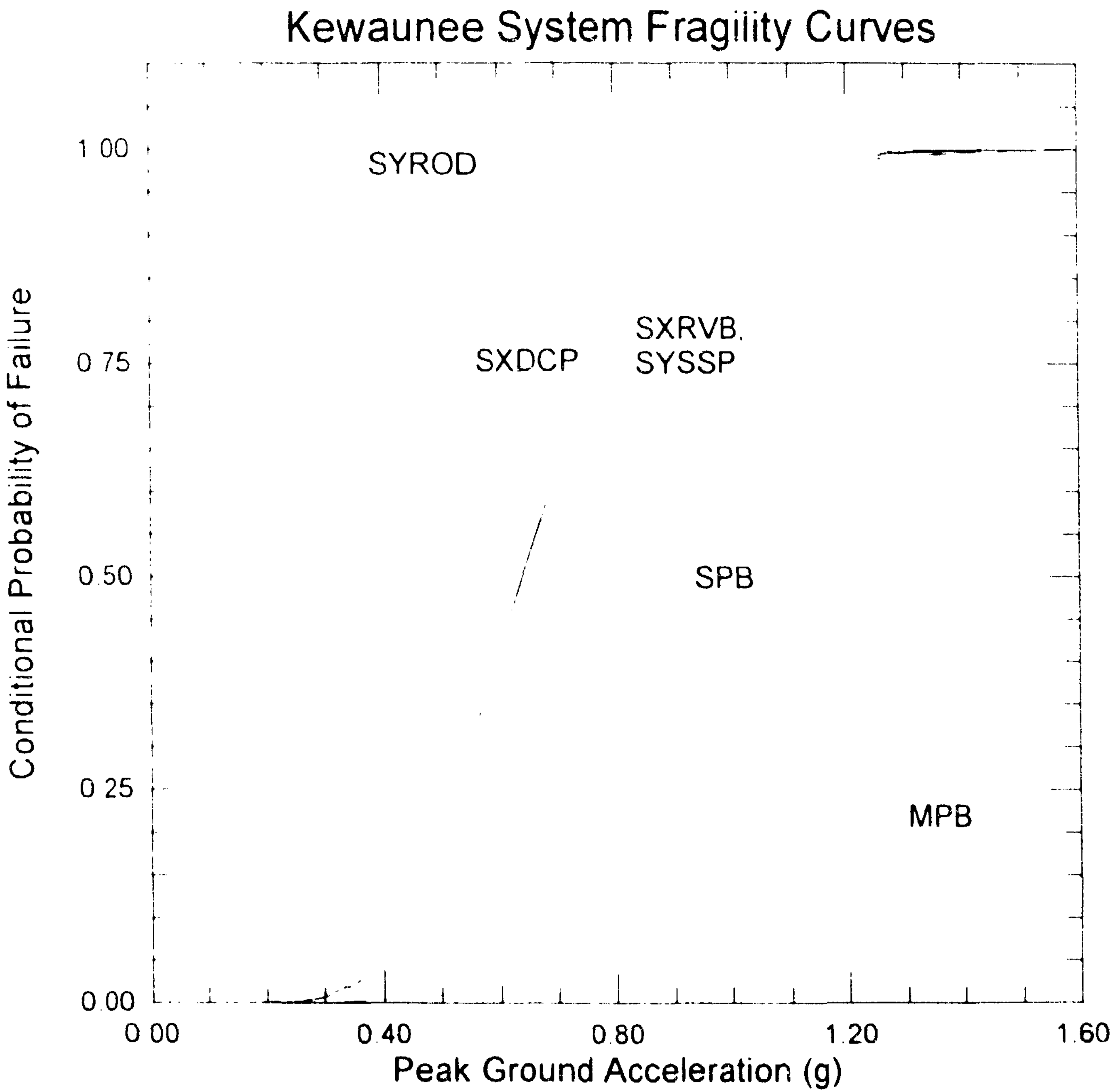
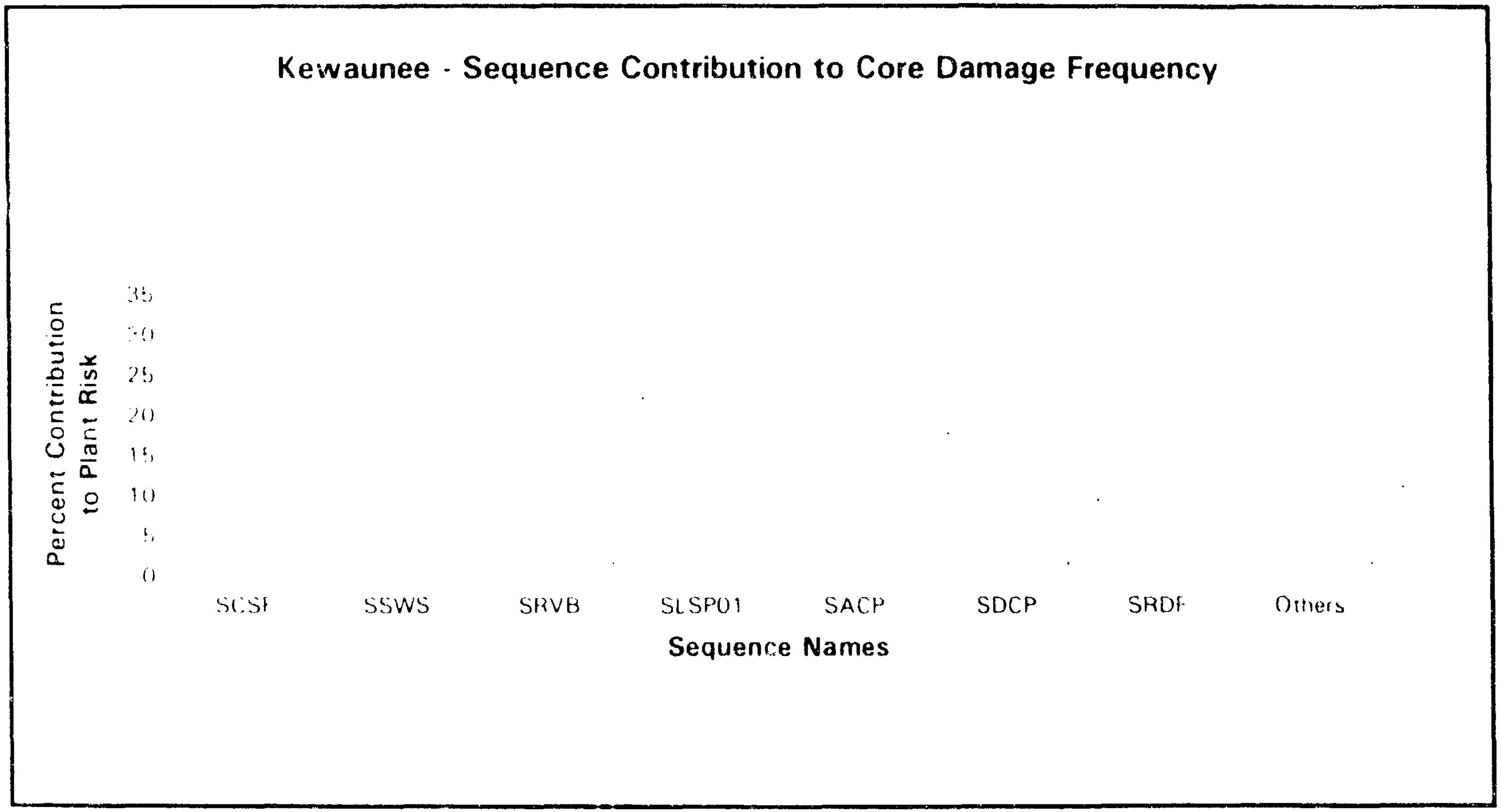

Kewaunee System Fragility Curves
1.00
0.75
0.50
0.25
0.00
0.00
0.40
0.80
1.20
1.60
Conditional Probability of Failure
Peak Ground Acceleration (g)
SXLI2.
SXLR1,
SXLR2,
SXLR4
SXLI1,SXHI0,SXHI2,
SXHI3,SXCCW

FIGURE 3-18


Kewaunee System Fragility Curves
SXOP1
SXOP2
SXACP
SXACC
Conditional Probability of Failure
1.00
0.75
0.50
0.25
0.00
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)


Kewaunee System Fragility Curves
SYROD
SXDCP
SXRVB,
SYSSP
SPB
MPB
1.00
0.75
0.50
0.25
0.00
0.00
0.40
0.80
1.20
1.60
Conditional Probability of Failure
Peak Ground Acceleration (g)


Kewaunee - Sequence Contribution to Core Damage Frequency
Percent Contribution to Plant Risk
35
30
25
20
15
10
5
0
SCSF
SSWS
SRVB
SLSP01
SACP
SDCP
SRDF
Others
Sequence Names

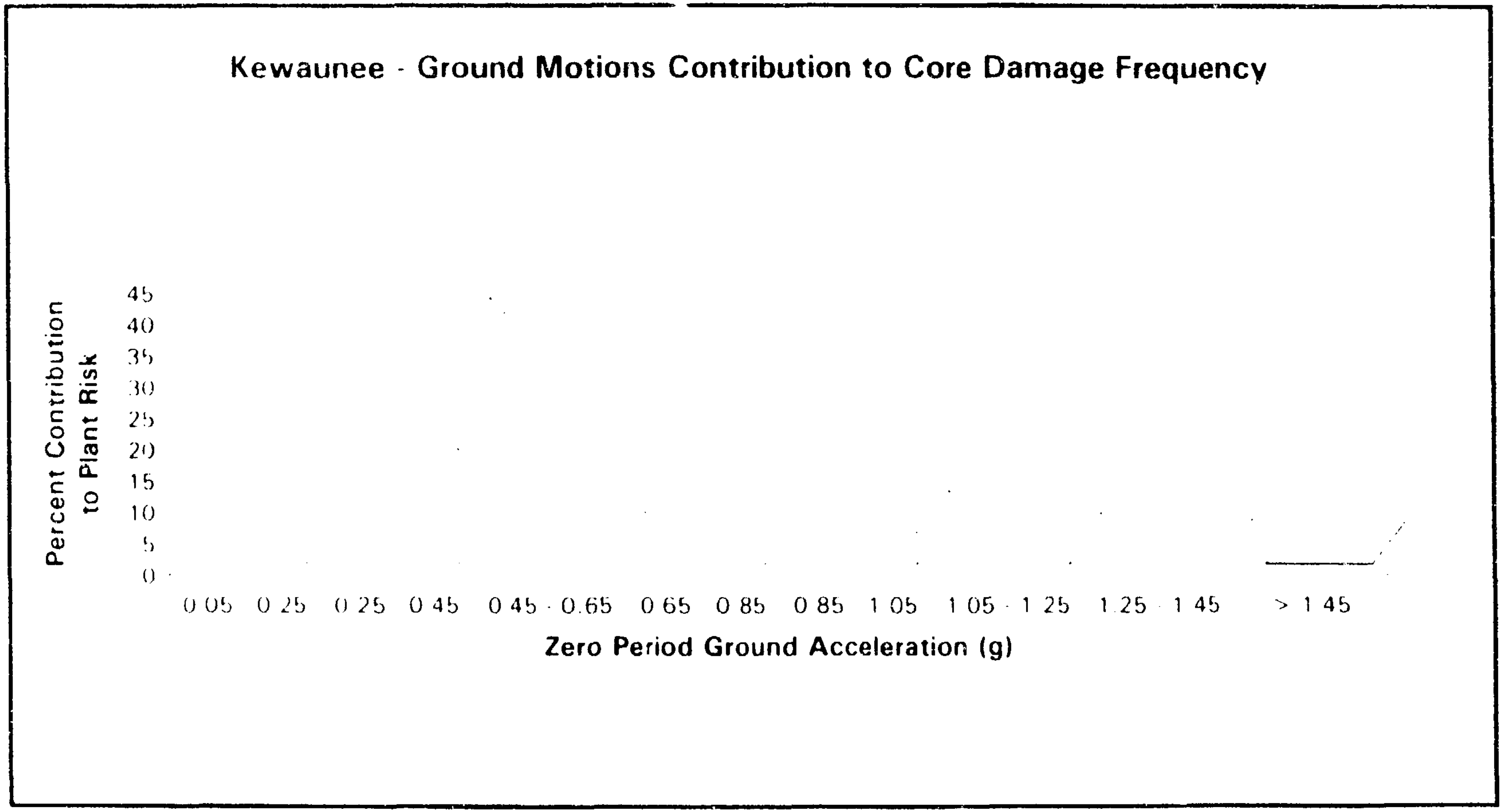

Kewaunee - Ground Motions Contribution to Core Damage Frequency
Percent Contribution to Plant Risk
45
40
35
30
25
20
15
10
5
0
0.05 0.25
0.25 0.45
0.45 - 0.65
0.65 0.85
0.85 1.05
1.05 - 1.25
1.25 - 1.45
> 1.45
Zero Period Ground Acceleration (g)

FIGURE 3-20

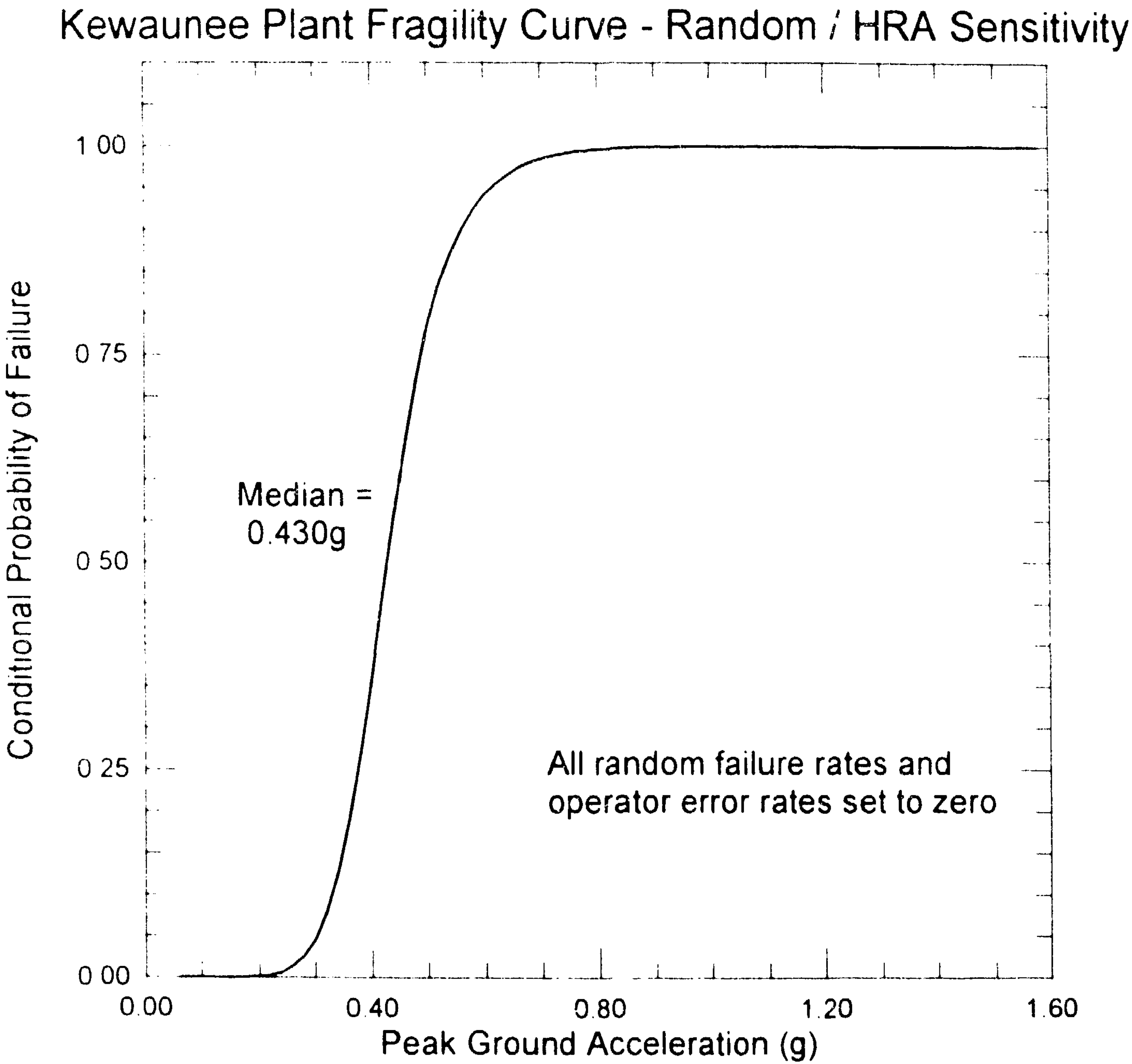
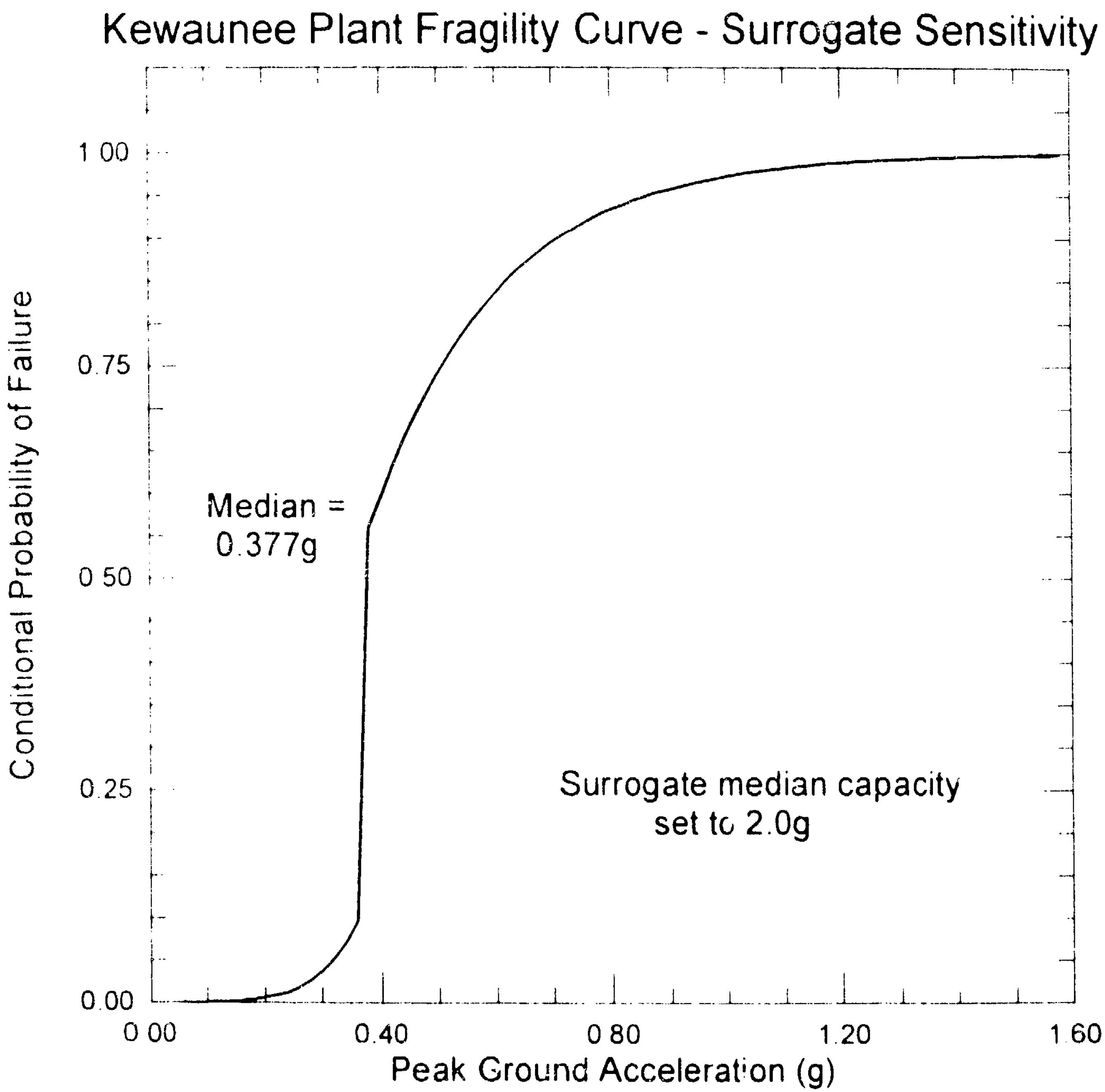
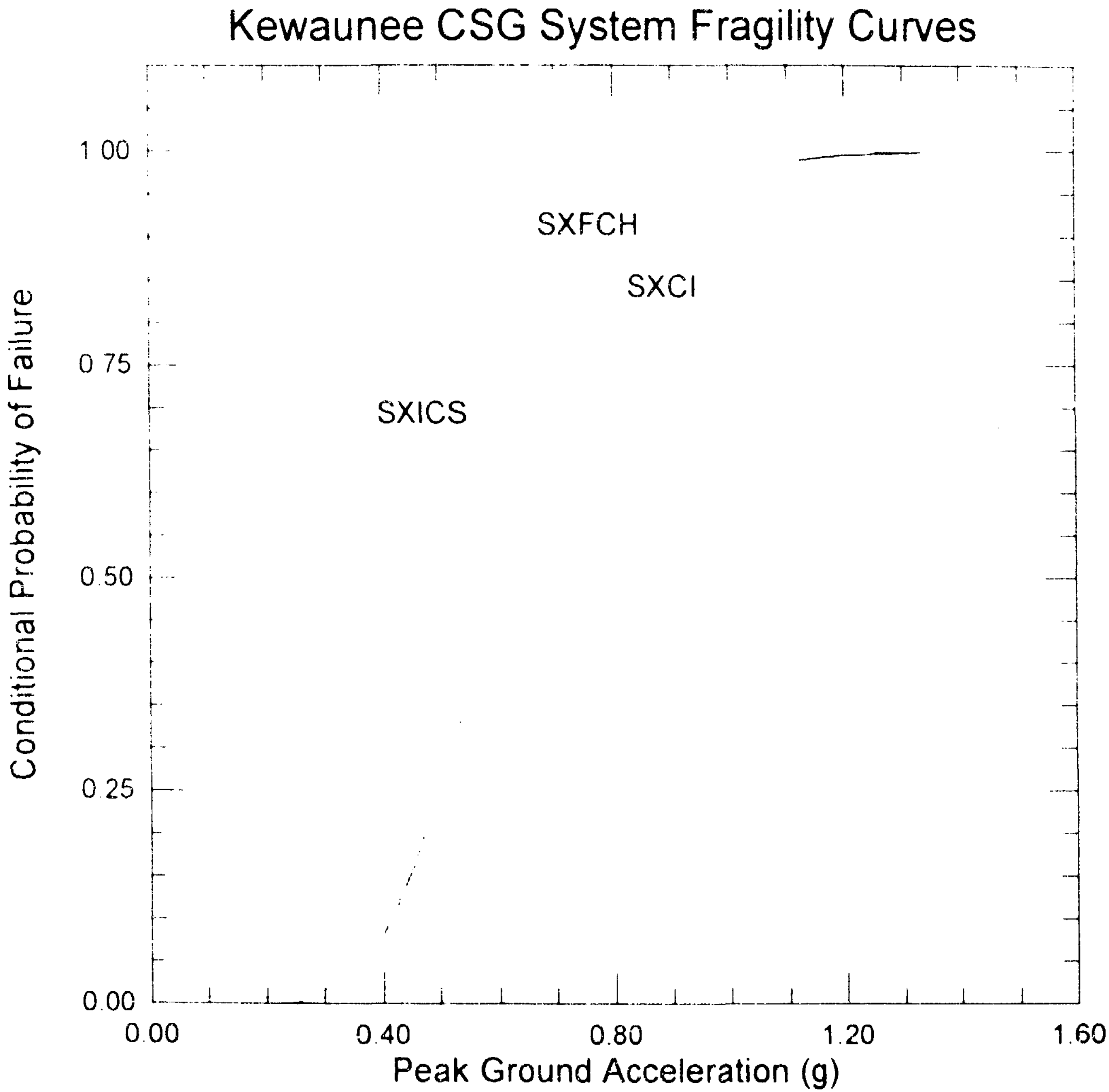

Kewaunee Plant Fragility Curve - Random / HRA Sensitivity
1 00
0 75
0 50
0 25
0 00
Conditional Probability of Failure
Median =
0.430g
All random failure rates and
operator error rates set to zero
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)

FIGURE 3-21


Kewaunee Plant Fragility Curve - Surrogate Sensitivity
Conditional Probability of Failure
1.00
0.75
0.50
0.25
0.00
Median =
0.377g
Surrogate median capacity
set to 2.0g
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)


Kewaunee CSG System Fragility Curves
1.00
0.75
0.50
0.25
0.00
SXFCH
SXCI
SXICS
Conditional Probability of Failure
0.00
0.40
0.80
1.20
1.60
Peak Ground Acceleration (g)

4. Fire Hazards Analysis

SECTION 4

FIRE HAZARDS ANALYSIS

TABLE OF CONTENTS

SECTION **PAGE**

SECTION 4

FIRE HAZARDS ANALYSIS

TABLE OF CONTENTS

SECTION 4

FIRE HAZARDS ANALYSIS

TABLE OF CONTENTS

SECTION **PAGE**

SECTION 4

FIRE HAZARDS ANALYSIS

TABLE OF CONTENTS

SECTION **PAGE**

SECTION 4

FIRE HAZARDS ANALYSIS

LIST OF TABLES

SECTION 4

FIRE HAZARDS ANALYSIS

LIST OF FIGURES

4.0 Methodology Selection

A combination of the Fire Induced Vulnerability Evaluation (FIVE) and a fire probabilistic risk assessment (PRA) was selected as the method to satisfy the NRC request described in GL 88-20, Supplement 4 (Reference 1).

Data and information pertinent to this analysis were collected. This included the Kewaunee Appendix R documentation (References 2 and 3), plant-specific information pertaining to fire events (Reference 4), fire-related procedures (Reference 5), transient combustible control procedures (Reference 6), fire brigade performance (References 7, 8, and 9). Locations of each of the components and their cable routes in the fire zones were determined using general arrangement drawings (Reference 10) and Appendix R information.

Possible fire-induced initiating events were identified in each fire zone based on disabling all components and cabling in the zone.

A screening analysis was then done. A fire destroying everything in each fire zone was postulated. If the fire endangered components modeled in the PRA and required for mitigation of the event initiated by the fire, the zone was retained for further analysis; otherwise it was not considered further in the analysis. A second screening analysis was then done, assuming that the contents of a room or zone would be destroyed, given a fire. Fire ignition frequency was calculated for each of the remaining zones and the contents (components and cables) identified. The ignition frequency was calculated using the method described in the EPRI FIVE methodology (Reference 11). The applicable accident sequences were identified for each zone, based on the supposition that a fire destroyed the contents of the room. The accident sequences were modified using the Westinghouse WALT code (Reference 12), and the core damage frequencies recalculated. The initiating event frequency was changed to the fire ignition frequency, and the failure probabilities of all components or cables located in that room or zone were changed from their random values to 1.0. If the fire-induced core damage frequency did not exceed 1E-06/yr, then that zone was screened from further analysis. 1E-06/yr was selected as the cutoff frequency because this is the value used in FIVE and the EPRI fire PRA methodology for screening purposes. If the core damage frequency exceeded the screening value, then that fire zone was retained for further analysis. The likelihood of fire propagation was also assessed, and affected zones were re-screened.

In conjunction with the frequency of fire initiation, the failure probability of automatic suppression was then factored into development of the initiating event frequency. Remaining zones were then modeled with the COMPBRN IIIe code (Reference 13) to determine heat transfer characteristics and the likelihood of certain components or cables in the zone being damaged, given a fire of a certain size and in a certain location. In order to gather the necessary data for this modeling, the plant was walked down in October of 1992. The following tasks were accomplished:

- Information used in this analysis was verified, and assumptions used in the analysis were confirmed.
- Fire propagation potential was determined and barrier integrity was verified.
- Layout of components and cable trays was verified.

- Three dimensional location information for components and cable trays in each compartment retained for further analysis was obtained. This was used to model fire damage to components and cables using COMPBRN.
- Interviews with plant personnel took place, and the information obtained included suppression system capabilities, likely transient combustibles in zones, and fire brigade training.
- Possible suppression agent induced damage to equipment was identified.
- The potential for interactions between control room and remote shutdown capabilities was examined.
- Possible seismic-fire interactions were identified.

COMPBRN was then run for those zones that remained after the screening analysis to determine if a certain component would be damaged, given a fire of a certain magnitude and in a certain location. Each of the zones was partitioned further into compartments, dependent on spatial separation or heat transfer barriers, to facilitate modeling. The three dimensional measurements of components and cables in the compartment were used to model heat transfer, given a fire, and determine if a component or cable tray would be damaged, given a fire in a certain location. Use of COMPBRN allowed determination of the extent and timing of damage. This made possible reductions in the initiating event frequency, by partitioning the "area of influence" over the compartment or zone area, and allowed credit of manual suppression efforts, if they could occur prior to damage.

COMPBRN was also needed to identify components and cables that would be affected by a postulated fire.

The internal event models were then modified to reflect the fire-induced initiating event frequency and the fire-damaged components. These changes allowed quantification of fire-induced damage in conjunction with random failures, resulting in fire-induced core damage frequency. Each fire scenario was quantified using the WLINK fault tree linking code (Reference 12) to get its own fire-induced core damage frequency. These frequencies were then summed to obtain total fire-induced core damage frequency.

4.1 Fire Hazards Analysis

4.1.1 Definitions and Assumptions

Following are the definitions of some terms used in the Kewaunee Appendix R effort.

Fire Area.	An area segregated from neighboring areas by three hour fire barriers. A fire area delineates the plant locations using the alternate and dedicated methods of safe shutdown available at the Kewaunee Nuclear Power Plant.
Fire Zone:	Fire zones are combined into fire areas. The fire zones at Kewaunee separate safety-related equipment from combustibles, and identify areas having similar fire protection characteristics.

Fire Compartment: Fire compartments are contained within fire zones, and are delineated by physical characteristics such as spatial separation or heat transfer barriers such as components. Fire compartments are defined for ease of modeling.

Area of Influence: Area over which a fire can damage certain components or cables, given a fire.

The following assumptions are used in this analysis:

1. It is assumed that a reactor trip would be generated (either automatically or manually) upon significant fire initiation in zones housing components and cabling required for safe shutdown.

2. A 24-hour period is assumed as the base mission time for this analysis. This time is consistent with the internal events analysis and is based on NUREG-1335 (Reference 13).

3. Fire-induced disabling of the control room heating ventilation and cooling (HVAC) is not assumed to result in control room uninhabitability. The control room is constantly manned, and a heating or cooling failure would be noticed, and corrective action taken, according to the applicable procedure, in a timely manner.

4. It is assumed that fire barriers remain intact for fires of less than rated duration. For instance, it is assumed that a 3-hour fire barrier could withstand a 2-hour fire.

5. Motor control centers (MCCs) and other metal-enclosed components are not considered to be vulnerable to a low-intensity external exposure fire. However, unprotected cables entering and exiting the metal-enclosed component are considered to be vulnerable. Low intensity fires internal to the cabinet are assumed to be confined to that cabinet.

6. The delineations and boundaries employed in the Appendix R analyses are used in this analysis. They are, however, examined to ensure applicability to this analysis.

7. It is assumed that all automatic fire suppression systems are sized to effectively mitigate a maximum sized fire.

8. It is assumed that the fire brigade failure probabilities account for any fire-induced access difficulties. Fire brigade response times are compared with damage times prior to assigning credit.

9. It is assumed that successful automatic suppression is initiated instantaneously upon successful fire detection.

10. Cables in conduit are assumed to be protected from thermal damage for a period of one hour. Conduit dissipates radiative heat. If a fire is of greater than 60 minutes in duration, cables in conduit are assumed to be vulnerable to damage.

11. Instrumentation and control cables at Kewaunee are qualified to IEEE-383 standards. For this analysis, it is assumed that all cables of interest have polyethylene (PE) insulation.

12. Information obtained from Appendix R documentation is assumed to be current and valid. A Wisconsin Public Service Corporation internal audit of all this documentation was completed in 1991.

13. Appendix R fire barriers are assumed to remain intact under rated fire conditions. Fire loading for all zones was reviewed, and most zones do not contain sufficient combustibles to sustain a fire of duration long enough to challenge the barriers. For those zones with high combustible loadings (e.g., diesel fuel oil day tank rooms, turbine lube oil room), barrier failure and potential propagation is considered.

    Where a 20 ft. separation is used to separate zones within the same fire area, it is assumed that this is adequate spatial separation to prevent inter-zonal propagation, if intervening combustibles are not present. The presence, or lack of, intervening combustibles which could allow fire propagation, was determined during the walkdown. It was found that transient combustibles are limited, reducing the chance of inter-zonal propagation.

14. Power cabling throughout Kewaunee is armored, and in trays that are separate from instrumentation and control cabling. This was verified during the walkdown.

4.1.2 Containment Fires

A fire in containment could have severe consequences. However, a damaging hot gas layer resulting from the fire is unlikely to form in most areas of containment due to the large volume inside containment. Vulnerable equipment and cables are located on the lower elevations, where a hot gas layer which could damage cables is not likely to form. Redundant trains of equipment were also found to be segregated such that a single fire plume or hot gas layer would not threaten both trains of equipment.

A large percentage of past fires in nuclear reactor containments have been reactor coolant pump (RXCP) oil fires. These are not likely at Kewaunee due to the installation of the RXCP oil collection system. As described in the Appendix R Design Description, the RXCP lube oil collection system poses no threat to components or cables within containment.

In addition, the combustible loading for containment is low. During maintenance activities, the combustible loading may increase, but Kewaunee has administrative controls to restrict the amount of transient combustibles.

For these reasons, containment fires are concluded to pose no significant threat at Kewaunee.

4.1.3 Evaluation of Fire-Induced Initiating Events

Initiating events that may theoretically be induced by a fire include the following:

- small break loss of coolant accident (LOCA)
- inadvertent opening of pressurizer power operated relief valves (PORVs) leading to small LOCA
- loss of charging and component cooling water leading to RXCP seal LOCA
- interfacing systems LOCA
- transients with main feedwater available
- transients without main feedwater available
- loss of offsite power
- anticipated transient without SCRAM
- loss of station and instrument air system
- loss of a 125 V DC bus
- loss of service water
- loss of component cooling water

Each is examined and described below:

- Small Break LOCA - inadvertent opening of pressurizer PORVs leading to small LOCA

  Prolonged opening of pressurizer PORVs could lead to a small LOCA. Fire-induced LOCAs are described in the Appendix R Design Description, Sections 2.1.1. and 2.1.2. The PORVs are air-operated, and controlled by solenoid valves. The solenoid valves have their fuses removed in procedure E-0-06 and E-0-07. This ensures closure of the PORVs, preventing a LOCA.

  Pressurizer PORVs are located in containment. As described in Section 4.2, containment fires are not considered to present hazards to components or cabling. Pressurizer PORV cabling is routed outside of containment, on its way from containment to the control room. Fire effects in the zones outside of containment through which pressurizer PORV cables pass are assessed in this report.

- Small Break LOCA - loss of charging and component cooling water leading to a RXCP seal LOCA

  RXCP seals at Kewaunee are maintained by charging and by component cooling water (CCW). The primary method of maintaining the integrity of the RXCP seals is with seal injection from the positive displacement charging pumps. If seal injection is lost, the seals are cooled with reactor coolant, which passes through the RXCP thermal barriers. These thermal barriers are cooled by CCW. A RXCP seal LOCA is highly unlikely unless both charging and CCW capabilities are lost. CCW pumps are segregated, precluding fire-induced loss of both CCW pumps due to the same fire. Cables are separated, precluding fire-induced loss of cabling for both trains. Sections 2.1.1 and 2.1.2 of the Appendix R Design Description, describe Kewaunee's defense-in-depth strategy to prevent fire-induced LOCAs.

- Interfacing systems LOCA

  A fire-induced interfacing systems LOCA is not a credible event at Kewaunee. A fire could not initiate this event. Fire-induced interfacing systems LOCAs are not considered further in this analysis.

- Transients with main feedwater available

  A fire could induce a transient with main feedwater available. This is the response that is most likely for most fires.

- Transients without main feedwater available

  A fire could disable main feedwater functions, resulting in a transient without main feedwater available.

- Loss of offsite power

  A fire-induced loss of offsite power would occur if a fire were to disable the main transformer, the main auxiliary transformer, the tertiary auxiliary transformer, and the reserve auxiliary transformer. These are separated from each other by three hour barriers, and are located outside the plant. The control cables for these transformers, however, are located in the plant in non-safeguards cable trays. A loss of offsite power could therefore occur as a result of fire damage to these trays. Even if the trays are not damaged, procedures E-0-06, Fire in an Alternate Zone, and E-0-07, Fire in a Dedicated Zone, require manual isolation of offsite power, thus creating a loss of offsite power.

- Anticipated transient without SCRAM (ATWS)

  ATWS could occur if the reactor trip function were disabled. An ATWS due to a fire is not credible at Kewaunee, due to the "fail-safe" design of the Reactor Protection System (RPS), and adequate separation. Possible failure modes, such as stuck rods, are not fire-induced. The control rods drop unless energized. A fire, therefore, would not disable the reactor trip function. It is far more likely to initiate a trip. This event is not considered further in this analysis.

- Loss of station and instrument air

  While safeguards station and instrument air compressors were located and their cables traced in compliance with Appendix R requirements, non-safeguards air compressors and cables were not. A loss of air only occurs when both safeguards and non-safeguards air compressors are disabled. The three non-safeguards air compressors are located in the turbine building basement. The safeguards air compressors and their cabling are located in separate fire areas, such that a single fire cannot disable all air compressors. Calculation performed for the Appendix R effort also showed that a fire cannot damage the station and instrument air piping. A fire-induced loss of air is not a credible event at Kewaunee.

- Loss of a 125 V DC bus

  125 V DC components and cables were located for the Appendix R effort. The two trains of 125 V DC components and cables are separated by three hour fire barriers, preventing a loss of 125 V DC capability, since the DC buses can be supplied by MCCs 52C, 62C, 52E, 62E, or batteries A or B. Due to this diversity and separation that would prevent a single fire from destroying 125 V DC capability, a fire-induced loss of 125 V DC capability is not considered further in this analysis. However, loss of a single 125 V DC bus is considered in this analysis.

- Loss of service water

  A loss of service water could result if a fire were to disable cabling for service water components. Redundant service water components are located in two separate fire areas. Service water cabling was traced as part of the Appendix R effort. It is separated by train such that a single fire could not disable service water functionality. This event is concluded to be incredible at Kewaunee. This event is not considered further in this analysis.

- Loss of component cooling water

  A loss of CCW could result if a fire were to disable cabling for CCW components. Redundant CCW components that are susceptible to fire damage are located in two separate fire areas. CCW cabling was traced as part of the Appendix R effort. It is separated by train such that a single fire could not disable component cooling water functionality. This event is concluded to be incredible at Kewaunee. This event is not considered further in this analysis.

As part of discussions for fire zones, discussion of applicable initiating events is included.

4.1.4 Screening Analysis

4.1.4.1 Zones containing components required for mitigation of fire-induced events

All zones designated for the Appendix R analysis containing components required for mitigation of fire-induced events are defined as follows:

Dedicated Fire Zone:	Those zones containing dedicated shutdown components. Should a fire in a dedicated zone occur, safe shutdown is accomplished with alternate components.
Alternate Fire Zone:	Those zones containing alternate shutdown components. Should a fire in an alternate zone occur, safe shutdown is accomplished with dedicated components.
Designated Fire Zone:	Those zones containing both dedicated and alternate system components that are not easily segregated.
Nondesignated Fire Zones:	Those zones containing equipment not specifically related to achieving safe shutdown.

Designated fire zones include SB-65 (shield building), RC-60 (containment) and TU-99 (diesel generator fuel tanks). Nondesignated fire zones are not listed. A fire in a nondesignated fire zone is not analyzed further, unless it contains important components modeled in the Individual Plant Examination (IPE) (Reference 15). Following is a list of each zone by category.

DEDICATED FIRE ZONES	ALTERNATE FIkE ZONES		DESIGNATED FIRE ZONES	NONDESIGNATED FIRE ZONES	
AX-23B	AX-21	SC-70B	SB-65	AD-10	MS-52
AX-24	AX-22	TU-22	RC-60	AT-15	MS-53
AX-33	AX-23A	TU-92	TU-99	AX-20	MS-54
SC-70A	AX-23C	TU-93		AX-25	MS-55
TU-90	AX-23D	TU-94		AX-26	MS-56
TU-91	AX-30	TU-95B		AX-27	MS-57
TU-95A	AX-32	TU-98		AX-31	TR-80
TU-95C	AX-34	TC-102		AX-36	TR-81
TU-97	AX-35			AX-39	TR-82
TC-100	AX-37			AX-41	TR-83
TC-101	AX-40			MS-50	TR-84
				MS-51	TR-85
				TU-96	TR-86

For the initial screening, a fire in each dedicated and alternate fire zone that destroys its contents is postulated. Fires in the nondesignated zones are discussed only if they contain components or cables credited in the IPE. Fires in designated zone SB-65 are not discussed because it is a narrow area in which personnel rarely enter during operation, and so transient combustibles, the only credible source of a fire, are kept to a minimum. Designated zone RC-60 is ruled out as discussed in Section 4.1.2. Fires in designated zone TU-99 would not affect the operability of diesels for 8 hours and could not spread to other areas, due to the below ground location of the fuel tanks.

4.1.4.2 Postulated fires in fire zones

4.1.4.2.a Fire in dedicated zones

AX-23B: AX 23B, the "Reactor Auxiliaries North Central", contains components and cabling for many systems required to achieve safe shutdown: Reactor Coolant (RC), Residual Heat Removal (RHR), Safety Injection (SI), Chemical and Volume Control (CVC), Main Steam (MS), CCW, Service Water (SW), Low Voltage Electrical (ELV), Reactor Building Ventilation (RBV), Auxiliary Building Ventilation (ACA), and Nuclear Instrumentation (NI). All these systems are modeled in the IPE. Rooms contained in AX-23B are listed below:

Elevation 566'-6": 14B-5, 14B-7, 14B-9

Elevation 586'-0": 8B-1, 8B-2, 8B-3, 8B-4, 12B-1, 12B-2, 12B-3, 14B, 14B-1, 14B-2, 14B-3

Elevation 606'-0": 153, 153-1, 156, 156-1, 156-2, 156-4, 157, 160, 166, 167

Separation between identified safety-related trains prevents a complete loss of function for most events. For instance, either charging or safety injection is necessary for reactor makeup. Although charging capability could be lost because all charging pumps are in zone AX-23B, safety injection pumps are in a separate fire zone, AX-23A, which is also in a separate fire area (alternate). So total loss of Reactor Coolant System makeup would not occur due to a fire in zone AX-23B.

Reference 7 describes design features that prevent fire-induced LOCAs. LOCA paths were identified, and necessary human actions, such as fuse removal, were incorporated in procedures.

If a fire is postulated in AX-23B, a transient could occur. This zone is retained for further analysis.

AX-24: AX-24, the "Fuel Handling Rooms" contains no equipment modeled in the IPE, with the exception of main steam cabling in 0.75" conduit. The main steam cabling passing through this area is for steam generator A level and pressure transmitters. According to assumption 10, cables in conduit are not vulnerable to the effects of fires of less than one hour in duration. The combustible loading for this zone, AX-24, would produce a fire of 8 minutes duration (Reference 8). This is not long enough to produce enough heat to damage cables within conduit. Because these cables and the other equipment in the fire zone are not vulnerable to fire-induced damage, or are not modeled in the Kewaunee IPE, this zone is not considered further in this analysis.

AX-33: AX-33, "Condensate and reactor makeup water tank room" contains components not vulnerable to fire effects. The only cabling for Appendix R systems in this zone is for the condensate storage tank level indication, which is not absolutely necessary for plant shutdown. Cabling for non-Appendix R systems does not pass through this fire zone, since it is located in the far southern end of the auxiliary building. This zone was inspected during the walkdown to verify that no important components which were credited in the IPE were located in this zone. Due to its large volume and the lack of vulnerable components or cabling in this fire zone, this zone is not considered further in this analysis.

SC-70A: SC-70A, "Screenhouse North", contains components and cabling for the SW, ELV, Turbine Building Ventilation (TAV), and Fire Protection (FP) Systems. A complete loss of service water is not a credible event due to adequate separation between trains. Likewise, a complete loss of ELV, TAV and FP are not credible due to train separation into separate fire areas. A fire in this zone could generate a transient. Therefore, this zone is retained for further analysis.

TU-90: TU-90, "Diesel Generator A Room", contains the A diesel generator, motor control center (MCC) 52A, and safeguards 4160 V electrical bus 5. This zone also contains cabling for many systems: RHR, Auxiliary Feedwater (AFW), SW, Diesel Generator Electric (DGE), High Voltage Electrical (EHV), ELV, TAV, ACA, FP, and the Station and Instrument Air (AS) Systems. Fire-induced disabling of these components could initiate a transient. This zone is retained for further analysis.

TU-91: TU-91, "Diesel Generator A Day Tank Room", contains the day tanks for diesel generator A. The day tanks are not explicitly modeled in the IPE. They are separated from TU-90 by a reinforced concrete barrier with vent holes held open by fusible links. It would take a very large explosive-type fire to damage the barrier. There is enough fuel in the day tanks to accomplish this (750 gallons in each of two tanks), but since they are enclosed and there are no active components, the likelihood is small enough to be considered insignificant compared to the fire initiation frequency of TU-90. A small, leak-type fire would not propagate further. Therefore, this room is not considered further in this analysis.

TU-95A: TU-95A, "Dedicated Shutdown Panel Room", contains safeguards 480 V electrical buses 51 and 52, turbine building basement fan coil unit A, the dedicated shutdown panel, station and instrument air compressor C, diesel start-up air compressor A, as well as cabling for the RC, RHR, SI, CVC, AFW, MS, CCW, SW, Diesel Generator Electric (DGE), EHV, ELV, Emergency AC and DC (EDC), RBV, TAV, ACA, FP, NI, and the Dedicated Analog Control System. A fire in this room that destroys all of its contents would generate a transient. This zone is retained for further analysis.

TU-95C: TU-95C, "Auxiliary Feedwater Pump A Room", contains auxiliary feedwater pump A, and cables for the RC, AFW, MS, SW, DGE, EHV, ELV, EDC, TAV, NI, and the Dedicated Analog Control Systems. A fire in this room would not initiate a complete loss of AFW, since motor driven AFW pump B and the turbine driven AFW pump are in separate fire zones, and protected by appropriate fire barriers. Likewise, fire-induced failure of cabling would not result in a complete loss of function, since redundant trains are in other fire zones. A significant fire in TU-95C could result in a transient. This zone is retained for further analysis.

TU-97: TU-97, "Battery Room A", contains 125 V battery B, 5.0 and 7.5 kVA inverters, MCC 52C, and DC distribution cabinets. Cables for the following systems pass through this room: RC, AFW, MS, DGE, SW, ELV, EDC, TAV, and NI. Adequate separation between TU-97 and areas housing redundant trains of the same systems prevents a complete loss of function for the systems. A fire destroying the contents of this room would initiate a transient or the loss of one 125 V DC train. This zone is retained for further analysis.

TC-100: TC-100, "Technical Support Center, elevation 586'-0"", contains no components required for safe shutdown, but does contain cabling for the RC, RHR, SI, CVC, AFW, MS, CCW, SW, DGE, EHV, ELV, EDC, RBV, TAV, ACA, and NI. The cables are located above a false ceiling. Below the ceiling is the technical support center (TSC) used in emergency. The TSC contains no significant mechanical equipment, only chairs, tables, computers, and other office furniture. It is also a very open area. Therefore, there are no significant transient combustibles of the type considered in the other analyses (i.e., oily rags, etc.). There are quartzoid bulb activated sprinklers that would operate in the event of a fire. Due to the lack of significant flammable materials, this zone is not considered further in this analysis.

TC-101: TC-101, Technical Support Center, elevation 606'-0", contains no safe shutdown components, but does contain cabling for the EDC system, for the non-vital batteries. It is unlikely that a trip resulting from a fire in zone TC-101 would be any worse than a normal transient. Since it is bounded by the IPE, this zone is not considered further in this analysis.

4.1.4.2.b Fire in alternate zones

AX-21: AX-21, "4160 V Switchgear Room", contains 4160 V components for non-safeguards electrical buses 1 and 2, and cables for the EHV and RBV Systems. Adequate separation between areas prevents the possibility of loss of function, whether due to component or cable loss in this room. Significant fire initiation in this room could result in loss of buses 1 or 2, or both, which would cause a reactor trip. The main feedwater pumps are powered from these buses, and are modeled in the IPE. Likewise, the RXCPs are powered from these buses, and their loss would cause a loss of flow, initiating a reactor trip. A significant fire in this room could initiate a transient. This room is retained for further analysis.

AX-22: AX-22, "Machine Shop, Tank and Pump Rooms", contains cabling for the RC, SW, and Control Room Air Conditioning (ACC) Systems. Due to adequate separation, a fire in this zone would not result in a complete loss of function for the above systems. Significant fire initiation could result in a transient. This zone is retained for further analysis.

AX-23A: AX-23A, Refueling Water Storage Tank Area, contains components and cabling for the following systems: RC, RHR, SI, CVC, AFW, MS, CCW, SW, DGE, EHV, ELV, EDC, RBV, ACA, FP, and Internal Containment Spray (ICS), as well as cables for control room lighting. Rooms contained in AX-23A are listed below:

Elevation 586'-0":	8B
Elevation 606'-0":	156-5
Elevation 626'-0":	238, 239, 313
Elevation 642'-3":	304A, 305, 306, 307A, 308, 309, 310
Elevation 657'-6":	403

If a fire is postulated in AX-23A, a transient could occur. Other events are prevented due to adequate separation between trains of components which, if both trains were disabled, could cause a loss of function, and initiate other events. This zone is retained for further analysis.

AX-23C: AX-23C, "RHR Pump B Pit", is room 14B-6 on elevation 566'-6", and contains RHR pump B, and ACA cables. A loss of RHR function is not credible since redundant components are separated by three hour fire barriers. An automatic trip would not occur upon disabling of the contents of this room. This room is retained for further analysis.

AX-23D: AX-23D, "Component Cooling Water Pump B Room", contains CCW water pump B, and ELV and ACA cables. A complete loss of component cooling due to fire is not credible, since CCW pump A is in a separate fire area. If a significant fire were to occur in AX-23D, a transient could occur. This room is retained for further analysis.

AX-30: AX-30, "Relay Room", contains equipment needed for the primary safe shutdown method (via the control room). The remote shutdown panel is used in the event of AX-30 or AX-35 being disabled. If a significant fire were to be initiated in AX-30, a number of events could be initiated if a function were to be disabled: loss of service water, loss of 125 V DC, loss of air, or a transient. This room is retained for further analysis.

AX-32: Rooms located in AX-32, "Service Rooms", are listed below:

Elevation 606'-0":	137, 139, 140, 141, 142, 143, 144, 145, 146, 150, 166, 167, 168, 219
Elevation 626'-0":	230, 230A, 231, 232, 236, 244, 251

These rooms contain no components modeled in the IPE, but they do contain cables for the RC, RHR, SI, CVC, MS, CCW, SW, ELV, EDC, TAV, ACA, ACC, and NI Systems, and for control room lighting. A complete loss of function is not likely, due to adequate separation between trains of redundant components. It is assumed that a transient would be initiated if a fire damaged the aforementioned cabling. This zone is retained for further analysis.

AX-34: AX-34, "Stairwell B", contains no safe shutdown components or cabling. This area is not considered further in this analysis.

AX-35: AX-35, "Control Room and A/C Equipment Room", contains control and indication circuitry for all plant functions controlled and monitored from the control room. A fire in the control room could generate many events: loss of service water, loss of 125 V DC, loss of air, or a transient. This room is retained for further analysis.

AX-37: AX-37, "Control Rod Drive Equipment Room", contains the reactor trip breakers, control rod drive equipment, pressurizer heater components and RC cabling. A fire in this room destroying its contents would initiate a reactor trip. This room is retained for further analysis.

AX-40: AX-40, "Records Storage Room", contains no components or cables modeled in the IPE. For this reason, this room is not considered further in this analysis.

SC-70B: SC-70B, "Screenhouse South", contains components and cables modeled in the IPE: SW, EHV, ELV, TAV, and FP. Although one train of service water and fire protection pumps is located in SC-70B, a complete loss of service water or fire protection (fire pumps) capability is not considered credible, as adequate separation (3-hour fire barriers) exists between trains. A loss of other components in the room could initiate a transient. This room is retained for further analysis.

TU-22: TU-22, "Turbine Room" contains many components credited in the IPE: main feedwater components, non-vital air compressors, MCCs powered from non-safeguards buses, non-safeguards 480 volt and 4160 volt buses, 125 volt non-safeguards DC buses, and condensate pumps. Rooms contained in TU-22 are listed below:

Elevation 586'-0": 6B

Elevation 606'-0": 120, 121, 122, 123, 124, 125, 126, 127, 128

Elevation 626'-0": 199, 200

Failure of components located in these rooms could initiate a transient. This zone is retained for further analysis.

TU-92: TU-92, "Diesel Generator B", contains diesel generator B, safeguards 4160 V electrical bus 6, MCC 62A, and cables for the SW, EHV, ELV, RBV, TAV, ACA, ACC, FP, and ICS Systems. Due to separation between trains, a fire destroying the contents of this room would not cause a complete loss of function. A transient could be initiated. This room is retained for further analysis.

TU-93: TU-93, "Diesel Generator B Day Tank Room", contains the day tanks for diesel generator B. The day tanks are not explicitly modeled in the IPE. They are separated from TU-92 by a reinforced concrete barrier with vent holes held open by fusible links. It would take a very large explosive-type fire to damage the barrier. There is enough fuel in the day tanks to accomplish this (750 gallons in each of two tanks), but since they are enclosed and there are no active components, the likelihood is small enough to be considered insignificant compared to the fire initiation frequency of TU-92. A small, leak-type fire would not propagate further. Therefore, this room is not considered further in this analysis.

TU-94: TU-94, "CO_2 Tank Room", contains air compressor A, and cabling for the RC, RHR, SI, AFW, MS, CC, SW, DGE, EHV, ELV, RBV, TAV, ACA, ACC, FP, AS, and ICS Systems. Adequate separation between TU-94 and areas housing redundant trains of the same systems, prevents a complete loss of function for the systems. A fire destroying the contents of this room could initiate a transient. This zone is retained for further analysis.

TU-95B: TU-95B, "Safeguards Alley" contains safeguards 480 volt electrical buses 61 and 62, auxiliary feedwater pump B, turbine building basement fan coil unit B and the turbine driven auxiliary feedwater pump. Cables contained in this zone belong to the RC, RHR, SI, AFW, MS, CCW, SW, DGE, EHV, ELV, EDC, TAV, ACA, ACC, FP, AS, and ICS Systems. Cables passing through the turbine-driven AFW pump room are located above a steam exclusion ceiling. Adequate separation between TU-95B and areas housing redundant trains of the same systems, prevents a complete loss of function for the systems. A fire destroying the contents of this zone could initiate a transient. This zone is retained for further analysis.

TU-98: TU-98, "Battery Room B", contains 125 volt battery B, 5.0 and 7.5 kVA inverters, MCC 62C, and DC distribution cabinets. Cables for the following systems pass through this room: RC, AFW, MS, SW, DGE, EHV, ELV, TAV, TSC Ventilation (TCV), ACA, ACC, and NI. Adequate separation between TU-98 and areas housing redundant trains of the same systems prevents a complete loss of function for the systems. A fire destroying the contents of this room would initiate a transient, or the loss of one 125 volt DC train. This zone is retained for further analysis.

TC-102: TC-102, "TSC Non Safeguards Battery and Electrical Equipment Room", contains the non safeguards batteries and associated electrical equipment (BRC-108 and BRD-108). The cables for the TCV system pass through this zone. An automatic trip could possibly be generated from a fire occurring in this zone, due to loss of main feedwater control or other non-safety related equipment/cables in this zone. This type of trip is bounded by the IPE (transient without main feedwater). Fire-induced losses of components and cabling in zone TC-102 are not considered further in this analysis, since they were examined in the IPE.

4.1.5 Results of Screening

The first screening results in nine zones being eliminated from further analysis. The following table summarizes the first screening:

Zones that are eliminated	Reason for elimination	Remaining Dedicated Zones	Remaining Alternate Zones
AX-24	Cables/components not vulnerable to fire damage	AX-23B SC-70A TU-90 TU-95A TU-95C TU-97	AX-21 AX-22 AX-23A AX-23C AX-23D AX-30 AX-32 AX-35 AX-37 SC-70B TU-22 TU-92 TU-94 TU-95B TU-98
AX 33	Large volume of room, no components or cabling credited in PRA		
TU 91	Zone does not contain components that were credited in PRA, Concrete barriers prevent propagation		
TC 100	No significant combustibles		
TC 101	IPE transients analysis bounds worst case fire		
AX 34	No components or cables which were credited in PRA		
AX 40	No components or cables which were credited in PRA		
TU 93	Zone does not contain components that were credited in PRA, Concrete barriers prevent propagation		
TC 102	IPE transients analysis bounds worst case fire		

The next screening step is to conservatively calculate core damage frequency assuming the entire contents of the zone are damaged, given a fire, and to determine if that zone's core damage frequency is below the screening value of 1E-06/yr. To do this, the fire initiation frequency is determined for each screened zone, and an inventory of the zone (components and cables) is made. The fire initiation frequency is determined by the EPRI method which allocates the likelihood of a fire based on the room's location in the plant, and the number and type of components in the zone. Both fixed and transient combustibles are considered in this step. The fire initiation frequency is set as the initiating event frequency. The contents of each zone are determined by reviewing the Appendix R documentation, in conjunction with general arrangement drawings, and verified during the walkdown. The fire frequency calculations are documented in Table 4-1. It is postulated that a fire in a certain zone would destroy the contents of that zone. To model this scenario, the Westinghouse WALT code (Reference 13) is used to recalculate core damage frequency. Changes are made to the loss-of-offsite power sequence, assuming total zonal fire-induced damage. The loss-of-offsite power sequence is selected because non-vital loads from buses 1, 2, 3, 4, 32, 35, 42, and 45 were not traced for Appendix R. Appendix R documentation is used at this stage of the analysis. The cables of the non-vital loads, which are not credited, are assumed to be in any of the zones, and vulnerable to fire-induced damage. The results of this screening are documented in Reference 14. As a result of the screening, the following zone is not considered further in the analysis: AX-37. Table 4-1 documents calculation of the fire initiation frequency. Results of the screening are presented in Table 4-2.

The following remaining zones are considered further in the analysis:

DEDICATED	ALTERNATE	
AX-23B	AX-21	SC-70B
SC-70A	AX-22	TU-22
TU-90	AX-23A	TU-92
TU-95A	AX-23C	TU-94
TU-95C	AX-23D	TU-95B
TU-97	AX-30	TU-98
	AX-32	
	AX-35	

The next screening step is to more realistically consider and model fire damage, rather than assume total zonal damage in the event of a fire. To do this, the walkdown was conducted, and measurements required to run COMPBRN were obtained. COMPBRN allows determination of extent and timing of damage, given a fire of a certain size in a certain location.

4.2 Review of Plant Information and Walkdown

Data and information pertinent to this analysis were collected in the plant walkdown. This included the Kewaunee Appendix R documentation, plant specific information pertaining to fire events, fire related procedures, transient combustible control procedures, fire brigade performance and cable routing information. Locations of components and their cables were determined.

The walkdown occurred October 27, 28 and 29, 1992, when the plant was at 100% power. The walkdown team consisted of four WPSC employees and two Westinghouse employees with the following expertise:

Job Title and Company	Expertise
Senior Risk Assessment Engineer - WPSC	PRA modeling of plant systems
Senior Nuclear Engineer - WPSC	1-1/2 years of fire engineering at another utility
Fire Protection Operations Supervisor - WPSC	In charge of fire protection at Kewaunee
Quality Assurance Auditor - WPSC	9 years as Fire Protection Operations Supervisor
Senior Engineer - Westinghouse	Fire PRA modeling
Associate Engineer - Westinghouse	PRA modeling of plant systems

In general, the plant was found to be very clean, and transient combustibles were limited. Except for containment, which was screened from further analysis earlier, the entire plant was inspected. All zones which were retained for further analysis were examined in detail.

Some zones were excluded from further analysis based on walkdown findings. The following paragraphs discuss these exclusions.

Cable Spreading Area of AX-32: All cables in this area are located in trays. There are numerous ionization detectors in this area. Wet pipe sprinklers of the quartzoid bulb design are located close to the cable trays, aimed at the cable trays. This area is under strict administrative control. No transient combustibles are stored in or near the zone. If maintenance such as cable pulling were to occur, it would be done during shutdown conditions. This analysis is limited to at-power conditions. There is a 480 V/120 V transformer in one corner of the room. A transformer fire is unlikely, especially with the relatively low voltages involved. There are cable trays located about six feet from the transformer. If it were to occur, a transformer fire would be of short duration. Because of the low voltages involved, and the short duration of a postulated fire, it is assumed that insufficient energy would be released to cause damage to the cables which are a minimum of six feet away. For the above reasons, this area is not considered further in this analysis.

Auxiliary Building (AX-23A and AX-23B): The controlled part of the auxiliary building is large and open. Formation of a hot gas layer is unlikely, since there are no confined spaces. Scenarios that could contain vulnerabilities are examined in detail, but most of the auxiliary building is not considered further in this analysis due to the large open layout.

Control Room (AX-35): The control room is continually manned. Cabinets in the control room have smoke detectors inside the cabinets. Studies have shown (Reference 16) that cabinet fires are confined to cabinets where the fire originated. Circuitry within these cabinets operate at low voltages. Although a fire in the control room is unlikely, the plant can still be safely shut down from the dedicated shutdown panel. The control room and the dedicated shutdown panel are in separate fire areas, so that a single fire could not disable both areas. Procedures E-0-06 and E-0-07 describe actions that must be taken in the event of a fire which causes loss of monitoring or control function (e.g., control room fire). Control room fires are not considered further in this analysis.

Turbine Building (TU-22): Due to its large area and volume, and lack of components and cabling required for safe shutdown, fires in the turbine building are not analyzed, except for individual scenarios in which a fire could damage components or cabling credited in the PRA.

Screenhouse (SC-70A and SC-70B):

SC-70A was chosen as being representative of SC-70B. The two zones are normally open to each other, and can be isolated by roll-up fire doors held open by fusible links. Each zone has a large exhaust fan (50,000 cfm) located near the vital MCCs (MCC 52D, MCC 62D). It is not feasible that a damaging hot gas layer would form near these MCCs, as any hot gases would be exhausted outdoors by the fans. Radiative damage is not credible, due to the spatial separations involved. The non-vital MCCs (MCC 35C and MCC 45C) are located in SC-70A, away from the exhaust fans. A fire near these MCCs in SC-70A is analyzed.

The most likely size of pilot fire was discussed with Kewaunee fire safety professionals. It was concluded that a trash can fire equivalent to combustion of 3 kg of heptane, or a fire resulting from a lube oil spill, are the most likely fire initiators.

4.3 Fire Growth and Propagation

Propagation from adjacent zones is considered. In accordance with assumption 13, Appendix R barriers are assumed to remain intact, and inter-zonal propagation is not found to present a hazard. Zonal boundaries and intervening combustibles were examined during the walkdown. Combustibles were found to be evenly distributed in the zones. Combustible loadings presented in Reference 8 were found to be applicable to this analysis.

Fire growth was modeled with COMPBRN, which is documented in Reference 16.

4.3.1 COMPBRN Runs

Following is a list of COMPBRN runs that were made, to determine extent and timing of damage.

- AX-23B
  - Scenario 1: Cable trays near flammable liquid storage cabinet, 586' elevation
  - Scenario 2: Cable trays near bottled gas storage, 586' elevation
- AX-23A
  - Scenario 1: Cable trays above SI pumps
  - Scenario 2: Cable trays near bottled gas storage, 657'-6" elevation
  - Scenario 3: Cable trays near MCC-35E, -45E
- TU-95A
  - Scenario 1: Cable trays above Bus 52
- TU-95C
  - Scenario 1. Cable trays above AFW Pump A
- TU-95B
  - Scenario 1: Cable trays above Bus 61
  - Scenario 2: Cable trays north of Bus 62
  - Scenario 3: Cable trays (2 trains) in AFW pump B room
- TU-90
  - Scenario 1: Cable trays above Bus 5
  - Scenario 2: Cable trays above MCC 52A
  - Scenario 3: Other cable trays in room
- TU-92
  - Scenario 1: Cable trays above Bus 6
  - Scenario 2: Cable trays above MCC 62A
  - Scenario 3: Other cable trays in room
- SC-70A
  - Scenario 1: Cable trays above MCC 35C and MCC 45C
- TU-97
  - Scenario 1: Cable trays above distribution cabinets
  - Scenario 2: Cable trays above batteries
- TU-98
  - Scenario 1. Cable trays above distribution cabinets
  - Scenario 2: Cable trays above batteries

- AX-30
  - Scenario 1: Fire near vertical cable trays
  - Scenario 2: Fire near cabinets (cables exposed on top of cabinets)
- AX-32
  - Scenario 1: Cables leading to MCC 62B
- AX-23D
  - Scenario 1: Cables near CCW pump
- TU-94
  - Scenario 1: Cable trays near air compressor A
- AX-21
  - Scenario 1: Cable tray above buses 1 and 2

4.3.1.1 TU-95A, Switchgear room 5B

4.3.1.1a Scenario: fire near safeguards 480 V electrical bus 52

A trash can fire is modeled by a 3 kg pool of heptane. TU-95A is modeled two ways--with ventilation and without ventilation. The case with ventilation yields no damage to any of the cable trays located directly above the heptane pool. The case without ventilation yields similar results. Both of these cases model a heptane pool at floor level. When the heptane pool is raised to 0.5 meters above floor level (estimated height of full trash can), both models yield damage to cable tray 1TT5S5 within one minute. Other cable trays and the adjacent MCC are not damaged. Both cases (with and without ventilation) with the heptane pool raised 0.5 m from the floor yield damage to cable tray 1TT5S5. The fusible link closure temperature on the fire dampers is 160 deg. F or 341 deg. K. The fire that is modeled (3 kg of heptane) causes closure of the dampers for the case with ventilation and without ventilation, since the hot gas layer of 5.6 m reaches a mean temperature of 385 deg. K without ventilation and 335 deg. K (close to 341 deg. K) with ventilation. The more conservative case with no ventilation is considered. Cable tray 1TT5S5 is affected with a fuel size of 3 kg heptane located 0.5 meters above the floor. Other cable trays 1TT1S5, 1TT1N, 1TT12N, 1TT11N and 1TT13N are not affected by the postulated fire. Ladder trays 1TL53S5, 1TL1S8, 1TL1N and 1TL14N are not considered in the analysis, since they contain power cables, and all power cables are enclosed in conduit, which is not considered to be vulnerable to heat damage for fires of less than one hour duration. The other train of cables, 1TT5S6, 1TT1S6, 1TT6N, and 1TT16N are routed above an Appendix R false ceiling, and are not considered to be vulnerable to damage.

It is concluded that a fire of 3 kg of heptane with no ventilation may damage cable tray 1TT5S5.

4.3.1.2 TU-97, Battery room A (room 129)

4.3.1.2a Scenario: fire near distribution cabinets

This room contains many cables, but most are power cables in conduit. Cables in conduit are not considered to be vulnerable to fires of less than one hour duration. There also are many distribution cabinets in the room. Cables entering the cabinets are considered to be vulnerable to fire damage. This room is modeled with and without ventilation. The room ambient temperature exceeds the temperature at which the fusible links in the dampers actuate in both cases, so the cases with no ventilation are considered. Three (3) kg of heptane located near the distribution cabinets, at floor level, and elevated 0.5 m yield no damage to the distribution cabinet or the overhead cable trays. However, if the amount of fuel was increased to 30 kg of heptane (not realistic), damage occurs to the distribution cabinet and the overhead cable trays. Thirty (30) kg of heptane, representing a large trash can, was determined by the Kewaunee fire safety professionals to be an unrealistic amount of fuel. Note that there were no transient combustibles located in TU-97 during the walkdown, and the plant fire protection safety professionals confirmed that there are none in this room during operations.

4.3.1.2b Scenario: fire near batteries

Where a 3 kg heptane pool elevated 0.5 m from floor level is modeled near the batteries, no damage to either the batteries or the overhead cable trays occurs. When the 3 kg heptane pool is replaced with a 30 kg pool, no damage to the batteries occurs, but cable tray 1TT132N is damaged. As described earlier, 30 kg of heptane is an unrealistic amount of fuel.

It is concluded that a fire of realistic magnitude (represented by 3 kg of heptane) in the battery room poses no threat to components or cables in the room.

4.3.1.3 TU-98, Battery room B (room 130)

4.3.1.3a Scenario: fire near distribution cabinets

Like battery room A, battery room B has many cables passing through it. Most cables are power cables in conduit. Ventilation and no ventilation are modeled, and in both cases the room ambient temperature exceeds the damper closure temperature of 341 deg. K. The case with no ventilation represents the battery room under fire conditions. Three (3) kg of heptane elevated 0.5 m from floor level, located near the distribution cabinets, yields no damage to the distribution cabinets or the overhead trays. When 30 kg of heptane in the same location is modeled as the fuel source, damage does occur to the distribution cabinet and to the overhead cable trays. However, 30 kg of heptane is an unrealistic amount of fuel.

4.3.1.3b Scenario: fire near batteries

When the fire is modeled near the batteries, a 3 kg pool of heptane elevated 0.5 m from the floor yields no damage to the batteries or the overhead cable trays. However, a 30 kg pool of heptane in the same location does cause damage to both the batteries and overhead cable trays. As described above, 30 kg of heptane is an unrealistic amount of fuel.

It is concluded that a fire of realistic magnitude (represented by 3 kg of heptane) in the battery room poses no threat to components or cables in the room.

4.3.1.4 AX-23A, Auxiliary building (RWST area)

4.3.1.4a Scenario: fire near MCC 35E/45E in fan room

The fan room is a large area, and this fire scenario is modeled as an open room. Three (3) kg of heptane located 0.5 m from floor level is modeled as the fuel source. This fire causes no damage to the MCCs or the overhead cable trays. It is concluded that a fire poses no threat to the MCCs or to the overhead cable trays.

4.3.1.4b Scenario: fire in corridor near MCC 62J

This corridor is small, and is modeled as a closed room with no ventilation. A postulated fire of 3 kg of heptane causes damage to MCC 62J, and cable trays 1FT6S6 and 1FX4S6. Note that in reality, it is very unlikely that transient combustibles would be in this corridor, as it is too small to permit trash storage and personnel traffic. However, for the purpose of this analysis, this fire is considered.

4.3.1.4c Scenario: fire near bottled gas storage at elevation 657'-6"

The bottled gas storage area at elevation 657'-6" is modeled as an open area with a fire of 3 kg of heptane, elevated 1.5 m above the floor to represent a bottled gas leak. Damage to overhead cable tray 1FT10N occurs.

4.3.1.4d Scenario: SI pump oil fire

A lubricating oil leak from a SI pump is postulated, to determine if the resulting fire endangers overhead cable trays. The SI pumps are not in a closed room, so an open area was modeled. The fuel that was modeled is engine oil, representing lubricating oil, at floor level, with the SI pump between the oil pool and the overhead cable trays. No damage to the overhead cable trays resulted.

4.3.1.5 AX-23B, Auxiliary building (reactor auxiliaries north center)

4.3.1.5a Scenario: fire near flammable liquid storage cabinet at elevation 586'-0"

A fire in the flammable liquid storage cabinet at elevation 586'-0" is modeled by a 3 kg heptane pool. The flammable liquid storage cabinet is in an open corridor, so this scenario is modeled as an open area. No damage to overhead cable trays 1AT3S5 or 1AT33N results. The flammable liquid storage cabinet, however, has been moved to a site in same compartment but farther from the cable trays.

When a 30 kg heptane fire is modeled, damage to the two cable trays results. It is unlikely that combustibles equivalent to 30 kg of heptane would reside in an open flammable liquid storage cabinet. The scenario with 3 kg of heptane is more realistic, and is considered.

4.3.1.5b Scenario: fire near bottled gas storage at elevation 586'-0"

3 kg of heptane elevated 1.58 m (to represent the height of a bottled gas cylinder) is used to model a fire in which a hose breaks loose from its bottle, and the leaking gas ignites. The bottled gas cylinders are located in an open hallway, so this scenario is modeled as an open area. Four cable trays, 1AT1N, 1AT23N, 1AT53N, and 1AT4S5 are located near the gas cylinders. No damage to these cable trays results.

A 30 kg heptane fire is modeled, and damage to two of the four cable trays results. However, it is unlikely that a fire equivalent to 30 kg of heptane would be initiated, with the strict transient combustible controls at Kewaunee. The 3 kg scenario is considered in this analysis.

4.3.1.6 TU-90, Diesel generator room A

Note that this room has a CO_2 flooding system, which is actuated by heat detectors in the room, or manually.

4.3.1.6a Scenario: fire near safeguards 4160 V electrical bus 5

Three (3) kg of heptane elevated 0.5 m and situated near safeguards 4160 V electrical bus 5 is postulated to model a trash can fire, and to determine if such a fire harms bus 5 or the overhead cable trays. Because the fire dampers close on high temperature (160 deg. F), and on CO_2 and thermal detector actuation, it is postulated that there is no ventilation if a fire occurs. Hence, TU-90 is modeled as a closed room. Neither the bus nor the overhead cable trays are damaged in this scenario.

Even when a large (and unrealistic) 30 kg heptane fire is modeled, no damage to either the bus or the overhead cable trays results, due to the large volume of the room.

4.3.1.6b Scenario: fire near MCC 52A

Three (3) kg of heptane located near the MCC is modeled to represent a trash can fire. A closed room is modeled in this scenario. No damage to the MCC or the overhead cable trays results.

However, a 30 kg heptane fire located near the MCC does cause damage to the MCC and the overhead cable trays. Upon review of transient combustible controls at Kewaunee, it was determined that a 30 kg heptane fire representing a full 55 gallon trash can was unrealistic. Hence this scenario was ruled out.
It is concluded that a fire near MCC 52A does not damage the MCC or the overhead cable trays.

4.3.1.6c Scenario: diesel engine oil fire

For this scenario, a 300 gallon engine oil fire below the diesel generator is postulated. As above, a closed room scenario is modeled. 300 gallons of engine oil (the entire inventory) is postulated to burn. The two overhead cable trays 1DT3S5 and 1DT4S5 are not damaged.

It is concluded that a diesel engine oil fire does not damage the overhead cable trays.

4.3.1.7 TU-92, Diesel generator room B

Similar to TU-90, this room has a total flooding CO_2 system, actuated manually or by heat detectors.

4.3.1.7a Scenario: fire near MCC 62A

Three (3) kg of heptane located near the MCC is selected to represent a trash can fire. A closed room is modeled in this scenario. Damage to MCC 62A results, and COMPBRN shows a .495 probability of cable tray 1DT3S6 being damaged. For this analysis, it is assumed that the MCC and cable tray 1DT3S6 are damaged if a trash can fire occurs near the MCC.

When a large (and unrealistic) 30 kg fire is modeled, both the MCC and the overhead cable tray are damaged.

This scenario yields damage while the scenario for MCC 52A does not, due primarily to the different geometries of TU-90 and TU-92.

4.3.1.7b Scenario: fire near safeguards 4160 V electrical bus 6

A 3 kg heptane fire elevated 0.5 m from the floor is modeled near safeguards 4160 electrical bus 6. As above, a closed room scenario is modeled. No damage to the bus or the overhead cable trays results. However, a 30 kg heptane fire elevated 0.5 m from the floor does cause damage to the bus and overhead cable tray 1DT4S6.

Upon review of transient combustible controls at Kewaunee, it was determined that a 30 kg heptane fire representing a full 55 gallon trash can is unrealistic. Hence this scenario is ruled out. It can be concluded that a fire near bus 6 does not damage it or the overhead cable trays.

4.3.1.7c Scenario: diesel engine oil fire

For this scenario, 300 gallons of engine oil below the diesel generator is postulated to burn. As above, a closed room scenario is modeled. Overhead cable tray 1DT8N is damaged.

4.3.1.7d Scenario: trash can fire located beneath cable trays at north end of room

Thirty (30) kg of heptane located beneath cable trays is modeled to represent a very large trash can fire. A closed room scenario is modeled. The overhead cable trays show no damage, due to the large volume of the room.

4.3.1.8 SC-70A, Screenhouse

4.3.1.8a Scenario: fire near MCC 35C/45C

Three (3) kg of heptane elevated 0.5 m from the floor, and located near MCC 35C/45C and overhead cable tray 1ST3N produces a fire yielding no damage to either the MCC or the cable tray. However, a 30 kg fire in the same location damages both the MCC and cable tray 1ST3N. Upon review of transient combustible controls at Kewaunee, it was determined that a 30 kg heptane fire representing a full 55 gallon trash can is unrealistic. Hence this scenario is ruled out.

It is concluded that a fire near the MCCs does not damage it or the overhead cable trays.

4.3.1.9 AX-30, Relay room

4.3.1.9a Scenario: fire near vertical cable trays

Three (3) kg of heptane located near vertical cable trays 1AT3N, 1AT4N, 1RT62S6, 1AT12S6, and 1AT13S6 is modeled. This is modeled with ventilation, since the hot gas layer temperature does not exceed the damper closure temperature with a fuel source of 3 kg of heptane. It is a large room. All these cable runs are Flameastic coated from floor level to 4 ft high. Therefore the first vulnerable section of cable is considered to be 4 ft high. Damage to cable trays 1RT62S6, 1AT12S6, and 1AT13S6 results.

4.3.1.9b Scenario: fire near four cabinets

Three (3) kg of heptane elevated 0.5 m from the floor, representing a trash can, is modeled 0.5 m from four cabinets. As in the above scenario, the hot gas layer temperature does not exceed the fire damper closure temperature, so ventilation is modeled. No damage to the cabinets or the overhead cable trays results. When this same scenario is modeled with 30 kg of heptane as the fuel source, still no damage results.

4.3.1.10 TU-95C, AFW pump A room

4.3.1.10a Scenario: AFW pump A oil fire

Fifteen (15) kg of engine oil is modeled at floor level below AFW pump A to simulate a lubricating oil leak to determine if this causes damage to the overhead cable trays. The pump is located in a small room, and the ambient temperature in the simulation for two cases - with and without ventilation - exceeds the damper closure temperature of 341 deg. K. There is uncertainty associated with the damage computation and COMPBRN shows a probability of damage to all cable trays to be 0.52. For this analysis, damage to all cable trays in the room (1TT8S5, 1TT4S5, 1TT4N, 1TT15N, and 1TT106S5) is assumed.

4.3.1.11 AX-32, Service rooms

4.3.1.11a Scenario: fire near MCC 62B

Three (3) kg of heptane located near MCC 62B, elevated 0.5 m from the floor is modeled to represent a full trash can. No damage to the MCC or the overhead cable trays results, due to the open area of AX-32.

4.3.1.11b Scenario: fire near MCC 62B extension

Three (3) kg of heptane located near MCC 62B extension, elevated 0.5 m from the floor is modeled to represent a full trash can. No damage to the MCC results, due to the open area of AX-32.

4.3.1.11c Scenario: fire in cable spreading area

There are limited transients combustibles allowed in this area. All power cables are armored. There are numerous ionization detectors located in this area, and wet pipe sprinklers (actuated by a quartzoid bulb) aimed at the numerous cable trays in this area. There is one 480/120 V transformer in the area, which could be a potential ignition source, but there are no combustibles located nearby. In addition, a transformer fire is not likely. A transformer fire is short and intense, but it is assumed that there would be insufficient energy released to damage the nearby (6 ft. away) cable trays. The main steam and feedwater lines pass through this area, but they are insulated and jacketed, minimizing heat release. There is no maintenance in this area during power operations. For these above reasons, it is concluded that a fire in the cable spreading area during power operations is not credible.

4.3.1.12 AX-23D: Component cooling water pump B room

4.3.1.12a Scenario: Component cooling water pump B oil fire

Fifteen (15) kg (approximately 5 gallons) of lubricating oil is modeled at floor level to represent an oil spill, the most probable fire initiator in the room. Within 14 minutes, damage occurs to cable tray 1AT20N. Moving the lubricating oil spill around to determine the area of influence over which it causes damage has no effect, because the damage occurs only due the hot gas layer. For this analysis, it is assumed that, given a fire in this room, everything in the room is lost.

4.3.1.13 TU-94: Air compressor A room

4.3.1.13a Scenario: fire in air compressor A

When 15 kg of lube oil is modeled beneath air compressor A, no damage to the air compressor or the overhead cable trays results. The average hot gas layer temperature is 411 deg. K, not high enough to cause damage.

4.3.1.13b Scenario: fire beneath cable trays

Three (3) kg of heptane, representing a trash can fire, located near air compressor A and beneath cable trays 1DT6S6, 1DT7S6 and 1TT5N results in no damage. When the size of the fuel source is increased to a 60 kg heptane fire (not realistic), still no damage results. The hot gas layer temperature does not exceed the cable damage threshold, due to the large volume of the room. The average temperature during the simulation was 397 deg. K for the 3 kg heptane case, and 444 deg. K for the 60 kg heptane case. Neither temperature exceeds the cable damage threshold, hence no damage occurs.

4.3.1.14 TU-95B: AFW pump B room and 480 V swgr buses 61 and 62 room

4.3.1.14a Scenario: AFW pump B oil fire

Fifteen (15) kg (5 gallons) of engine oil located beneath AFW pump B at floor level is modeled to represent a lubricating oil leak. The hot gas layer temperature exceeds the damper closure temperature of 341 deg. K for two cases - with and without ventilation. The mean hot gas layer temperature for the case with no ventilation is 665 deg. K, and 5.60 m in thickness. This temperature exceeds the damage threshold temperature of the cables, 500 deg. K, and indicates damage to all cables in the room. There is uncertainty associated with the damage computation, and COMPBRN shows a probability of damage to all cable trays in the room to be 0.05. For this analysis, damage to all the cable trays in the room (1TT8S5, 1TT4S5, 1TT4N, 1TT15N, 1TT8S6, 1TT4S6, 1TT10N, and 1TT20N) is assumed.

4.3.1.14b Scenario: fire beneath cable trays north of safeguards 480 V electrical bus 62

Three (3) kg of heptane is modeled 0.5 m from the floor to simulate a full trash can located directly beneath the train B cable trays (1TT7S5, 1TT3S5, 1TT3N, 1TT14N). The hot gas layer temperature exceeded the damper closure temperature of 341 deg. K for two cases - with and without ventilation. The case with no ventilation is the one considered. The mean hot gas layer temperature is 402 deg. K, below the damage threshold temperature of the cables. There is no damage to the cable trays.

4.3.1.14c Scenario: fire between safeguards 480 V electrical buses 61 and 62

Three (3) kg of heptane are modeled 0.5 m from the floor to simulate a full trash can located between safeguards 480 V electrical buses 61 and 62. The hot gas layer temperature exceeds the damper closure temperature of 341 deg. K for two cases - with and without ventilation. The case with no ventilation is considered. The mean hot gas layer temperature is 402 deg. K, below the damage threshold temperature of the cables. As a result, no damage to the cable trays results.

4.3.1.15 AX-21: 4160 V switchgear room

4.3.1.15a Scenario: fire near non-safeguards 4160 V electrical buses 1 and 2

Three (3) kg of heptane are modeled 0.5 m from the floor to simulate a full trash can located between non-safeguards 4160 V electrical buses 1 and 2. The hot gas layer temperature does not exceed the damper closure temperature of 341 deg. K, so the case with ventilation is considered. The mean hot gas layer temperature is 321 deg. K, below the cable damage threshold temperature. However, cable tray 1AT9N shows damage within the first minute, due to radiant heat exposure.

The results from these COMPBRN runs are summarized below, and are also shown in Table 4-3.

Room	Damaged Cable Trays/Components
TU-95A	1TT5S5
TU-97	no damage
TU-98	no damage
AX-23A:	
- MCC 35E/45E	no damage
- MCC 62J	1FT6S6, 1FX4S6
- bottled gas, el. 657'	1FT10N
- SI pump	no damage
AX-23B:	
- flammable liquid storage cab	no damage
- bottled gas, el. 586'	no damage
TU 90:	
- bus 5	no damage
MCC 52A	no damage
- diesel fire	no damage
TU 92:	
- MCC 62A	1DT4S6, MCC 62A
- bus 6	no damage
diesel fire	1DT8N
SC-70A	no damage
AX 30:	
fire near vertical trays	1RT62S6, 1AT12S6, 1AT13S6

- fire near cabinet	no damage
TU-95C	1TT8S5, 1TT4S5, 1TT4N, 1TT5N, 1TT106S5
AX-32:	
- MCC 62B	no damage
- MCC 62B ext	no damage
- cable spreading area	not modeled*
AX-35	not modeled*
AX-23D	1AT20N
TU-94:	
- air comp A fire	no damage
- beneath cable trays	no damage
TU-95B:	
- cable trays above bus 61	no damage
- cable trays north of bus 62	no damage
- cable trays in AFW B room	1TT8S5, 1TT4S5, 1TT4N, 1TT15N, 1TT8S6, 1TT4S6, 1TT10N, 1TT20N
- turbine-driven AFW pump room	not modeled**
AX-21	1AT9N

* Neither the cable spreading area nor the control room are modeled. These rooms are discussed in Section 4.2.

** All cable trays are located above the steam exclusion ceiling.

The area of influence for the fires causing damage is determined by iterating the COMPBRN runs while moving the fire source around. Table 4-3 shows the area of influence for each scenario.

Those scenarios showing no damage are not analyzed further.

4.4 Evaluation of Component Fragilities and Failure Modes

If a component or cable is determined to be damaged by fire, then it is assumed to fail with a probability of 1.0. In the case of the AFW pump rooms, damage to the cable trays in the room is estimated by COMPBRN to occur with a probability of 0.52 for the A pump room and 0.05 for the B pump room. This figure is factored in the initiating event frequency calculation for the AFW pump B room.

4.5 Fire Detection and Suppression

4.5.1 Automatic and Manual Fire Detection and Suppression

Manual fire detection is based upon room occupancy. All rooms except the control room are assumed to be unoccupied. Manual fire detection elsewhere is based upon maximum patrol interval, which is determined to be two hours.

Manual fire suppression capability is demonstrated by plant-specific documentation, which documents fire brigade drills from 1988 to 1991. It is concluded that the response time from alarm initiation to assembly of the fully turned out brigade is 6.0 minutes for the turbine building and screenhouse, and 6.4 minutes for the auxiliary building.

Automatic fire detection is accomplished by ionization detectors, which are described in the NFPA Handbook (Reference 17). Thermal detectors also exist in the diesel generator rooms.

In the initiating event frequency calculations, automatic fire suppression is listed for each scenario. Automatic suppression is assumed to be initiated instantaneously upon successful fire detection.

4.6 Analysis of Plant Systems, Sequences, and Plant Responses

4.6.1 Identification of Components Affected by Fire

After the vulnerable cable trays are identified, their contents are determined. Table 4-4 lists the contents of the cable trays that were found to be vulnerable to damage in the postulated fire scenarios. This table also lists the fault tree identifiers for the damaged cables/components, and the fault trees in which the identifiers appear.

When determining the contents of cable trays that had been identified as vulnerable to damage, a number of cable trays were found to contain only cables for components that are not modeled in the IPE. As a result, a number of scenarios are not analyzed in detail. The cable trays that contain only cables for components that are not modeled in the IPE are listed below:

Cable tray	Room in which cable tray resides
1AT20N	AX-35
1FX4S6	AX-23A, near MCC 62J
1FT10N	AX-23A, near bottled gas at elevation 657'-0"

4.6.2 Determination of Initiating Event Frequency in Each Zone Requiring Quantification

Figures 4-1 through 4-8 show the derivation of the initiating event frequency for each scenario. The transient with main feedwater, transient without main feedwater or loss of offsite power event sequence was used depending on the scenario.

4.6.3 Fire-Induced Core Damage Frequency Quantification

Core damage frequencies are computed for each fire scenario. Table 4-5 shows results from the eight scenarios that require quantification of core damage frequency (CDF). The fire CDF is 9.81E-05, which is about the same as the internal events CDF (8.74E-05). Tables 4-7 through 4-14 show the results for each scenario. Following is a discussion of the results from quantification of core damage frequency for each fire scenario.

Results show that fires in the AFW pump rooms (Scenarios FI6 and FI7) dominate all the scenarios requiring quantification.

4.6.3.1 FI1: Fire near MCC 62J in AX-23A

A fire near this MCC, which is located in a closed corridor, has a small impact on the plant. Table 4-7 lists the top 50 dominant cutsets for this scenario. The transient with main feedwater available event tree (Figure 4-9) is used in this scenario.

Since this fire does not prohibit the operation of safe shutdown equipment from the control room, it is assumed operators do not use procedure E-0-06, Fire in Alternate Zone, which requires manual isolation of offsite power and evacuation of the control room.

4.6.3.2 FI2: Fire near MCC 62A in TU-92

A fire near this MCC, which is located in the B diesel generator room, is postulated. Cable tray 1DT4S6 and the MCC are affected. Table 4-8 shows the top 50 dominant cutsets for this scenario. The transient with main feedwater available event tree (Figure 4-9) is used in this scenario.

Since this fire does not prohibit the operation of safe shutdown equipment from the control room, it is assumed operators do not use procedure E-0-06, Fire in Alternate Zone, which requires manual isolation of offsite power and evacuation of the control room.

4.6.3.3 FI3: Fire in AX-21, near non-safeguards 4160 V electrical buses 1 and 2

A postulated fire near non-safeguards 4160 V electrical buses 1 and 2 damages cable tray 1AT9N, containing main feedwater cabling. The cabling leading to the buses is not damaged. Table 4-9 shows the top 50 dominant cutsets for this scenario. The transient without main feedwater event tree (Figure 4-10) is used in this scenario.

Since this fire does not prohibit the operation of safe shutdown equipment from the control room, it is assumed operators do not use procedure E-0-06, Fire in Alternate Zone, which requires manual isolation of offsite power and evacuation of the control room.

4.6.3.4 FI4: Diesel fire in TU-92

Cable tray 1DT8N is damaged in this postulated fire scenario. Table 4-10 shows the top 50 dominant cutsets for this scenario. Due to the loss of the RAT, which supplies power to buses 1 through 4, a modified loss of offsite power sequence (Figure 4-11), which allows the TAT to supply bus 5, is used in this scenario.

Since this fire does not prohibit the operation of safe shutdown equipment from the control room, it is assumed operators do not use procedure E-0-06, Fire in Alternate Zone, which requires manual isolation of offsite power and evacuation of the control room.

4.6.3.5 FI5: Fire in relay room near vertical cable trays

Safeguards 4160 V electrical bus 6 and cable trays 1RT62S6, 1AT12S6, and 1AT13S6 are disabled in this postulated fire scenario. Table 4-11 shows the dominant cutsets for this scenario. The dedicated shutdown system event tree (Figure 4-12) is used in this scenario.

Due to the proximity of the relay room to the control room and the chance of the fire spreading and affecting safe shutdown capability from the control room, it is assumed that the operators evacuate the control room and go to the dedicated shutdown panel using procedure E-0-06, Fire in Alternate Zone. Therefore, it is assumed that offsite power is not available and that operators manually restore power to safeguards 4160 V electrical bus 5.

4.6.3.6 FI6: AFW pump A oil fire

This scenario is still analyzed, despite the 0.48 probability of no damage to the cable trays. All cable trays in this room and the AFW pump A are assumed damaged. Table 4-12 shows the top 50 dominant cutsets for this scenario. The alternate shutdown system event tree (Figure 4-13) is used for this scenario. Fire-induced disabling of the vulnerable cable trays causes damage to cables that control components vital to safe shutdown, giving rise to a high core damage frequency.

Operators are instructed to use procedure E-0-07, Fire in Dedicated Zone, which requires manual isolation of offsite power, when train A equipment is not available. Since this is the case for this fire, it is assumed that offsite power is not available and that operators manually restore power to safeguards 4160 V electrical bus 6.

4.6.3.7 FI7: AFW pump B oil fire

This scenario is still analyzed, despite the 0.95 probability of no damage to the cable trays. All of the following cable trays show susceptibility to damage: 1TT8S5, 1TT4S5, 1TT4N, 1TT8S6, 1TT4S6, 1TT10N, and 1TT20N. Electrical buses 3, 4, and 6 are disabled due to damage to the contents of these trays. Table 4-13 shows the top 50 dominant cutsets for this scenario. The dedicated shutdown system event tree (Figure 4-12) is used for this scenario.

Fire-induced disabling of the vulnerable cable trays causes damage to cables that control components vital to safe shutdown, giving rise to a high core damage frequency. Failure of safeguards 4160 V electrical bus 6 is assumed because the cables for the source breakers for these components are vulnerable to fire-induced damage.

Since this fire renders operation of equipment from the control room impossible, it is assumed that the operators evacuate the control room and go to the dedicated shutdown panel using procedure E-0-06, Fire in Alternate Zone. Therefore, it is assumed that offsite power is not available and that operators manually restore power to safeguards 4160 V electrical bus 5.

4.6.3.8 FI8: Fire near safeguards 480 V electrical buses 51 and 52

Cable tray 1TT5S5 and the source breaker cabling for safeguards 4160 V electrical bus 5 are damaged in this scenario. Table 4-14 shows the dominant cutsets for this scenario. The alternate shutdown system event tree (Figure 4-13) is used in this scenario.

Operators are instructed to use procedure E-0-07, Fire in Dedicated Zone, which requires manual isolation of offsite power, when train A equipment is not available. Since this is the case for this fire, it is assumed that offsite power is not available and that operators manually restore power to safeguards 4160 V electrical bus 6.

4.6.3.9 Core damage timing

Because the accident sequences induced by a fire are similar to those analyzed for the internal events analysis, fire-induced core damage timing is similar to that determined for the internal events analysis. The core damage timing for fire induced events is shown in Tables 4-9 through 4-13.

4.6.4 Dominant Fire-Induced Core Damage Sequences

Table 4-15 lists the 22 systemic core damage sequences of the Fire PRA. Of these, 13 are dominant according to the definition provided in NUREG-1335. These sequences represent 99.95% of the total core melt frequency. Following is a description of each of these dominant sequences. The frequency and percent contribution of each sequence is given in Table 4.B-9.

- Sequence #1 - AFW pump A oil fire followed by failures of AFW and bleed and feed.

  This results in a loss of heat sink and early core damage. The primary means by which heat sink is lost is a station blackout. Since offsite power and the A diesel generator are unavailable as a result of the fire, a failure of the B diesel generator or the B train of service water, which cools the diesel generator, results in a station blackout. Since the fire also disables control cabling for the turbine driven AFW pump, there is no heat sink and therefore core melt occurs. The primary human errors in this sequence are failure to manually establish electrical power from the B diesel generator and failure to locally restore instrument air for the pressurizer PORVs, used for bleed and feed.

- Sequence #2 - AFW pump B oil fire followed by failure of AFW.

  This results in a loss of heat sink and early core damage. Other means of removing heat, (main feedwater, condensate, bleed and feed) are unavailable due to the unavailability of offsite power and safety injection pumps from the DSP. This sequence is dominated by human error. Failure to establish electrical power from the A diesel generator locally and failure to establish service water and AFW from the DSP are major contributors to this sequence.

- Sequence #3 - AFW pump A oil fire followed by failure of CCW.

  This results in a loss of cooling to the RXCP thermal barrier, which can result in a small LOCA due to RXCP seal failure. Since SI and RHR pumps are cooled by CCW and charging is not available because of cabling damaged by the fire, there is no way to get makeup flow to the Reactor Coolant System to replace that lost through the RXCP seals and late core damage results. The only human error in this sequence is the failure to manually restart CCW and this is a relatively minor contributor.

- Sequence #4 Fire near MCC 62A in B diesel generator room followed by a failure of charging and CCW.

  This results in a loss of cooling to the RXCP seals and late core damage. The primary means of loss charging and CCW is loss of power to the emergency buses, buses 5 and 6. Human error is not important in this sequence.

- Sequence #5 - Oil fire in B diesel generator followed by a failure of charging and CCW.

  This results in loss of cooling to the RXCP seals and late core damage. The primary contributor is a mechanical failure of CCW and operator error in establishing charging flow.

- Sequence #6 - Fire near safeguard 480 V electrical buses 51 and 52 followed by a failure of CCW.

  This results in a loss of cooling to the RXCP seals and late core damage. Human errors in this sequence include failures to manually establish on-site power, service water and CCW.

- Sequence #7 - Oil fire in B diesel generator followed by failure of AFW and bleed and feed.

  This results in a loss of heat sink and early core damage. All B train equipment, powered by safeguards 4160 V electrical bus 6, is unavailable due to the fire. Therefore, the primary means of failure is mechanical failure of the A motor driven and turbine driven AFW pumps followed by failures of the SI, CCW or Instrument Air Systems, all of which are needed for bleed and feed. Another means of failure is a failure of DC bus BRA-104, which is needed to start both the A motor driven and turbine driven AFW pumps, and the A SI pump. Human error, the failure to establish bleed and feed, is a minor contributor to this sequence.

- Sequence #8 - Oil fire in B diesel generator followed by failure of AFW, success of bleed and feed, and failure of high pressure recirculation.

  This results in a loss of heat sink and late core damage. All B train equipment, powered by bus 6, is unavailable due to the fire. Therefore, the primary means of failure is mechanical failure of the A motor driven and turbine driven AFW pumps followed by failures of valves associated with switchover to containment sump recirculation. Human error, in this case failure to stop the A RHR pump to limit its time in miniflow, is a minor contributor to this sequence.

- Sequence #9 - Oil fire in A AFW pump followed by failure of AFW, success of bleed and feed, and failure of high pressure recirculation.

  This results in a loss of heat sink and late core damage. Since the fire destroys both the A AFW pump and control cabling for the turbine driven pump, the primary means of failure is a mechanical failure of the B AFW pump followed by failure of valves associated with switchover to containment sump recirculation. Human error is not important in this sequence.

- Sequence #10 - Oil fire in B AFW pump followed by failure of charging and CCW.

  This results in a loss of cooling to the RXCP seals and late core damage. Both CCW and charging must be established from the DSP. This sequence is dominated by the combination of failure to establish CCW and failure to establish charging. In this case, moderate dependency is applied, as stated in Section 4.8.7.4.

- Sequence #11 - Fire in the relay room followed by failure of AFW.

  This results in a loss of heat sink and early core damage The dominant contributor to this sequence is human error. The A diesel generator must be started locally, and service water and either charging or CCW must be started from the DSP.

- Sequence #12 - Fire near buses 51 and 52 followed by failure of AFW and bleed and feed.

  This results in a loss of heat sink and early core damage. Since AFW, SI, and instrument air are all started locally, human error is the dominant contributor to this sequence. Of these human errors, the largest contributor is failure to establish AFW and instrument air. These are not modeled as dependent actions because AFW is started from the control room and the instrument air compressor is simultaneously started locally.

- Sequence #13 - Fire near MCC 62A in B diesel generator room followed by failure of AFW, main feedwater, and bleed and feed.

  This sequence consists mostly of mechanical failures of AFW followed by the operator failing to stop the RXCPs, in order to prevent the additional heat input into the Reactor Coolant System.

4.7 Fire-Induced Containment Failures

4.7.1 Containment Failure Frequency Quantification Procedure

As in the IPE, the core damage sequences are coupled with the containment safeguards systems in order to determine which systems would be available after core damage. The containment safeguards systems used are containment isolation (CI), containment fan coil units, ICS and low pressure SI recirculation. Fire-induced failures of these systems are accounted for in the same way as those systems used in the core melt frequency determination. The containment event tree (CET) is identical to that used for the IPE and is therefore not reproduced in this report. The frequencies of each CET end state are presented in Table 4-6. The phenomenological studies performed for the IPE are bounding for the fire PRA as well, because no additional vulnerabilities were discovered in the fire PRA. As in the IPE, a 48 hour mission time is used.

4.7.2 Quantification Results

Table 4-7 shows the results of the Level 2 fire analysis. The fire-induced containment failure frequency is 3.64E-05 which is considerably larger than the internal events containment failure frequency (8.03E-06) but not out of line with containment failure frequency values from other PRAs in the industry. The fire-induced containment failure frequency is 31% of fire-induced CDF.

4.8 Treatment of Fire Risk Scoping Study Issues

4.8.1 Dependency Between Control Room and Remote Shutdown Panel Circuitry

The remote shutdown panel and the control room are in two separate fire areas. There is segregation between the two areas, and the cables contained in each area. Shutdown functions in dedicated and alternate fire areas are separated. Dependency between the control room and the remote shutdown panel is not credible at Kewaunee.

4.8.2 Cable Routing Verification

Cable routing was verified for the 1990 WPSC internal audit of fire documentation. Updated cable tracing information is used in this analysis.

4.8.3 Suppression Agent Induced Damage

Page 6-14 of the Kewaunee Fire Protection Program Analysis states, "In the use of Halon 1301 and CO_2, toxicity, thermal shock, concentration and soak times and overpressurization have all been considered in the initial design." Insufficient data exists to quantitatively analyze potential Halon and CO_2 damage. During the walkdown, obvious potential hazards for CO_2 and Halon damage were identified and have been addressed in this report.

Water damage from fire suppression activities could disable components. Damage due to fire effects or suppression activities is assumed to fail affected components. Many safety-related areas have no automatic suppression systems, so inadvertent automatic system actuation is impossible. The fire brigade is trained to avoid spraying energized electrical components.

Kewaunee Operations Experience Assessment 83-78 (Reference 18) addresses suppression agent induced damage at Kewaunee and concludes that it is not a significant issue.

4.8.4 Fire Brigade Effectiveness

Reference 7 documents fire brigade drills from 1988 to 1991. Fire brigade response times were determined for anywhere in the auxiliary building, and anywhere in the turbine building (including the screenhouse). Response times are 6.0 minutes for anywhere in the turbine building, and 6.4 minutes for anywhere in the auxiliary building.

Reference 8 documents the responses of Kewaunee to a fire brigade survey described in the Fire Risk Scoping Study (Reference 19). According to the survey, Kewaunee has a comprehensive fire brigade.

The fire brigade is trained and drilled, as described in Fire Brigade Training Procedures.

4.8.5 Seismic-Fire Interactions

4.8.5.1 Seismically induced fires

Pumps that contain lube oil could be vulnerable to lube oil spillage, and possibly initiate a fire. At Kewaunee, all safety-related pumps are also seismic class I, which means they can survive a design basis earthquake. Non-safety related pumps such as the main feedwater pumps are not seismically anchored. However, non-safety related pumps are not co-located with safety related components. The turbine lube oil storage tank was evaluated by a Seismic Qualification Utility Group (SQUG) Seismic Review Team during the unresolved safety issue (USI) A-46 walkdowns conducted in March and April, 1993. Stevenson and Associates performed an anchorage evaluation of the tank, and determined that the tank was seismically adequate. Diesel generator day tanks and gas bottles are also seismically anchored. It is concluded that seismically induced fires that threaten safety related components are not credible at Kewaunee.

4.8.5.2 Seismic degradation of fire suppression capabilities

Fire suppression capabilities in the safeguards areas at Kewaunee include fire water from class I service water and CO_2. The two fire pumps at Kewaunee are qualified according to the USI A-46 program. Table B.2-1 of the Kewaunee Updated Safety Analysis Report (USAR) states that the fire protection systems serving class I equipment are classified as class I. The water sprinkler piping and the sprinkler heads in non-safeguards areas are not seismically anchored. The CO_2 tank is seismically anchored. Seismically induced actuation of the diesel generator Cardox system was examined (Reference 20), and concluded not to present a hazard. A seismic event could damage fire water capability, but damage to the CO_2 system is not credible.

Mercury switches are a special concern, because they can spuriously operate during even a minor seismic event. The following switches were determined from the seismic walkdown to be mercury switches:

A and B fire pumps, jockey pump switches

The malfunction of these switches could cause the pumps to not operate on demand, or to operate spuriously. If the pumps were to operate spuriously, they would pressurize the line. This could result in damage to the pumps by dead heading, but it could not result in spurious actuation of the sprinklers. Therefore, the worst case is the unavailability of the pumps.

Cardox pressure switch

This switch operates the compressor that controls CO_2 pressure in the Cardox system. A malfunction of this switch could cause the compressor to not operate on demand, or to operate spuriously. If the compressor failed to operate on demand, it would cause the relief valve to open, venting CO_2 out the roof of the turbine building. If it were to operate spuriously,the pressure could get too low, also causing the system to be inoperable. Neither failure of the switch could cause the Cardox system to spuriously operate.

A failure of any or all of the above switches could result in the unavailability of the fire protection system when needed. The fire and seismic walkdowns assessed the possibility of a seismically induced fire, and none were found. The probability of an independent fire concurrent with a seismic event is negligible.

4.8.5.3 Inadvertent actuation of fire suppression systems

A seismic event could actuate charged fire water sprinkler systems. Because charged suppression systems are not located in safety-critical areas at Kewaunee, the impact of inadvertent actuations is minimized.

4.8.6 Use of Plant Specific Data

After reviewing fire Licensee Event Reports and Kewaunee reports, it was concluded that insufficient data exists to update generic fire data with plant specific data.

4.8.7 Consideration of Uncertainties

Sources of uncertainty in a fire PRA include the following:

- Fire initiation frequencies
- Judgement of fire propagation
- Fire suppression probabilities
- Human error calculations

- COMPBRN IIIe code calculations
- Barrier failure
- Random failure probabilities

4.8.7.1 Uncertainty in generation of fire initiation frequencies

The most current database is employed in deriving the fire initiation frequencies. In addition, the method used to derive these frequencies is comprehensive, and accounts for both fixed and transient combustibles.

4.8.7.2 Judgement of fire propagation

Judgement of whether a fire could propagate to an adjacent area or zone is based on Appendix R documentation (barrier construction, combustible loading, and door fire ratings), and general arrangement drawings. These were verified during walkdown inspection. In addition, during the walkdown, control of transient combustibles that could lead to fire propagation was verified.

4.8.7.3 Fire suppression probabilities

Fire suppression system unavailabilities were obtained from the FIVE Methodology. Effectiveness of the fire suppression was determined by the following:

- Time to damage of components/cables is determined using the COMPBRN IIIe code.
- Time to actuate detection devices is determined using the COMPBRN IIIe code.
- If t_{damage} is less than $t_{detector\ actuation}$, then credit for fire suppression is not granted. If t_{damage} exceeds $t_{detector\ actuation}$, then the unavailabilities of detector and suppression systems are factored into development of the initiating event frequency.

4.8.7.4 Human error calculations

The human actions required during a fire-induced transient (with the exception of fighting the fire which is described in section 4.8.4) are similar to those actions described in the IPE. There are, however, three differences that need to be addressed.

a) Additional stress is put on the operators due to the fire and the resulting loss of equipment and instrumentation.

b) Local actions may be hampered by the presence of the fire.

c) A severe fire results in entry to procedures E-0-06, Fire in Alternate Zone, or E-0-07, Fire in Dedicated Zone. These procedures have many additional steps, such as removing fuses and isolating offsite power. In addition, E-0-06 requires evacuation of the control room and activation of the dedicated shutdown panel (DSP).

For events in which E-0-06 and E-0-07 are not necessary, the additional stress is accounted for by assuming that moderate stress, as defined by the IPE, becomes high stress in a fire situation. This results in a multiplier of 5 (high stress) placed on both the action and the recovery, replacing a multiplier of 2 (medium stress). Since the action is multiplied by the recovery, the net effect is to place a multiplier of $(5/2)^2$ or 6.25 on each medium stress action. For simplicity and conservatism, a factor of 10 is applied to all actions with two exceptions. The exceptions are 1) those actions taken before the event that result in returning a piece of equipment to service after test and maintenance, and 2) those actions taken late in the accident, such as switchover to containment sump recirculation, after which the initial stress due to the fire has died down.

Only one local action is modeled for the fire analysis. This action is opening valves SW-1300A and SW-1300B to provide additional service water flow to cool the component cooling heat exchangers in the event that these valves do not open automatically. This action does not take place in any of the fire zones that were not screened out, nor are any of those zones traversed by the operator on the way to these valves. This action is not necessary until after the switchover to containment sump recirculation (at least 3.5 hours), so the auxiliary operator has ample time to perform it, even if he was involved in the fire-fighting effort. Therefore, no additional penalty, other than the factor of 10 discussed above, is taken for this action.

In a fire that causes the inability to monitor or control major plant parameters from the control room, operators are instructed to enter E-0-06. This procedure instructs the operators to evacuate the control room, isolate alternate (train B), and nonsafeguards equipment including offsite power, manually restore power from the A diesel generator and shut the plant down from the DSP. Bleed and feed, which is used in the IPE in case AFW and main feedwater fails, is not available from the DSP and is therefore not considered in this analysis. Since the operator actions are very different from any scenario in the IPE, they have been evaluated separately using the same methodology employed in the IPE. Following are all operator actions in E-0-06 that appear in the fire PRA and their human error probabilities (HEPs):

ACTION	HEP
Isolate nonessential equipment and restore power to electrical bus 5	2.69E-01
Establish service water from the DSP	8.06E-02
Establish AFW from the DSP	5.19E-02
Establish charging from the DSP	1.03E-01
Establish CCW to the RXCP thermal barriers	3.08E-02

One operator error dependency exists - charging is only necessary if CCW fails. Therefore, according to the dependency rules used in the IPE, moderate dependency is applied. This results in a human error probability of 2.31E-01 for establishing charging when CCW has failed due to operator error.

These values are used for scenarios FI5 and FI7.

In a fire that causes the inability to monitor or control dedicated equipment and instrumentation from the control room, operators are instructed to enter E-0-07. This procedure is similar to E-0-06, except that the operators continue to occupy the control room. Charging is isolated in

E-0-07 since all three charging pumps are in a dedicated zone so their ability to provide cooling to the RXCP seals is not considered. Since the operator actions are very different from any scenario in the IPE, they have been evaluated separately using the same methodology employed in the IPE. Following are all operator actions that appear in E-0-07 and their HEPs:

ACTION	HEP
Isolate nonessential equipment and restore power to electrical bus 6	7.87E-03
Establish service water manually	1.72E-03
Establish AFW manually	3.18E-03
Establish CCW to the RXCP thermal barriers	1.59E-03
Establish safety injection	5.57E-03
Establish instrument air	3.98E-01

One operator error dependency exists - safety injection is only necessary if AFW fails. Therefore, according to the dependency rules used in the IPE, moderate dependency is applied. This results in a human error probability of 1.50E-01 for establishing safety injection when AFW has failed due to operator error.

These values are used for scenarios FI6 and FI8. Since the control room and control room alarms and indications are available, operators can perform other tasks while in E-0-07. The HEP values for these tasks are values used for the IPE with a factor of 10 applied due to additional stress as explained above.

4.8.7.5 COMPBRN calculations

The uncertainty option in COMPBRN was run to gauge the uncertainty in the COMPBRN calculations and these results are discussed along with the discussion of the base case COMPBRN runs.

4.8.7.6 Barrier failure

Ratings from the NFPA Handbook (Reference 21) for fire resistance of barriers were used, and compared with the combustible loadings of fixed and transient combustibles. Combustible loadings were verified during the walkdown by observing combustibles in each zone.

4.8.7.7 Random failure probabilities

In the IPE, sensitivity studies were performed to determine the effects of varying certain parameters on the core damage frequencies. Sources of uncertainty for random failure probabilities were identified in the IPE. There are no failure modes unique to the fire analysis that generate new random failure uncertainties. Random failure uncertainties are dominated by other sources of uncertainty in fire analysis.

4.9 USI A-45 and Other Safety Issues

Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements, as it pertains to the fire analysis, is discussed in detail in Section 3.2 of this report.

The Kewaunee response to NRC Bulletin 92-01, regarding Thermo-Lag 330-1 Fire Barriers, states that there is no Thermo-Lag material at Kewaunee. The response is documented in Reference 22.

The concerns raised in GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," (Reference 23), were investigated qualitatively. This includes examination of inadvertent and advertent actuations of fire protection systems, and seismic-fire interactions. It is concluded that since automatic fire protection systems are minimally present in areas where safety-critical equipment resides, there is minimal impact if fire protection systems were to actuate. Likewise, seismic-fire interactions are minimized due to the absence of automatic fire protection systems in critical areas, and adequate anchoring of ignition sources (i.e., fuel tanks) in safeguards areas. Additional discussion of seismic-fire interactions is present in sections 4.6.5.1 through 4.6.5.3 of this report.

4.10 Areas of Conservatism

The following areas of conservatism are inherent in this analysis:

a. Total damage is assumed for any component or cable found to be vulnerable to fire damage using COMPBRN. The conservatisms present in COMPBRN are described in the COMPBRN manual. Exceptions to assuming total damage are found in the modeling of a fire in the AFW pump rooms. COMPBRN indicates that the probability of damage, given a fire is only 0.52 in the A pump room and 0.05 in the B pump room. This probability is factored in calculation of the initiating event frequency for this scenario.

b. Although strict transient combustible controls exist at Kewaunee, all trash cans are equipped with lids held open by fusible links, and administrative controls are adhered to, it is postulated in this analysis that a fire may be started by violation of administrative controls and that transient combustibles would fuel the fire.

c. Although most fires historically have either been detected and extinguished by personnel prior to damage or have been self-extinguishing, credit is not taken for this in this analysis.

d. In the evaluation of the cable trays to determine what equipment is affected by damage to a given cable tray, it is assumed in many cases that if a cable is associated with a given piece of equipment, then damage to that cable renders the piece of equipment inoperable.

4.11 Summary of Key Findings

AFW pump A and B oil fires give rise to high core damage frequencies, 5.27E-5/yr for AFW pump A, and 2.97E-5/yr for AFW pump B. The sum of these two core damage frequencies accounts for 84% of the total fire induced core damage frequency. All other scenarios yield lower core damage frequencies Three other scenarios produce reportable core damage frequencies: scenarios, FI4 (diesel generator B oil fire - 7.44E-06/year), FI2 (MCC 62A Scenario - 4.65E-06/year), and FI8 (fire near MCCs 51 and 52 - 3.33E-06/year). Neither of the AFW pump rooms has automatic suppression in the room, so credit is not taken for automatic suppression. Conservative assumptions are made in calculating the initiating event frequency, and when determining the contents of the cable trays passing through these rooms.

4.12 References

1. GL 88 20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f), June 1991.

2. Appendix R Design Description, Wisconsin Public Service Corporation, 1990.

3. "Fire Protection Program Analysis, Kewaunee Nuclear Power Plant," Volumes 1 & 2, Wisconsin Public Service Corporation, Revision 3, February 1991.

4. Licensee Event Reports for fire events at the Kewaunee Nuclear Power Plant.

5. Procedures for shutdown given a fire in Alternate and Dedicated Zones:

   E-0-06, "Fire in Alternate Fire Zone", November 6, 1990.

   E-0-07, "Fire in Dedicated Fire Zone", November 6, 1990.

6. FPP 08-8, "Control of Transient Combustibles", WPSC KNPP Fire Plan Procedure, November 19, 1991.

7. Fire brigade response times compiled from fire drills during 1988 to 1991.

8. Fire Brigade Survey for Kewaunee Nuclear Power Plant, 1992.

9. Procedures for Fire Brigade Training:
   T-FBT-L.P. 08-2
   T-FBT-L.P. 08-3
   T-FBT-L.P. 08-4
   T-FBT-L.P. 08-5
   T-FBT-L.P. 08-6
   T-FBT-L.P. 08-7
   T-FBT-L.P. 08-8
   T-FBT-L.P. 08-9
   T-FBT-L.P. 08-10
   T-FBT-L.P. 08-11
   T-FBT-L.P. 08-12
   T-FBT-L.P. 08-13
   T-FBT-L.P. 08-14
   T-FBT-L.P. 08-15
   T-FBT-L.P. 08-16
   T-FBT-L.P. 08-17
   T-FBT-L.P. 08-18
   T-FBT-L.P. 08-19
   T-FBT-L.P. 08-20
   T-FBT-L.P. 08-21

10. Fire Area Strategies, Kewaunee Nuclear Power Plant, 1990.

11. "Fire Induced Vulnerability Evaluation Methodology (FIVE) Plant Screening Guide," Prepared by Professional Loss Control for EPRI under Contract No. RP 3000-41, December 1991.

12. WALT and WLINK codes, WLINK Code System, Version 3.11, July 1992.

13. "COMPBRN IIIe: An Interactive Computer Code for Fire Risk Analysis," University of California at Los Angeles, EPRI NP-7282, May 1991.

14. WALT files used for screening, November 1992.

15. Kewaunee Nuclear Power Plant Individual Plant Examination, December 1992.

16. EPRI/NUMARC fire workshop, June 1992, Baltimore, Maryland.

17. Fire Protection Handbook, 17th edition, NFPA, 1991.

18. Operational Experience Assessment #83-78, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment (IE IN 83-41)," WPS, June 30, 1983.

19. NUREG/CR-5088, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues, January 1989.

20. Letter from Walter Djordjevic of Stevenson Associates to Greg Ridder of Wisconsin Public Service Corporation, April 15, 1992.

21. Fire Protection Handbook, Seventeenth Edition, NFPA, 1991.

22. Letter from C.R. Steinhardt of WPSC to U.S. Nuclear Regulatory Commission, re: NRC Bulletin 92-01, July 27, 1992.

23. NUREG/CR-5580, Volume 1, "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," Lambright, J. A. et. al., December 1992.

Table 4-1: Kewaunee Fire Initiation Frequencies

*** KEWAUNEE FIRE INITIATION FREQUENCIES ***

a = number of ignition sources in compartment
b = total number of ignition sources in selected plant location
c = total number of ignition sources compartments in plant
F(f) = fire ignition frequency by location and source
ignition source frequency F(if) = WF(L) x WF(LS) x F(f)
compartment fire frequency F(I) = sum(F(if))

DEDICATED ZONES

ZONE

AX 23B	SELECTED PLANT LOCATION	AUXILIARY BLDG				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	24	72	0.3333333333	0.019	0.006333
	PUMPS	6	16	0.375	0.019	0.007125
	PLANT-WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	3	25	0.12	0.0013	0.000156
	TRANSFORMERS	2	21	0.0952380952	0.0079	0.000752
	VENTILATION SYSTEMS	2	21	0.0952380952	0.0095	0.000904
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0.015271

SC 70A	SELECTED PLANT LOCATION	INTAKE STRUCTURE				
	LOCATION WEIGHTING FACTOR - WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	4	6	0.6666666667	0.0024	0.0016
	FIRE PUMPS	1	2	0.5	0.004	0.002
	SW PUMPS	2	4	0.5	0.0032	0.0016
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	FIRE PROTECTION PANELS	1	2	0.5	0.0024	0.0012
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0.006852

TU 90	SELECTED PLANT LOCATION	DIESEL ROOM A				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	DIESEL GENERATOR A	1	3	0.3333333333	0.026	0.008666
	4160V BUS 5, MCC 52A	14	31	0.4516129032	0.0024	0.001083
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	1	25	0.04	0.0013	0.000052
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0.010253

TU 95A	SELECTED PLANT LOCATION	SWGR DEDICATED SHUTDOWN PANEL ROOM				
	LOCATION WEIGHTING FACTOR WF(L)	0 25				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	10	71	0 1408450704	[illegible]	0 000528
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0 0476190476	0 0095	0.000452
	AIR COMPRESSOR	1	6	0.1666666667	0 0047	0.000783
	COMPARTMENT FREQUENCY - F(l) = sum[F(if)] =					0.001764

TU 95C	SELECTED PLANT LOCATION	AFW PUMP A ROOM				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	PUMP 1A	1	8	0 125	0.0063	0.000787
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0.0476190476	0 0095	0 000452
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.001239

TU 97	SELECTED PLANT LOCATION	BATTERY ROOM (ROOM 129)				
	LOCATION WEIGHTING FACTOR - WF(L)	0 3333333333				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	BATTERIES	1	3	0 3333333333	0 0032	0.000355
	ELECTRICAL CABINETS	14	31	0 4516129032	0 019	0 002860
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSFORMERS	1	26	0.0384615385	0 0079	0 000303
	VENTILATION SYSTEMS	2	21	0.0952380952	0.0095	0 000904
	BATTERY CHARGERS	1	4	0 25	0.004	0 001
	COMPARTMENT FREQUENCY - F(l) = sum[F(if)] =					0.005424

ALTERNATE ZONES

ZONE

AX 2	SELECTED PLANT LOCATION	SWGR (ROOM 16B)				
	LOCATION WEIGHTING FACTOR - WF(L)	0 25				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	
	ELECTRICAL CABINETS	8	71	0 1126760563	0 015	0 000422
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0 0476190476	0 0095	0 000452
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.000875

AX 22	SELECTED PLANT LOCATION	CONDENSATE STORAGE TANK ROOM				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	17	72	0 2361111111	0.019	0 004486
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	3	25	0 12	0 0013	0.000156
	ELEVATOR MOTOR	1	3	0 3333333333	0 0063	0.0021
	GRINDING/CUTTING	1	2	0 5	0.031	0.0155
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0 022242

AX 23A	SELECTED PLANT LOCATION	MCC 62J ROOM, EL 642'				
	LOCATION WEIGHTING FACTOR - WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	1	72	0 0138888889	0 019	0 000263
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
		--	---	---	---	--
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0 000263

AX 23A	SELECTED PLANT LOCATION	AUXILIARY BLDG (GAS BOTTLE STORAGE AREA, EL 657')				
	LOCATION WEIGHTING FACTOR - WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	2	72	0.0277777778	0 019	0 000527
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)F(f)	F(if)
	VENTILATION SYSTEMS	2	21	0.0952380952	0 0095	0 000904
	BOTTLED GAS	1	2	0 5	0 0032	0 0016
	COMPARTMENT FIRE FREQUENCY - F(I) = sum[F(if)] =					0.003032

AX 23C	SELECTED PLANT LOCATION	RHR PUMP B PIT				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	RHR PUMP	1	16	0 0625	0.019	0 001187
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0 0476190476	0 0095	0 000452
	COMPARTMENT FREQUENCY - F(I) = sum[F(if)] =					0 001639

AX 23D	SELECTED PLANT LOCATION	CCW PUMP B ROOM				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	CCW PUMP	1	16	0 0625	0 019	0 001187
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0 0476190476	0 0095	0 000452
	COMPARTMENT FIRE FREQUENCY F(I) sum [F(if)] =					0 001639

AX 30	SELECTED PLANT LOCATION	CABLE SPREADING ROOM (RELAY ROOM)				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	69	69	1	0.0032	0.0032
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
		---	---	---	---	---
	COMPARTMENT FREQUENCY · F(l) = sum[F(if)] =					0.0032

AX 32	SELECTED PLANT LOCATION	AUXILIARY BLDG (SERVICE ROOMS)				
	LOCATION WEIGHTING FACTOR · WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	DRYER (FOR HP CLOTHING)	3	3	1	0.0087	0.0087
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	3	25	0.12	0.0013	0.000156
	COMPARTMENT FIRE FREQUENCY · F(l) = sum[F(if)] =					0.008856

AX 35	SELECTED PLANT LOCATION	AUXILIARY BLDG (SERVICE ROOMS)				
	LOCATION WEIGHTING FACTOR · WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	15	15	1	0.0095	0.0095
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	1	25	0.04	0.0013	0.000052
	COMPARTMENT FIRE FREQUENCY F(l) = sum[F(if)] =					0.00955

AX 35	SELECTED PLANT LOCATION	AUXILIARY BLDG (SERVICE ROOMS)				
	LOCATION WEIGHTING FACTOR · WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	15	15	1	0.0095	0.0095
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	1	25	0.04	0.0013	0.000052
	VENTILATION SYSTEMS	2	21	0.0952380952	0.0095	0.000904
	COMPARTMENT FIRE FREQUENCY · F(l) = sum[F(if)] =					0.010456

AX 37	SELECTED PLANT LOCATION	SWGR ROOM				
	LOCATION WEIGHTING FACTOR WF(L)	0.25				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	SWGR, C D	8	75	0.1066666667	0.015	0.00040
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	RPS MG SETS	2	2	1	0.0055	0.0055
	COMPARTMENT FIRE FREQUENCY · F(l) = sum[F(if)] =					0.0059

SC 70B	SELECTED PLANT LOCATION	INTAKE STRUCTURE				
	LOCATION WEIGHTING FACTOR - WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	2	6	0.3333333333	0.0024	0.0008
	FIRE PUMPS	1	2	0.5	0.004	0.002
	SERVICE WATER PUMPS	2	4	0.5	0.0032	0.0016
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	FIRE PROTECTION CABINETS	1	2	0.5	0.0024	0.0012
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.006052

TU 22	SELECTED PLANT LOCATION	TURBINE BLDG				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	26	26	1	0.013	0.013
	T/G EXCITOR	1	1	1	0.004	0.004
	T/G HYDROGEN	4	4	1	0.0055	0.0055
	T/G OIL	5	5	1	0.013	0.013
	MAIN FEED PUMPS	2	2	1	0.004	0.004
	OTHER PUMPS	4	4	1	0.0063	0.0063
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	10	25	0.4	0.0013	0.00052
	TRANSFORMERS	5	21	0.2380952381	0.0079	0.001880
	VENTILATION SYSTEMS	2	21	0.0952380952	0.0095	0.000904
	AIR COMPRESSORS	3	6	0.5	0.0047	0.00235
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.051455

TU 92	SELECTED PLANT LOCATION	DIESEL ROOM B				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	BUS 6, MCC 62A	14	31	0.4516129032	0.0024	0.001084
	DIESEL GENERATOR	1	3	0.3333333333	0.026	0.008666
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSIENTS	1	25	0.04	0.0013	0.000052
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.010254

TU 94	SELECTED PLANT LOCATION	CO_2 STORAGE TANK ROOM - ROOM 4B				
	LOCATION WEIGHTING FACTOR WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
			--	--	--	--
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	AIR COMPRESSOR A	1	6	0.1666666667	0.0047	0.000783
	COMPARTMENT FIRE FREQUENCY F(l) = sum[F(if)] =					0.001235

TU 95B	SELECTED PLANT LOCATION	AFW PUMP B ROOM				
	LOCATION WEIGHTING FACTOR - WF(L)	1				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	AFW PUMP B	1	8	0.125	0.0063	0.00078
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	VENTILATION SYSTEMS	1	21	0.0476190476	0.0095	0.000452
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.001240

TU 95B	SELECTED PLANT LOCATION	SWGR ROOM (BUSES 61 & 62)				
	LOCATION WEIGHTING FACTOR - WF(L)	0.25				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	ELECTRICAL CABINETS	9	71	0.1267605634	0.015	0.000475
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	AIR COMPRESSOR B	1	6	0.1666666667	0.0047	0.000783
	INSTR AIR DRYERS	2	2	1	0.0087	0.0087
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.009958

TU 98	SELECTED PLANT LOCATION	BATTERY ROOM (RM 130)				
	LOCATION WEIGHTING FACTOR - WF(L)	0.3333333333				
	COMPONENT IGNITION SOURCES	a	b	WF(LS) = a/b	F(f)	F(if)
	BATTERIES	1	3	0.3333333333	0.0032	0.000355
	ELECTRICAL CABINETS	14	31	0.4516129032	0.019	0.002860
	PLANT WIDE IGNITION SOURCES	a	c	WF(LS) = a/c	F(f)	F(if)
	TRANSFORMERS	1	26	0.0384615385	0.0079	0.000303
	VENTILATION SYSTEMS	2	21	0.0952380952	0.0095	0.000904
	BATTERY CHARGER	1	4	0.25	0.004	0.001
	COMPARTMENT FIRE FREQUENCY - F(l) = sum[F(if)] =					0.005424

Table 4-2: Screening Results, Dedicated and Alternate Zones

Alternate Zone	Fire-induced Core Damage Frequency (/yr)	Dedicated Zone	Fire-induced Core Damage Frequency (/yr)
AX-21	7.001E-5	AX-23B	1.094E-3
AX-22	2.719E-6	SC-70A	7.650E-6
AX-23A	1.038E-3	TU-90	5.501E-4
AX-23C	1.097E-6	TU-95A	2.187E-4
AX-23D	3.326E-6	TU-95C	5.725E-5
AX-30	3.920E-7	TU-97	2.289E-4
AX-32	8.079E-4		
AX-35	2.450E-6		
AX-37	7.129E-7		
SC-70B	1.584E-6		
TU-22	6.716E-6		
TU-92	1.582E-3		
TU-94	1.470E-4		
TU-95B	1.193E-2		
TU-98	5.092E-4		

NOTE: These core damage frequencies were calculated for screening purposes only, and do not realistically reflect the scenario.

Table 4-3: AREA OF INFLUENCE & ROOM AREA FOR EACH DAMAGE SCENARIO
(Sheet 1 of 3)

Fire Zone	Damage Scenario	Damage Time	Area of Influence	Room Area	(Infl.Area)/(Room Area)
AX-21	Buses 1 and 2 Room	1 min	$(0.9\ m)*(13.75\ m) = 12.375\ m^2$	$114.07\ m^2$	0.108
AX-23A	MCC 35E/45E	no damage	---	---	---
	MCC 62J	2 min.	entire contents of room damaged	---	1
	Bottled Gas, El. 657'	1 min.	$(5.5\ m)*(0.5\ m) = 2.75\ m^2$	$(7.01\ m)*(13.1\ m) = 91.9\ m^2$	2.99E-02
	SI Pump	no damage	---	---	---
AX-23B	Flammable Liq.Cab	no damage	---	---	---
	Bottled Gas, El. 586'	no damage	---	---	---
AX-23D	CCW Pump B Room	14 min.	entire contents of room destroyed	---	1
AX-30	Fire Near Vertical Trays	2 min.	$(1/2)(\pi)(1.5\ m)^2 = 3.53\ m^2$	$(16.2\ m) * (16.8\ m) = 270.8\ m^2$	1.31E-02
	Fire Near Cabinets	no damage	---	---	---
AX-32	MCC 62B	no damage	---	---	---
	MCC 62B Ext	no damage	---	---	---
	Cable Spreading Area	not modeled	---	---	---
AX-35	Control Room	not modeled	---	---	---
SC-70A	Screenhouse	no damage	---	---	---

Table 4-3: AREA OF INFLUENCE & ROOM AREA FOR EACH DAMAGE SCENARIO
(Sheet 2 of 3)

Fire Zone	Damage Scenario	Damage Time	Area of Influence	Room Area	(Infl. Area)/(Room Area)
TU-90	Bus 5	no damage	---	---	---
	MCC 52A	no damage	---	---	---
	Diesel Fire	3 min	---	---	---
TU-92	MCC 62A	3 min.	---	---	---
	Bus 6	no damage	---	---	---
	Diesel Fire	3 min.	---	---	---
TU-94	Air Comp. A Fire	no damage	---	---	---
	Beneath Cable Trays	no damage	---	---	---
TU-95A	Switchgear Room 5B	2 min.	$(0.2\ m)(13\ m) = 2.6\ m^2$	$98.65\ m^2$	2.64E-02

Table 4-3: AREA OF INFLUENCE & ROOM AREA FOR EACH DAMAGE SCENARIO
(Sheet 3 of 3)

Fire Zone	Damage Scenario	Damage Time	Area of Influence	Room Area	(Infl. Area)/(Room Area)
TU-95B	Cable Trays Above Bus 61	no damage	---	---	
	Cable Trays North of Bus 62	no damage	---	---	
	Cable Trays in AFW B Room	3 min.	entire contents of room damaged	---	
	Turbine-Driven AFW Pump Room	not modeled (steam exclusion ceiling protects cables)	---	---	
TU-95C	AFW Pump A Room	1 min	entire contents of room damaged	---	1
TU-97	Battery Room A	no damage	---	---	---
TU-98	Battery Room B	no damage	---	---	---

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI1: AX-23A, MCC 62J scenario	1) Cable tray 1FT6S6 2) Cable tray 1FX4S6	a) valve BT 2B b) valve MS 100B Affected cables control components which are not modeled in PRA.	07-MV---BT2B-OO 06-MV-MS100B-OC	AF3 AF3
FI1: AX-23A, MCC 62J scenario	Components	MCC 62J	40-BS-MCC62J-SG	AF3
FI2: TU-92, DG B room, MCC 62A scenario	Cable tray 1DT4S6 Breaker 610 to MAT not modeled	a) AFW pump B b) Air compressor B c) Breaker 611 d) DG 1B e) MCC 62A f) RHR pump B - injection g) SI pump B - injection	05BPM--AFW1B-PS 01-PM-SIAC1B-PS 39-CB-1-611--FC 10-GE-DG1B---PS 40-BS-MCC62A-SG 341PM--RHR1B-PS 331PM---SI1B-PS	AF3 IAS, IAST, IASTA, IASTB BUS6 DGB IAS, IAST, IASTA, IASTB RHRB HPI, HRI
FI2: TU-92, DG B room, MCC 62A scenario	Cable tray 1DT4S6	h) SW pump B1 i) SW pump B2 j) SW strainer B1 k) TAT q) SW valve B r) AFW pump B s) Air compressor B t) Breaker 607	02-PM-SW1B1--PR 02-PM-SW1B2--PR 02-FL-1B1----PL 39-TR-TAT----SG 02-MV-SW10B--OC 05BPM -AFW1B-PS 01-PM-SIAC1B-PS 39-CB--1-607-CO	SWB, SWT SWB, SWT SWB, SWT, BUS5, BUS6 SWB, AF3 IAS, IASP, IASP., IAST, IASTA, IASTB BRB104T, BUS61, BUS62, IASTB
FI2: TU-92, DG B room, MCC 62A scenario	Components	a) MCC 62A	40-BS-MCC62A-SG	IAS, IAST, IASTA, IASTB

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI3: AX 21, Room 16B, Fire between buses 1 and 2	Cable tray 1AT9N	a) MFW pump A b) MFW pump B	05APM--FWP1A-PR 05APM--FWP1B-PR	OM2 OM2
FI4: TU-92, DG B diesel fire	Cable tray 1DT8N Breaker 301 not modeled	a) RAT	39-TR-RAT----SG	BUS1, BUS2, BUS3, BUS4, BUS5 BUS6
FI4: TU-92, DG B diesel fire	Components	a) DG B	10-GF-DG1B---PS	DGB
FI5: AX-30, Relay room Fire near vertical trays, cable trays 1AT12S6, 1AT13S6 and 1RT62S6 damaged	Damage disables all potential source breakers for bus 6. Components dependent on bus 6 (directly or indirectly) are also disabled (starred components). Valve CC400B not modeled AFW pump 1B lube oil pump not modeled	a) TAT b) Valve BT3A c) Valve FW10A d) Valve FW7A e) Valve SI208 f) Valve CC400B	39-TR-TAT----SG 07-MV---BT3A-OO 05A-AV-FW10A-CC 05A-AV-FW7A--CC 33RMV-SI208--FC 34RMV-CC400B-CC	BUS5, BUS6 AF3 OM2 OM2 RHRA, RHRB RHRB
FI6: Fire in TU-95C, AFW pump A room (multiple cable trays affected)	Cable tray 1TT106S5 AFW pump A aux lube oil pump not modeled	a) Valve AFW10A	05BMV-AFW10A-OC	AF3

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI6: Fire in TU-95C, AFW pump A room Starred components already disabled	Cable tray 1TT15N Breakers 301, 305, 307, 308, 309, and 407 are not modeled	a) Breaker 13201 b) Breaker 13242 c) Breaker 14201 d) Breaker 303 e) CDP A* f) RAT	40-CB--13201-CO 40-CB--13242-FC 40-CB--14201-CO 39-CB--1-303-CO 03-PM--CDP1A-PR 39-TR-RAT----SG	BUS32 BUS32, BUS42 BUS42, BUS42T BUS32 OM2 BUS1, BUS2, BUS3, BUS4, BUS5, BUS6
FI5: Fire in TU-95C, AFW pump A room	Cable tray 1TT4N Breaker 511 is not modeled	a) CDP A* b) TAT	03-PM--CDP1A-PR 39-TR-TAT----SG	OM2 BUS5, BUS6
FI6: Fire in TU-95C, AFW pump A room	4) Cable tray 1TT4S5 Control room control of AFW pump A - not modeled in PRA Control room control of FCU A and B, and DGA - not modeled in PRA Control room control of SW pumps A1 and A2, train A ESF Signals - not modeled in PRA	a) Air compressor A b) Charging pump C c) MFW pump A d) MFW pump B	01-PM-SIAC1A-PR 35-PM-CHGP1C-PR 05APM--FWP1A-PR 05APM--FWP1B-PR	IAS, IAST, IASTA, IASTB CHG OM2 OM2
FI6: Fire in TU-95C, AFW pump A room	5) Cable tray 1TT8S5	a) SI pump A c) AFW pump C lube oil pump d) TAT e) Valve SW502 f) Breaker 501 g) Breaker 503 h) Breaker 505	33IPM---SI1A-PS 05BPM-ALOP1C-PS 39-TR-TAT----SG 02-MV-SW502--CC 39-CB-1-501--CO 39-CB-1-503--FC 30-CB-1-505--FO	HPI, HR1, AF3 BUS5, BUS6 AF3 BUS5 BUS5 BRA104T, BUS51, BUS52, IASTA

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI6. Fire in TU-95C AFW pump A room	Components	a) AFW pump A	05BPM--AFW1A-PS	AF3
FI7: Fire in AFW Pump B Room, TU-95B	Components	a) AFW pump B b) TBB FCU B	05BPM--AFW1B-PS 17-FN-TBB1B--PS	AF3 IAS, IAST, IASTA, IASTB
FI7: Fire in AFW Pump B Room, TU-95B	Cable tray 1TT10N Breaker 610 to MAT not modeled Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) CCW Pump B FCU* b) Breaker 611* c) Breaker 601* d) Valve SV 33778*	17-FN-CCWP1B-PS 39-CB-1-611--FC 39-CB-1-601--CO 17-SVSV33778-CC	CCW BUS6 BUS6 CCW
FI7: Fire in AFW Pump B Room, TU-95B	Cable tray 1TT15N Breakers 301, 305, 307, 308, 309, and 407 are not modeled Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4 and 6 are starred.	a) Breaker 13201* b) Breaker 13242* c) Breaker 14201* d) Breaker 303* e) CDP A* f) RAT	40-CB--13201-CO 40-CB--13242-FC 40-CB--14201-CO 39-CB--1-303-CO 03-PM--CDP1A-PR 39-TR-RAT----SG	BUS32 BUS32, BUS42 BUS42, BUS42T BUS32 OM2 BUS1, BUS2, BUS3, BUS4, BUS5, BUS6

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI7 Fire in AFW Pump B Room. TU 95B	Cable Tray 1TT20N Breakers 401 and 402 are not modeled Bus BRD 103 not modeled Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) Breaker 13501* b) Breaker 13545* c) Breaker 14501* d) Breaker 407* e) Breaker 405* f) Breaker 406* g) Distribution bus BRA-127 h) Distribution bus BRB-127 i) Condensate pump B* j) FW pump A* k) FW pump B*	40-CB--13501-CO 40-CB--13545-FC 40-CB--14501-CO 39-CB-1-407--FC 39-CB--1-405-CO 39-CB--1-406-CO 38IBS-BRA127-SG 38IBS-BRB127-SG 03-PM--CDP1B-PR 05APM--FWP1A-PR 05APM--FWP1B-PR	BUS35 BUS35, BUS45 BUS45, BUS45T BUS4 BUS45, BUS45T BUS42, BUS42T BRA127 BRB127, BRD115 OM2 OM2 OM2
FI7. Fire in AFW Pump B Room. TU-95B	Cable Tray 1TT4N Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) CDP A* b) TAT	03-PM--CDP1A-PR 39-TR-TAT----SG	OM2 BUS5, BUS6

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
F17: Fire in AFW Pump B Room, TU-95B Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	Cable tray 1TT4S5 Control room control of AFW pump A - not modeled in PRA Control room control of FCU A and B, and DGA - not modeled in PRA Control room control of SW pumps A1 and A2, train A ESF Signals, valves SI2A, SI3, SI302, SW1300A, SW903A, and SW903B - not modeled in PRA	a) Air compressor A b) Charging pump C* c) MFW pump A* d) MFW pump B*	01-PM-SIAC1A-PR 35-PM-CHGP1C-PR 05APM--FWP1A-PR 05APM--FWP1B-PR	IAS, IAST, IASTA, IASTB CHG OM2 OM2
F17: Fire in AFW Pump B Room, TU-95B	Cable tray 1TT8S5 Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) SI pump B* b) AFW pump C lube oil pump c) TAT d) Valve SW502	33IPM---SI1B-PS 33RPM---SI1B-PS 05BPM-ALOP1C-PS 39-TR-TAT----SG 02-MV-SW502--CC	HPI, HR1 HR1 AF3 BUS5, BUS6 AF3

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI7: Fire in AFW Pump B Room, TU-95B	Cable tray 1TT4S6 AFW Pump B aux lube oil pump not modeled Note that a fire in this area disables buses 1, 2, 3, 4 and 6. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) AFW pump B* b) Air compressor B* c) DG B* d) FW pump A* e) FW pump B* f) Valve AFW 10B g) Valve SW10B* j) Valve SW601B*	05BPM--AFW1B-PS 01-PM-SIAC1B-PS 10-GE-DG1B---PS 05APM--FWP1A-PR 05APM--FWP1B-PR 05BMV-AFW10B-OC 02-MV-SW10B--OC 02-MV-SW601B-CC	AF3 IAS, IAST, IASTA, IASTB DGB OM2 OM2 AF3 SWB AF3
FI7: Fire in AFW Pump B Room, TU-95B	Cable tray 1TT8S6 AFW Pump B aux lube oil pump not modeled Valve SW903C not modeled Damage disables all potential source breakers for bus 6. Note that a fire in this area also disables buses 1, 2, 3, and 4. All components which are supplied by buses 1, 2, 3, 4, and 6 are starred.	a) AFW pump B* b) Breaker 16201* c) Breaker 601* d) Breaker 611* e) Breaker 607* f) CCW pump B* g) DG B* h) RHR pump B - injection* i) RHR pump B - recirc.* j) Pump pit B FCU* k) SI pump B* l) SW pump B1* m) SW pump B2 * n) TAT o) Valve BT2B* p) Valve BT3A q) Valve SW10B*	05BPM--AFW1B-PS 40-CB--16201-CO 39-CB-1-601--CO 39-CB-1-611--FC 39-CB--1-607-CO 31-PM-CCW1B--PS 10-GE-DG1B---PS 34IPM--RHR1B-PS 34RPM--RHR1B-PS 34RFNPMPPITB-PS 33IPM---SI1B-PS 33RPM---SI1B-PS 02-PM-SW1B1--PR 02-PM-SW1B2--PR 39-TR-TAT----SG 07-MV---BT2B-OO 07-MV---BT3A-OO 02-MV-SW10B--OC	AF3 BRB104T, BUS62, IASTB BUS6 BUS6 BRB104T, BUS61, BUS62, IASTB, CCW DGB LPI, RHRB LR1, RHRB RHRB HPI, HR1 HR1 SWB,SWT SWB, SWT BUS5, BUS6 AF3 AF3 SWB

Table 4-4: FAULT TREE IDENTIFIERS OF DAMAGED CABLES/COMPONENTS

Scenario, Fire Zone	Damaged Cable Tray or Component	Contents of Cable Tray	Corresponding Fault Tree Identifier	Fault Trees in which Identifier Appears
FI8: Fire near buses 51 and 52 in TU-95A	Cable tray 1TT5S5	a) AFW Pump A*	05BPM--AFW1A-PS	AF3
		b) IA Compressor C*	01-PM-SIAC1C-PS	IAS, IAST, IASTA, IASTB
	Note that damage to this cable tray disables all potential source breakers for bus 5. All affected components are starred.	c) Breaker 15101*	40-CB--15101-CO	BUS51
		d) Breaker 15201*	40-CB--15201-CO	BRA104T, BUS52, IASTA
		e) Breaker 15203*	40-CB--15203-CO	CHG
		f) Breaker 501*	39-CB-1-501--CO	BUS5
		g) Breaker 503*	39-CB-1-503--FC	BUS5
			39-CB-1-503--CO	BUS5
		h) Breaker 505*	39-CB-1-505--FO	BRA104T, BUS51, BUS52, IASTA,
		i) Bus 51*	40-BS-BUS51--SG	BUS51
		j) Bus 52*	40-BS-BUS52--SG	BRA104T, BUS52, IASTA
		k) Chg pump A*	35-PM-CHGP1A-PR	CHG
		l) CCW pump A*	31-PM-CCW1A--PR	CCW
		m) DG A*	10-GE-DG1A--PS	DGA
		n) TAT	39-TR-TAT----SG	BUS5, BUS6
		n) MCC 5262	40-BS-MCC5262SG	IAS, IAST, IASTA, IASTB
FI8: Fire near buses 51 and 52 in TU-95A	Cable tray 1TT5S5	o) RHR pump A*	341PM--RHR1A-PS	RHRA
			34RPM--RHR1A-PS	RHRA
	Note that damage to this cable tray disables all potential source breakers for bus 5. All affected components are starred.	p)RHR pump pit A Fan*	34RFNPMPP!TA-PS	RHRA
		q) SI pump A*	331PM---SI1A-PS	HPI, HR1
			33RPM---SI1A-[illegible]	HR1
		r) SW pump A1*	02-PM-SW1A1--PS	SWA SWT
		s) SW pump A2*	02-PM-SW1A2--PR	SWA, SWT
		t) Valve BT2A	07-MV---BT2A-OO	AF3
		u) Valve BT3B*	07-MV---BT3B-OO	AF3
		v) Valve SW10A*	02-MV-SW10A--OC	SWA

Table 4-5: FIRE SEQUENCES AND THEIR CORE DAMAGE FREQUENCIES			
Sequence	**Fire Zone**	**Scenario Description**	**Core Damage Frequency**
FI1	AX-23A	MCC 62J scenario	2.81E-9/yr
FI2	TU-92	MCC 62A scenario	4.65E-6/yr
FI3	AX-21	Fire between buses 1 and 2, beneath cable tray 1AT9N	1.41E-9/yr
FI4	TU-92	Diesel B oil fire	7.44E-6/yr
FI5	AX-30	Fire in relay room	3.21E-7/yr
FI6	TU-95C	AFW pump A oil fire	5.27E-5/yr
FI7	TU-95B	AFW pump B oil fire	2.97E-5/yr
FI8	TU-95A	Fire near buses 51 and 52	3.33E-6/yr
Total Fire-induced Core Damage Frequency			9.80E-5/yr

Table 4-6: CET END STATES AND FREQUENCIES

CET END STATE	FREQUENCY	RELEASE CATEGORY
HAAAA	9.68E-06	S
HAAFA	1.29E-06	S
HAFAA	1.18E-06	S
HAFFA	2.90E-07	S
HFAAA	1.67E-09	S
HFFAA	1.85E-05	S
HFFFA	4.57E-05	A
LAAAA	7.66E-07	S
LAAFA	4.23E-08	S
LAFAA	2.91E-07	S
LAFFA	1.70E-08	S
LFAAA	3.65E-10	S
LFFAA	1.76E-06	A
LFFFA	5.09E-08	A
HAAAF	1.49E-06	G
HAAFF	2.20E-07	G
HAFAF	1.86E-07	G
HAFFF	2.48E-08	G
HFFAF	2.82E-06	G
HFFFF	3.15E-05	G
LAAAF	4.86E-08	G
LAAFF	6.43E-09	G
LAFAF	1.73E-08	G
LAFFF	1.71E-09	G
LFFAF	1.05E-07	G
LFFFF	1.40E-08	G

Notes

1 End States are given the following identifier:
  PLSFC Where:
  P H for high pressure (>400 Psia) reactor vessel failure, L for low pressure vessel failure.
  L A if low pressure recirculation is available, F if it has failed
  S A if containment spray is available, F if it has failed
  F A if containment fan coil units are available, F if they have failed
  C A if containment is isolated, F if isolation has failed
2 Release catagories are defined in Table 4-7

Table 4-7: AIRBORNE RELEASE CATEGORIES AND FREQUENCIES

Release Category	Definition	Frequency	Conditional Probability[1,2]
S	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful)	3.21E-05	0.28
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired)	3.64E-05	0.31
A	No containment failure within 48 hr mission time, but failure could eventually occur without accident management action; noble gases and less than 0.01% volatiles released	4.75E-05	0.41

NOTES:

1 Conditional probability of release category given core damage.

2 Core damage frequency for Level 2 = 1.16E-04/yr. This is larger than the Level 1 core damage frequency because success probabilities are not considered.

Title: FI1: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
			DIESEL GENERATOR	1B	FAILURE TO START	AND RUN	2.13E-02	10-GE-DG1B---PS
10	2.80E-11	1.00	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			DIESEL GENERATOR	1A	FAILURE TO START	AND RUN	2.13E-02	10-GE-DG1A---PS
			FEEDER BREAKERS	ON 4160V BUS 6	FAIL TO OPEN		4.20E-02	39-CB-BUS6FB-FO
11	2.74E-11	.97	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SOV AFW-111A	FAILS TO OPEN			2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
12	2.74E-11	.97	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			SOV AFW-111B	FAILS TO OPEN			2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
13	2.46E-11	.87	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	[illegible]E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
14	2.28E-11	.81	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			NO SERVICE WATER	DUE TO	COMMON CAUSE	FAILURES	7.30E-04	02----SWS----CM
15	[illegible]E-11	.65	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF CONTROL	SIGNAL TO MU-3A	VALVE FAILS OPEN		3.00E-06	03-AS-HTLVLC-OP
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
16	[illegible]3E-11	[illegible]	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
17	[illegible]E-11	[illegible]	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			DIESEL GENERATOR	1A	FAILURE TO START	AND RUN	2.13E-[illegible]	10-GE-DG1A---PS

Title: F11: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	1.30E-09	46.20	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
2	6.13E-10	21.78	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
3	2.23E-10	7.92	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
4	7.35E-11	2.61	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3B TRANSFERS OPEN	1.20E-05	03-AV--MU3B--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
5	7.35E-11	2.61	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3A TRANSFERS OPEN	1.20E-05	03-AV--MU3A--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
6	6.31E-11	2.24	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
7	6.[illegible]E-11	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB--CM
8	3.32E-11	1.18	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW----CM
			OPERATOR FAILS TO ESTABLISH MAIN FEEDWATER	1.19E-03	05A--OM2-OM4-HE
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	5.00E-01	36-OB2OB5OB6OHE
9	3.[illegible]E-11	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS [illegible] FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO

Title: F11: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
18	1.22E-11	.43	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
19	7.99E-12	.28	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			AOV SW-301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
20	6.99E-12	.25	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			AOV SW-301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
21	6.[illegible]6E-12	.22	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.72E-03	05BPM-ALOP1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
22	5.70E-12	.20	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH MAIN FEEDWATER	1.19E-03	05A--OM2-OM4-HE
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	5.00E-01	36-OB2OB5OB6OHE
23	5.59E-12	.20	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			TRAVELING WATER SCREEN 1B2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1B2-PS
24	4.99E-12	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
[illegible]	[illegible]	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-F11
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: FI1: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C UNAVAILABLE DUE TO TEST OR MAINTENANCE	3.00E-03	05BPM--AFW1C-TM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
26	4.51E-12	.16	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			BREAKER FROM MCC-62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A6-FC
27	4.51E-12	.16	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			BREAKER 1-601 FAILS TO OPEN	3.00E-03	39-CB-1-601--FO
28	4.51E-12	.16	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			BREAKER 1-603 FAILS TO CLOSE	3.00E-03	39-CB-1-603--FC
29	3.94E-12	.14	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
30	3.94E-12	.14	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			BREAKER 1-501 FAILS TO OPEN	3.00E-03	39-CB-1-501--FO
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
31	3.94E-12	.14	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
32	3.55E-12	.13	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A--PS
			AOV SW-301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
33	[illegible]	.13	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24

Title: F11: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
			AOV SW-301A	FAILS TO OPEN			5.32E-03	02-AV-SW301A-CC
			DIESEL GENERATOR	1B	FAILURE TO START	AND RUN	2.13E-02	10-GE-DG1B---PS
34	3.47E-12	.12	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			COMMON CAUSE	FAILURES OF	AFW SYSTEM		2.12E-04	05B--AFW-----CM
			NO SERVICE WATER	DUE TO	COMMON CAUSE	FAILURES	6.22E-05	02----SW-----CM
35	3.36E-12	.12	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SOV AFW-111A	FAILS TO OPEN			2.00E-03	05BSVAFW111A-CC
			SOV AFW-111B	FAILS TO OPEN			2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
36	3.26E-12	.12	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS
			SOV AFW-111C	FAILS TO OPEN			2.00E-03	05BSVAFW111C-CC
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
37	3.02E-12	.11	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SOV AFW-111A	FAILS TO OPEN			2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1B	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
[illegible]	[illegible]	.10	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			CHECK VALVE	AFW-4A	FAILS TO OPEN		1.00E-04	05BCV--AFW4A-FO
			MOV BT 3B	FAILS TO CLOSE			4.77E-03	07-MV---BT3B-OO
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
39	2.48E-12	.09	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		[illegible]E-04	LOSP-24
			DIESEL GENERATOR	1A	FAILURE TO START	AND RUN	[illegible]-02	10-GE-DG1A---PS
			TRAVELING WATER	SCREEN 1B2	MECHANICAL	FAILURE	3.72E-03	02-FLT-TW1B2-PS
40	2.48E-12	.09	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			TRAVELING WATER	SCREEN 1A1	MECHANICAL	FAILURE	3.72E-03	02-FLT-TW1A1-PS
			DIESEL GENERATOR	1B	FAILURE TO START	AND RUN	2.13E-02	10-GE-DG1B---PS
41	[illegible]	.09	FIRE NEAR MCC	62J OCCURS			2.63E-04	IEV-F11
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS

Title: FI1: MCC 62J Scenario
File: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets; 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			BREAKER FROM BUS BRA-102 TO BUS BRA-104 TRANSFERS OPEN	2.40E-05	38-CBA102-04-CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
42	2.25E-12	.08	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
43	2.02E-12	.07	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
44	2.00E-12	.07	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			BREAKER FROM MCC-62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A6-FC
45	2.00E-12	.07	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			BREAKER 1-601 FAILS TO OPEN	3.00E-03	39-CB-1-601--FO
46	2.00E-12	.07	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			BREAKER 1-603 FAILS TO CLOSE	3.00E-03	39-CB-1-603--FC
47	[illegible]	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
48	[illegible]	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			BREAKER 1-501 FAILS TO OPEN	3.00E-03	39-CB-1-501--FO
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
49	[illegible]	[illegible]	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC

Title: FI1: MCC 62J Scenario
ile: FIRE1.WLK (File created by linking FIRE1.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.8140E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
50	1.64E-12	.06	FIRE NEAR MCC 62J OCCURS	2.63E-04	IEV-FI1
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	6.22E-05	02----SW-----CM

Title: F12: MCC 62A Scenario
File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	4.02E-06	86.51	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-501 FAILS TO OPEN	3.00E-03	39-CB-1-501--FO
2	1.93E-07	4.15	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
3	8.56E-08	1.84	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
4	6.97E-08	1.50	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
5	3.22E-08	.69	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM BUS BRA-102 TO BUS BRA-104 TRANSFERS OPEN	2.40E-05	38-CBA102-04-CO
6	3.22E-08	.69	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15201 TRANSFERS OPEN	2.40E-05	40-CB--15201-CO
7	3.22E-08	.69	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER SUPPLY BREAKER 1-505 TRANSFERS OPEN	2.40E-05	39-CB--1-505-CO
8	2.60E-08	.56	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER 1-52 FAILURE	1.94E-05	40-TR--1-52--SG
9	[illegible]	[illegible]	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			AOV SW 301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
10	1.50E-08	.32	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
11	[illegible]	.26	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F12: MCC 62A Scenario

File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)

Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
12	1.21E-08	.26	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
13	1.07E-08	.23	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER BRA-104 CIRCUIT 7 TRANSFERS OPEN	8.00E-06	38-CBA104-07-CO
14	8.55E-09	.18	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
15	7.92E-09	.17	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
16	7.65E-09	.16	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
17	6.62E-09	.14	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
18	5.73E-09	.12	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
19	3.82E-09	.08	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
20	3.[illegible]E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A--FS

Title: F12: MCC 62A Scenario
File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
21	3.22E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			125VDC BUS BRA-104 FAILURE	2.40E-06	38-BS-BRA104-SG
22	3.22E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			125VDC BUS BRA-102 FAILURE	2.40E-06	38-BS-BRA102-SG
23	3.22E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BUS 5 FAILURE	2.40E-06	39-BS-BUS5---SG
24	3.22E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BUS 52 FAILURE	2.40E-06	40-BS-BUS52--SG
25	3.12E-09	.07	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
26	2.93E-09	.06	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
27	2.77E-09	.06	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			RESERVE AUX TRANSFORMER FAILURE	4.30E-05	39-TR-RAT----SG
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
28	1.89E-09	.04	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.72E-03	05BPM-ALOP1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
29	1.78E-09	.04	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH MAIN FEEDWATER	1.19E-03	05A--OM2-OM4-HE
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	5.00E-01	36-0820850860HE
30	1.53E-09	.03	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1C UNAVAILABLE DUE TO TEST OR MAINTENANCE	3.00E-03	05BPM--AFW1C-TM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE

Title: F12: MCC 62A Scenario
File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
31	1.23E-09	.03	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			RESERVE AUX TRANSFORMER FAILURE	4.30E-05	39-TR-RAT----SG
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
32	1.02E-09	.02	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			SOV AFW-111C FAILS TO OPEN	2.00E-03	05BSVAFW111C-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
33	8.48E-10	.02	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			AOV SW-301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
34	7.88E-10	.02	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-503 FAILS TO CLOSE	3.00E-03	39-CB-1-503--FC
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
35	5.93E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
36	4.78E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
37	4.78E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
38	4.68E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
39	4.28E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE AFW-1A FAILS TO OPEN	1.00E-04	05BCV--AFW1A-FO
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT-AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
[illegible]	[illegible]	[illegible]	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12

Title: F12: MCC 62A Scenario
File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-311A FAILS TO OPEN	1.00E-04	05BCV-MU311A-FO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
41	3.75E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3B TRANSFERS OPEN	1.20E-05	03-AV--MU3B--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
42	3.75E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3A TRANSFERS OPEN	1.20E-05	03-AV--MU3A--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
43	3.14E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
44	3.14E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			MOV MS-102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
45	3.07E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			RESERVE AUX TRANSFORMER FAILURE	4.30E-05	39-TR-RAT----SG
			AOV SW-301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
46	2.75E-10	.01	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			MOV MS-102 UNAVAILABLE DUE TO TEST OR MAINTENANCE	5.40E-04	06-MV--MS102-TM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
47	2.32E-10	[illegible]	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.72E-03	05BPM-ALOP1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
48	2.18E-10	.00	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH MAIN FEEDWATER	1.19E-03	05A--OM2-OM4-HE
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	5.00E-01	36-08208508BCHE

Title: F12: MCC 62A Scenario
File: FIRE2.WLK (File created by linking FIRE2.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 4.6470E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
49	2.14E-10	.00	FIRE NEAR MCC	62A IN B DIESEL	GENERATOR ROOM	OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			RESERVE AUX	TRANSFORMER	FAILURE		4.30E-05	39-TR-RAT----SG
			TRAVELING WATER	SCREEN 1A1	MECHANICAL	FAILURE	3.72E-03	02-FLT-TW1A1-PS
50	2.09E-10	.00	FIRE NEAR MCC	62A IN B DIESEL	GENERATOR ROOM	OCCURS	1.34E-03	IEV-F12
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER 1-503	FAILS TO CLOSE			3.00E-03	39-CB-1-503--FC
			SW PUMP 1A1	MECHANICAL	FAILURE		1.40E-02	02-PM-SW1A1--PS
			ROTATING	STRAINER 1A2	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A2-PS

Title: F13: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	4.69E-10	33.29	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
2	2.21E-10	15.68	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
3	8.52E-11	6.05	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	[illegible].23E-03	341-RHR-STOP-HE
4	8.06E-11	5.72	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
5	7.81E-11	5.54	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	3.88E-03	36-0B2085086-HE
6	3.28E-11	2.33	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			COMMON MODE FAILURE OF TRAIN A AND B BAT TO RWST SWITCHOVER	1.63E-03	55--SY--BATAB-CM
7	2.66E-11	1.89	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3B TRANSFERS OPEN	1.20E-05	03-AV--MU3B--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
8	2.66E-11	1.89	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU-3A TRANSFERS OPEN	1.20E-05	03-AV--MU3A--CO
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
9	2.28E-11	1.62	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO

Title: F13: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
10	2.23E-11	1.58	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
11	1.97E-11	1.40	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			COMMON CAUSE FAILURE OF RHR	9.79E-04	34---RHR-----CM
12	1.64E-11	1.16	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			HPR UNAVAILABLE DUE TO COMMON CAUSE	8.14E-04	33R-----HPR--CM
13	1.52E-11	1.08	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			HPI UNAVAILABLE DUE TO COMMON CAUSE	7.56E-04	33I-----HPSI-CM
14	1.46E-11	1.04	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-03	34I-RHR-STOP-HE
15	1.45E-11	1.03	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			CONTAINMENT SUMP STRAINERS PLUGGED	7.20E-04	34RFL---SUMP-PL
16	1.34E-11	.95	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	3.88E-03	36-OB2OB5OB6-HE
17	1.16E-11	.82	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
18	1.01E-11	.72	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO

Title: F13: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
19	9.89E 12	.70	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
20	9.89E 12	.70	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
21	8.90E-12	.63	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
22	8.48E-12	.60	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			COMMON CAUSE FAILURE OF BOTH LPI TRAINS	4.21E-04	34I--LPI-----CM
23	8.25E 12	.59	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
24	6.64E 12	.47	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF CONTROL SIGNAL TO MU-3A VALVE FAILS OPEN	3.00E-06	03-AS-HTLVLC-OP
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
25	6.62E 12	.47	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
26	5.64E 12	.40	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMMON MODE FAILURE OF TRAIN A AND B BAT TO RWST SWITCHOVER	1.63E-03	55--SY--BATAB-CM

Title: FI3: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
27	5.36E-12	.38	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			AUXILIARY BLDG BASEMENT COOLING FAILS DUE TO COMMON CAUSE	2.66E-04	17-SY-ABBC---CM
28	5.13E-12	.36	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
29	4.41E-12	.31	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV MS 102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
30	3.67E-12	.26	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			BLEED AND FEED FAILS DUE TO COMMON CAUSE FAILURES	1.82E-04	36--0B2------CM
31	3.39E-12	.24	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMMON CAUSE FAILURE OF RHR	9.79E-04	34---RHR-----CM
32	3.04E-12	.22	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			COMMON MODE FAILURE OF TRAIN A AND TRAIN B	1.51E-04	55--SY---SIAB-CM
33	2.89E-12	[illegible]	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			ADV SW 301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
34	2.81E-12	.20	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			HPR UNAVAILABLE DUE TO COMMON CAUSE	8.14E-04	33R-----HPR--CM
35	2.61E-12	.19	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-FI3
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F13: Fire Between Buses
File: F.WE3.WLK (File created by , FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			HPI UNAVAILABLE DUE TO COMMON CAUSE	7.56E-04	331-----HPSI-CM
36	2.53E-12	.18	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			AOV SW-301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO
37	2.49E-12	.18	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			CONTAINMENT SUMP STRAINERS PLUGGED	7.20E-04	34RFL---SUMP-PL
38	2.19E-12	.16	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.72E-03	05BPM-ALOP1C-PS
			OPERATOR FAILS TO STOP REACTOR COOLANT PUMPS	2.33E-02	36-RXCP-STOP-HE
39	2.02E-12	.14	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
			TRAVELING WATER SCREEN 1B2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1B2-PS
40	1.79E-12	.13	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-03	341-RHR-STOP-HE
41	1.79E-12	.13	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-03	341-RHR-STOP-HE
42	1.77E-12	.13	FIRE IN BUS 1 AND BUS 2 ROOM OCCURS	9.50E-05	IEV-F13
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
			FEEDER BREAKERS ON 4160V BUS 6 FAIL TO OPEN	4.20E-02	39-CB-BUS6FB-FO

Title: F13: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
43	1.76E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	3.00E-03	05BPM--AFW1C-TM
			OPERATOR FAILS	TO STOP	REACTOR COOLANT	PUMPS	2.33E-02	36-RXCP-STOP-HE
44	1.65E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SOV AFW-111A	FAILS TO OPEN			2.00E-03	05BSVAFW111A-CC
			AFW PUMP 1B	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO ESTABLISH	BLEED AND FEED		3.88E-03	36-OB2OB5OB6-HE
45	1.65E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			SOV AFW-111B	FAILS TO OPEN			2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO ESTABLISH	BLEED AND FEED		3.88E-03	36-OB2OB5OB6-HE
46	1.63E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			FEEDER BREAKERS	ON 4160V BUS 5	FAIL TO OPEN		4.80E-02	39-CB-BUS5FB-FO
			BREAKER FROM	MCC-62D	FAILS TO CLOSE		3.00E-03	40-CB-62D/A6-FC
47	1.63E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			FEEDER BREAKERS	ON 4160V BUS 5	FAIL TO OPEN		4.80E-02	39-CB-BUS5FB-FO
			BREAKER 1-601	FAILS TO OPEN			3.00E-03	39-CB-1-601--FO
48	1.63E-12	.12	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			LOSS OF ALL	POWER FROM GRID	DURING 24 HOURS		1.19E-04	LOSP-24
			FEEDER BREAKERS	ON [illegible]60V BUS 5	FAIL TO OPEN		4.80E-02	39-CB-BUS5FB-FO
			BREAKER 1-603	FAILS TO CLOSE			3.00E-03	39-CB-1-603--FC
49	1.62E-12	.11	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS	TO STOP BOT[illegible]	RHR PUMPS		4.23E-03	34I-RHR-STOP-HE
50	1.48E-12	.11	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		9.50E-05	IEV-F13
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			AFW PUMP 1B	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	1.80E-03	05BPM--AFW1B-TM

Title: FI3: Fire Between Buses 1 and 2
File: FIRE3.WLK (File created by linking FIRE3.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 1.4090E-09

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	3.88E-03	36-OB2OB5OB6-HE

TABLE 4-10 F14 ***** DOMINANT CUTSETS ***** Page: 1

Title: F14: Diesel B Fire
File: FIRE4.WLK (File created by linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
1	1.11E-06	14.91	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP		1.21E-02	35--CHP------HE
			CCW PUMP A	MECHANICAL	FAILURE		1.03E-02	31-PM--CCW1A-PS
2	4.02E-07	5.40	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP		1.21E-02	35--CHP------HE
			TRAVELING WATER	SCREEN 1A1	MECHANICAL	FAILURE	3.72E-03	02-FLT-TW1A1-PS
3	3.24E-07	4.35	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP		1.21E-02	35--CHP------HE
			BREAKER FROM	MCC-52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A6-FC
4	2.14E-07	2.87	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER FROM	BUS BRA-102 TO	BUS BRA-104	TRANSFERS OPEN	2.40E-05	38-CBA102-04-CO
5	2.14E-07	2.87	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER 15201	TRANSFERS OPEN			2.40E-05	40-CB--15201-CO
6	2.14E-07	2.87	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			QUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			TRANSFORMER	SUPPLY BREAKER	1-505	TRANSFERS OPEN	2.40E-05	39-CB--1-505-CO
7	2.05E-07	2.75	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			CCW PUMP A	MECHANICAL	FAILURE		1.03E-02	31-PM--CCW1A-PS
8	1.73E-07	2.32	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			TRANSFORMER	1-52	FAILURE		1.94E-05	40-TR--1-52--SG
9	1.49E-07	2.00	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV SI-4A	FAILS TO OPEN			7.50E-03	331MV---SI4A-CC
10	1.49E-07	2.00	B DIESEL	GENERATOR OIL	FIRE OCCURS		8.92E-03	IEV-FI4
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A	MECHANICAL	FAILURE		1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN	AFW PUMP 1C	MECHANICAL	FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV SI-2A	FAILS TO OPEN			7.50E-03	331MV---SI2A-CC

Title: F14: Diesel B Fire
File: FIRE4.WLK (File created by Linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
11	1.49E-07	2.00	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV SI-350A FAILS TO OPEN	7.50E-03	34RMV-SI350A-CC
12	1.49E-07	2.00	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV SI-351A FAILS TO OPEN	7.50E-03	34RMV-SI351A-CC
13	1.49E-07	2.00	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV CC-400A FAILS TO OPEN	7.50E-03	34RMV-CC400A-CC
14	1.49E-07	2.00	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV RHR 300A FAILS TO OPEN	7.50E-03	33RMVRHR300A-FO
15	1.42E-07	1.91	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMPRESSOR 1A MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1A-PS
			COMPRESSOR 1C MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1C-PS
16	9.74E-08	1.31	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP------CM
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
17	9.48E-08	1.27	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			MOV SI-5A FAILS TO CLOSE	4.76E-03	33RMV---SI5A-FC
18	8.43E-08	1.13	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-03	341-RHR-STOP-HE
19	7.98E-08	1.06	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F14: Diesel B Fire
File: FIRE4.WLK (F - created by linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			OPERATOR FAILS TO START ONE CHARGING PUMP	1.21E-02	35--CHP------HE
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
20	7.73E-08	1.04	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	3.88E-03	36-082085086-HE
21	7.41E-08	1.00	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
22	5.98E-08	.80	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
23	5.84E-08	.78	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			-NOT-TRAIN B UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.94E-03	-55--SY-ESFTRB-TM
			TRAIN A UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.94E-03	55--SY-ESFTRA-TM
24	5.10E-08	.69	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			FEEDER BREAKERS ON 4160V BUS 5 FAIL TO OPEN	4.80E-02	39-CB-BUS5FB-FO
25	4.86E-08	.65	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			AIR COMPRESSORS FAIL DUE TO COMMON CAUSE	2.44E-03	01-PM-IASP---CM
26	4.79E-08	.64	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START ONE CHARGING PUMP	1.21E-02	35--CHP------HE
			FAILURE OF TIME DELAY RELAY TDR-BLS/B5	4.44E-04	55--RE-TDBSB5-RF
27	4.12E-08	.55	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SI PUMP 1A FAILS TO START AND RUN	2.07E-03	33RPM---SI1A-PS

Title: F14: Diesel B Fire
File: FIRE4.WLK (File created by linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
28	3.98E-08	.53	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SW CONTROL VALVE AOV 31746 FAILS TO OPEN	2.00E-03	17-AVCV31746-CC
29	3.98E-08	.53	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SW CONTROL VALVE SOV 33313 FAILS TO OPEN	2.00E-03	17-SVSV33313-CC
30	3.98E-08	.53	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SOV SW-1211A FAILS TO OPEN	2.00E-03	34RSVSW1211A-CC
31	3.56E-08	.48	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START ONE CHARGING PUMP	1.21E-02	35--CHP------HE
			BOTH CCW PUMPS FAIL DUE TO LOSS OF OFFSITE POWER	3.30E-04	31-PM-CCW1AB-CM
32	3.52E-08	.47	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP------CM
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
33	3.33E-08	.45	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			RHR PUMP 1A FAILS TO START AND RUN	1.67E-03	34RPM--RHR1A-PS
34	3.25E-08	.44	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMMON MODE FAILURE OF TRAIN A AND B BAT TO RWST SWITCHOVER	1.63E-03	55--SY--BATAB-CM
35	3.11E-08	.42	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			SI PUMP 1A FAILS TO START AND RUN	1.56E-03	33IPM---SI1A-PS
36	2.93E-08	.39	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15206 TRANSFERS OPEN	2.40E-05	40-CB--15206-CO
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS

Title: F14: Diesel B Fire
File: FIRE4.WLK (File created by linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
37	2.84E-08	.38	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-FI4
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP------CM
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
38	2.83E-08	.38	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			RHR PUMP 1A FAILS TO START AND RUN	1.42E-03	341PM--RHR1A-PS
39	2.52E-08	.34	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-FI4
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW 111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
40	[illegible]	.32	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			RELIEF VALVE CVC-101A FAILS TO CLOSE	1.60E-02	35-AMCVC101A-FC
			RELIEF VALVE CVC-101C FAILS TO CLOSE	1.60E-02	35-AMCVC101C FC
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
41	2.26E-08	.30	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IE F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF ALL POWER FROM GRID DURING 24 HOURS	1.19E-04	LOSP-24
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
42	[illegible]	.29	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			RHR TRAIN A IN TEST/MAINTENANCE	1.09E-03	341--RHRTRNA-TM
43	[illegible]	.29	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMPARATOR PC-628 FAILS	1.08E-03	55--AD--PC628-FA
44	[illegible]	.29	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-FI4
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			125VDC BUS BRA-104 FAILURE	2.40E-06	38-BS-BRA104-SG
45	[illegible]	.29	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-FI4
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			125VDC BUS BRA 102 FAILURE	2.40E-06	38-BS-BRA102-SG
46	[illegible]	.29	B DIESEL GENERATOR OIL FIRE OCCURS	8.92E-03	IEV-FI4
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

TABLE 4-10 F14 ***** DOMINANT CUTSETS ***** Page: 6

Title: F14: Diesel B Fire
File: FIRE4.WLK (File created by linking FIRE4.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 7.4440E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME			EVENT PROB.	IDENTIFIER
			BUS 5	FAILURE		2.40E-06	39-BS-BUS5---SG
47	2.14E-08	.29	B DIESEL	GENERATOR OIL	FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS	DUE TO FIRE		1.00E+00	FIRE-DAMAGE
			BUS 52	FAILURE		2.40E-06	40-BS-BUS52--SG
48	2.14E-08	.29	B DIESEL	GENERATOR OIL	FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS	DUE TO FIRE		1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP	1.21E-02	35--CHP------HE
			FAILURE OF	SEQUENCER RELAY	BLS/B5	1.98E-04	55--RE--BLSB5 RF
49	2.14E-08	.29	B DIESEL	GENERATOR OIL	FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS	DUE TO FIRE		1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP	1.21E-02	35--CHP------HE
			FAILURE OF	SEQUENCER RELAY	BS/B5 TO OPERATE	1.98E-04	55--RE---BSB5-RF
50	2.14E-08	.29	B DIESEL	GENERATOR OIL	FIRE OCCURS	8.92E-03	IEV-F14
			EQUIPMENT FAILS	DUE TO FIRE		1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO START ONE	CHARGING PUMP	1.21E-02	35--CHP------HE
			RELAY BLX/B5	FAILS TO OPERATE		1.98E-04	55--RE--BLXB5-RF

Title: F15: Fire in Relay Room near Vertical Trays
File: FIRE5.WLK (File created by Linking FIRE5.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.2110E-07

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	1.82E-07	56.68	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START AND LOAD DIESEL FROM DSP	2.69E-01	42-DGA-DSP---HE
2	5.46E-08	17.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH SERVICE WATER FROM DSP	8.06E-02	02-SWA-DSP---HE
3	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW FROM DSP	5.19E-02	05BAF3-DSP---HE
4	1.44E-08	4.48	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
5	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.53E-02	05BPM--AFW1A-PS
6	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV SW 301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
7	3.34E-09	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.60E-01	35-CHP-DSP---HE
8	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
9	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
10	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 520 FAILS TO CLOSE	3.00E-03	40-CB-520/A6-FC
11	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC
12	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F15: Fire in Relay Room near Vertical Trays
File: FIRE5.WLK (File created by linking FIRE5.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.2110E-07

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			BREAKER 1-501 FAILS TO OPEN	3.00E-03	39-CB-1-501--FO
13	1.36E-09	.42	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
14	1.34E-09	.42	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	[illegible]	FIRE-DAMAGE
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
15	9.09E-10	.28	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
16	[illegible]	.22	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
[illegible]	[illegible]	.15	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02---SWS----CM
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE AFW 1A FAILS TO OPEN	1.00E-04	05BCV--AFW1A-FO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-311A FAILS TO OPEN	1.00E-04	05B--MU311A-FO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.4[illegible]E-02	02-PM-SW1A1--PS
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	3.7[illegible]E-03	02-FLR-RS1A2-PS
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15

Title: F15: Fire in Relay Room near Vertical Trays
File: FIRE5.WLK (File created by linking FIRE5.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.2110E-07

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
				1.00E+00	FIRE-DAMAGE
			EQUIPMENT FAILS DUE TO FIRE		
			ROTATING STRAINER 1A1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A1-PS
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
[illegible]	2.85E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A5-FC
26	2.85E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A4-FC
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
27	2.75E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
28	2.30E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			BOTH CCW PUMPS FAIL DUE TO LOSS OF OFFSITE POWER	3.30E-04	31-PM-CCW1AB-CM
29	2.21E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP-----CM
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
30	1.63E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM BUS BRA 102 TO BUS BRA 104 TRANSFERS OPEN	2.40E-05	38-CBA102-04-CO
31	1.63E-11	.01	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15208 TRANSFERS OPEN	2.40E-05	40-CB--15208-CO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 152[illegible] TRANSFERS OPEN	2.40E-05	40-CB--15201-CO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER SUPPLY BREAKER 1-5[illegible]5 TRANSFERS OPEN	2.40E-05	39-CB--1-5[illegible]5-CO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15206 TRANSFERS OPEN	2.40E-05	40-CB--15206-CO
[illegible]	[illegible]	[illegible]	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15

TABLE 4 11 F15 ***** DOMINANT CUTSETS ***** Page: 4

Title: F15: Fire in Relay Room near Vertical Trays
File: FIRES.WLK (File created by linking FIRES.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.2110E-07

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER 1-52 FAILURE	1.94E-05	40-TR--1-52--SG
36	9.38E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			ROTATING STRAINER 1A1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A1-PS
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A2-PS
37	8.14E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU 3B TRANSFERS OPEN	1.20E-05	03-AV--MU3B--CO
38	8.14E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV MU 3A TRANSFERS OPEN	1.20E-05	03-AV--MU3A--CO
39	7.57E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A4-FC
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A2-PS
40	7.57E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			ROTATING STRAINER 1A1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A1-PS
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A5-FC
41	7.[illegible]E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP------CM
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
42	7.31E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A2-PS
43	[illegible]E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP--HE
			CHECK VALVE CC 3A FAILS TO OPEN	1.00E-04	31-CV---CC3A-FO
44	6.83E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			SW STRAINER 1A2 PLUGGED	7.20E-04	02-FL--1A2---PL
45	6.[illegible]E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A4-FC
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A5-FC

Title: F15- Fire in Relay Room near Vertical Trays
File: FIRES.WLK (File created by linking FIRES.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.2110E-07

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
46	5.90E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A5-FC
47	5.42E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER BRA 104 CIRCUIT 7 TRANSFERS OPEN	8.00E-06	38-CBA104-07-CO
48	[illegible].35E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			RELIEF VALVE CVC 101A FAILS TO CLOSE	1.60E-02	35-AMCVC101A-FC
			RELIEF VALVE CVC 101C FAILS TO CLOSE	1.60E-02	35-AMCVC101C-FC
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
49	2.63E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			LOSS OF CONTROL SIGNAL TO MU 3A VALVE FAILS OPEN	3.00E-06	03-AS-HTLVLC-OP
50	[illegible].21E-12	.00	FIRE IN RELAY ROOM OCCURS	6.78E-07	IEV-F15
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			CCW HEAT EXCHANGER 1A SHELL LEAK	2.74E-05	31-HE-CCW1A-HS

Title: F16. AFW Pump A Scenario (TU 95C)
File: FIRE6.WLK (File created by linking FIRE6.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 5.2700E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	1.37E-05	26.00	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR 1B FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
2	6.64E-06	12.60	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS
3	5.[illegible]8E-06	9.64	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START AND LOAD DIESEL MANUALLY	7.87E-03	42-DGB-MAN---HE
4	4.18E-06	7.93	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
5	3.43E-06	6.51	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV SW 301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
6	2.40E-06	4.55	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRAVELING WATER SCREEN 1B2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1B2-PS
[illegible]	[illegible]94E-06	3.68	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A6-FC
8	[illegible]94E[illegible]	3.68	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-603 FAILS TO CLOSE	3.00E-03	39-CB-1-603--FC
[illegible]	[illegible]94E-06	3.68	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-601 FAILS TO OPEN	3.00E-03	39-CB-1-601--FO
[illegible]	[illegible]3E-06	[illegible]	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			[illegible] MP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.37E-03	31-PM--CCW1B-TM
11	1.27E-06	2.41	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
12	[illegible]11E-06	2.11	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH SERVICE WATER MANUALLY	1.72E-03	02-SWB-MAN---HE

Title: F16: AFW Pump A Scenario (TU 95c)
File: F16.WLK (File created by linking FIRE6.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 5.2700E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
13	1.03E-06	1.95	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CCW MANUALLY	1.59E-03	31-CCL-MAN---HE
14	8.87E-07	1.68	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-S1AC1B-PS
15	8.16E-07	1.55	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
16	5.13E-07	.97	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW 111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
17	4.71E-07	.89	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
18	4.62E-07	.88	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
19	3.08E-07	.58	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			OPERATOR FAILS TO ESTABLISH SAFETY INJECTION MANUALLY	1.50E-01	33-HPI-MAN--DHE
20	2.13E-07	.40	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BOTH CCW PUMPS FAIL DUE TO LOSS OF OFFSITE POWER	3.30E-04	31-PM-CCW1AB-CM
21	1.73E-07	.33	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-S1AC1B-PS
22	1.26E-07	.24	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
23	1.13E-07	.21	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.75E-04	10-GE-DG1B--TM

Title: F16: AFW Pump A Scenario (TU 95C)
File: FIRE6.WLK (File created by linking FIRE6.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 5.2700E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
24	1.12E-07	.21	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW HEAT EXCH 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.73E-04	31-HE--CCW1B-TM
25	1.09E-07	.21	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW 111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
26	9.80E-08	.19	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			COMPRESSOR 1B MECHANICA FAILURE	8.44E-02	01-PM-SIAC1B-PS
27	7.89E-08	.15	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV SI 4B FAILS TO OPEN	7.50E-03	33IMV---SI4B-CC
28	7.89E-08	.15	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV SI 350B FAILS TO OPEN	7.50E-03	34RMV-SI350B-CC
29	7.89E-08	.15	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV SI 351B FAILS TO OPEN	7.50E-03	34RMV-SI351B-CC
30	7.89E-08	.15	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV CC 400B FAILS TO OPEN	7.50E-03	34RMV-CC400B-CC
31	7.89E-08	.15	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV RHR 300B FAILS TO OPEN	7.50E-03	33RMVRHR300B-FO
32	6.45E-08	.12	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE CC 3B FAILS TO OPEN	1.00E-04	31-CV--CC3B-FO
33	5.86E-08	.11	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			OPERATOR FAILS TO ESTABLISH SAFETY INJECTION MANUALLY	5.57E-03	33-HPI-MAN--HE
34	5.59E-08	.11	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F16: AFW Pump A Scenario (TU-95C)
File: FIRE6.WLK (File created by Linking FIRE6.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 5.2700E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AOV SW 420B FAILS TO OPEN	5.32E-03	02-AV-SW420B-FO
35	5.44E-08	.10	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
36	5.00E-08	.09	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV SI 5B FAILS TO CLOSE	4.76E-03	33RMV---SI5B-FC
37	4.45E-08	.08	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			OPERATOR FAILS TO STOP BOTH RHR PUMPS	4.23E-03	341-RHR-STOP-HE
38	4.08E-08	.08	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			OPERATOR FAILS TO ESTABLISH BLEED AND FEED	3.88E-03	36-082085086-HE
39	3.36E-08	.06	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			ROTATING STRAINER 1B2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1B2-PS
40	3.36E-08	.06	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			ROTATING STRAINER 1B1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1B1-PS
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
41	3.15E-08	.06	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			BREAKER FROM MCC 62A FAILS TO CLOSE	3.00E-03	40-CB-62A/A5-FC
42	2.71E-08	.05	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			BREAKER FROM MCC-62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A5-FC
43	2.71E-08	.05	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A4-FC
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
44	2.57E-08	.05	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

TABLE 4.12 F16 ***** DOMINANT CUTSETS ***** Page: 5

Title: F16: AFW Pump A Scenario ('J-95C)
File: FIRE6.WLK (File created by nking FIRE6.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 5.2700E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			CHECK VALVE AFW-4B FAILS TO OPEN	1.00E-04	05BCV--AFW4B-FO
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
45	2.57E-08	.05	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE AFW-1B FAILS TO OPEN	1.00E-04	05BCV--AFW1B-FO
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
46	2.57E-08	.05	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AIR COMPRESSORS FAIL DUE TO COMMON CAUSE	2.44E-03	01-PM-IASP---CM
47	2.31E-08	.04	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			OPERATOR FAILS TO LOCALLY OPEN MOVS SW1300A(B)	2.20E-03	31-LO-SW1300-HE
48	2.18E-08	.04	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			SI PUMP 1B FAILS TO START AND RUN	2.07E-03	33RPM---SI1B-PS
[illegible]	2.10E-08	.04	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			SOV SW 402B FAILS TO OPEN	2.00E-03	02-SV-SW402B-FO
[illegible]	[illegible]E-08	.04	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	6.45E-04	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			[illegible] 33113 FAILS TO OPEN	2.00E-03	36-SV-33113--FO

Title: F17: AFW Pump B Fire
File: FIRE7.WLK (File created by linking FIRE7.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.9670E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	1.67E-05	56.29	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START AND LOAD DIESEL FROM DSP	2.69E-01	42-DGA-DSP---HE
2	5.00E-06	16.85	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH SERVICE WATER FROM DSP	8.06E-02	02-SWA-DSP---HE
3	3.22E-06	10.85	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW FROM DSP	5.19E-02	05BAF3-DSP---HE
4	1.32E-06	4.45	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR 1A FAILURE TO START AND RUN	2.13E-02	10-GE-DG1A---PS
5	1.01E-06	3.40	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A MECHANICAL FAILURE	1.63E-02	05BPM--AFW1A-PS
6	3.30E-07	1.11	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV SW 301A FAILS TO OPEN	5.32E-03	02-AV-SW301A-CC
7	3.06E-07	1.03	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.60E-01	35-CHP-DSP--DHE
8	2.96E-07	1.00	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			MOV BT 2A FAILS TO CLOSE	4.77E-03	07-MV---BT2A-OO
9	2.31E-07	.78	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRAVELING WATER SCREEN 1A1 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1A1-PS
10	1.97E-07	.66	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE
11	1.86E-07	.63	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A6-FC
12	1.86E-07	.63	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F17: AFW Pump B Fire
File: FIRE7.WLK (File created by linking FIRE7.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.9670E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			BREAKER 1-509 FAILS TO CLOSE	3.00E-03	39-CB-1-509--FC
13	1.86E-07	.63	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-501 FAILS TO OPEN	3.00E-03	39-CB-1-501--FO
14	1.24E-07	.42	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111A FAILS TO OPEN	2.00E-03	05BSVAFW111A-CC
15	1.22E-07	.41	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
16	8.31E-08	.28	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1A UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.34E-03	05BPM--AFW1A-TM
17	6.58E-08	.22	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			CCW PUMP A MECHANICAL FAILURE	1.03E-02	31-PM--CCW1A-PS
18	4.53E-08	.15	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E-00	FIRE-DAMAGE
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
19	1.31E-08	.04	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
20	1.22E-08	.04	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
21	6.20E-09	.02	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE MU-301 FAILS TO OPEN	1.00E-04	27-CV-MU301--FO
22	6.20E-09	.02	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE AFW-4A FAILS TO OPEN	1.00E-04	05BCV--AFW4A-FO
23	6.20E-09	.02	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE AFW-1A FAILS TO OPEN	1.00E-04	05BCV--AFW1A-FO
24	6.20E-09	.02	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F17: AFW Pump B Fire
File: FIRE7.WLK (File created by linking FIRE7.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.9670E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			CHECK VALVE MU-311A FAILS TO OPEN	1.00E-04	05BCV-MU311A-FO
25	3.23E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			ROTATING STRAINER 1A2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A2-PS
26	3.23E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			ROTATING STRAINER 1A1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1A1-PS
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
27	2.60E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A1--PS
			BREAKER FROM MCC-52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A5-FC
28	2.60E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 52D FAILS TO CLOSE	3.00E-03	40-CB-52D/A4-FC
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
29	2.52E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1 UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			SW PUMP 1A2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1A2--PS
30	2.11E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CHARGING USING DSP	1.03E-01	35-CHP-DSP---HE
			BOTH CCW PUMPS FAIL DUE TO LOSS OF OFFSITE POWER	3.30E-04	31-PM-CCW1AB-CM
31	2.02E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW DUE TO COMMON CAUSE FAILURES	1.06E-03	35--CHP------ M
			OPERATOR FAILS TO ESTABLISH COMPONENT COOLING FROM DSP	3.08E-02	31-CCL-DSP- HE
32	1.49E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM BUS BRA-102 TO BUS BRA-104 TRANSFERS OPEN	2.40E-05	38-CBA102-04-CO
33	1.49E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15208 TRANSFERS OPEN	2.40E-05	40-CB--15208 CO
34	1.49E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 15201 TRANSFERS OPEN	2.40E-05	40-CB--15201-CO
35	1.49E-09	.01	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	6.20E-05	IEV-F17

Title: F17: AFW Pump B Fire
File: FIRE7.WLK (File created by linking FIRE7.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.9670E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			TRANSFORMER	SUPPLY BREAKER	1-505	TRANSFERS OPEN	2.40E-05	39-CB--1-505-CO
36	1.49E-09	.01	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER 15206	TRANSFERS OPEN			2.40E-05	40-CB--15206-CO
37	1.20E-09	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			TRANSFORMER	1-52	FAILURE		1.94E-05	40-TR--1-52--SG
38	8.58E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			ROTATING	STRAINER 1A1	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A1-PS
			ROTATING	STRAINER 1A2	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A2-PS
39	7.44E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AOV MU-3B	TRANSFERS OPEN			1.20E-05	03-AV--MU3B--CO
40	7.44E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			AOV MU-3A	TRANSFERS OPEN			1.20E-05	03-AV--MU3A--CO
41	6.92E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER FROM	MCC 52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A4-FC
			ROTATING	STRAINER 1A2	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A2-PS
42	6.92E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			ROTATING	STRAINER 1A1	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A1-PS
			BREAKER FROM	MCC-52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A5-FC
43	6.77E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			NO CHARGING FLOW	DUE TO	COMMON CAUSE	FAILURES	1.06E-03	35--CHP-----CM
			CCW PUMP A	MECHANICAL	FAILURE		1.03E-02	31-PM--CCW1A-PS
44	6.69E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			ROTATING	STRAINER 1A2	MECHANICAL	FAILURE	3.72E-03	02-FLR-RS1A2-PS
45	6.39E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS	TO ESTABLISH	CHARGING USING	DSP	1.03E-01	35-CHP-DSP--HE
			CHECK VALVE	CC-3A	FAILS TO OPEN		1.00E-04	31-CV---CC3A-FO
46	6.25E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17

Title: F17: AFW Pump B Fire
File: FIRE7.WLK (File created by linking FIRE7.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 2.9670E-05

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME				EVENT PROB.	IDENTIFIER
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1	MECHANICAL	FAILURE		1.40E-02	02-PM-SW1A1--PS
			SW STRAINER 1A2	PLUGGED			7.20E-04	02-FL-1A2----PL
47	5.58E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER FROM	MCC-52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A4-FC
			BREAKER FROM	MCC-52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A5-FC
48	5.39E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			SW PUMP 1A1	UNAVAILABLE	DUE TO TEST OR	MAINTENANCE	2.90E-03	02-PM-SWP1A1-TM
			BREAKER FROM	MCC-52D	FAILS TO CLOSE		3.00E-03	40-CB-52D/A5-FC
49	4.96E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			BREAKER BRA-104	CIRCUIT 7	TRANSFERS OPEN		8.00E-06	38-CBA104-07-CO
50	4.89E-10	.00	AUXILIARY	FEEDWATER PUMP B	OIL FIRE OCCURS		6.20E-05	IEV-F17
			EQUIPMENT FAILS	DUE TO FIRE			1.00E+00	FIRE-DAMAGE
			RELIEF VALVE	CVC-101A	FAILS TO CLOSE		1.60E-02	35-AMCVC101A-FC
			RELIEF VALVE	CVC-101C	FAILS TO CLOSE		1.60E-02	35-AMCVC101C-FC
			OPERATOR FAILS	TO ESTABLISH	COMPONENT	COOLING FROM DSP	3.08E-02	31-CCL-DSP---HE

TABLE 4-14 F18 ***** DOMINANT CUTSETS ***** Page: 1

Title: F18: Fire in TU-95A Between Buses 51 and 52
File: FIRE8.WLK (File created by linking FIRE8.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.3300E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
1	9.90E-07	29.73	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR FAILURE TO START AND RUN	2.13E-02	10-GE-DG1B---PS
2	4.79E-07	14.38	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW PUMP B MECHANICAL FAILURE	1.03E-02	31-PM--CCW1B-PS
3	3.66E-07	10.99	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO START AND LOAD DIESEL MANUALLY	7.87E-03	42-DGB-MAN---HE
4	2.47E-07	7.42	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AOV SW-301B FAILS TO OPEN	5.32E-03	02-AV-SW301B-CC
5	1.73E-07	5.20	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRAVELING WATER SCREEN 1B2 MECHANICAL FAILURE	3.72E-03	02-FLT-TW1B2-PS
6	1.40E-07	4.20	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC 62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A6-FC
7	1.40E-07	4.20	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-603 FAILS TO CLOSE	3.00E-03	39-CB-1-603--FC
8	1.40E-07	4.20	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 1-601 FAILS TO OPEN	3.00E-03	39-CB-1-601--FO
9	[illegible]	3.30	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	2.37E-03	31-PM--CCW1B-TM
[illegible]	[illegible]	[illegible]	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURE OF BOTH DIESEL GENERATORS	1.97E-03	10-GE-DGAB---CM
11	[illegible]	[illegible]	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH SERVICE WATER MANUALLY	1.72E-03	02-SWB-MAN---HE
12	7.39E-08	2.22	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH CCW MANUALLY	1.59E-03	31-CCL-MAN---HE

Title: F18: Fire in TU 95A Between Buses 51 and 52
File: FIRE8.WLK (File created by linking FIRE8.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.3300E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
13	5.89E-08	1.77	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
14	4.13E-08	1.24	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
15	3.39E-08	1.02	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			NO SERVICE WATER DUE TO COMMON CAUSE FAILURES	7.30E-04	02----SWS----CM
16	2.22E-08	.67	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			OPERATOR FAILS TO ESTABLISH SAFETY INJECTION MANUALLY	1.50E-01	33-HPI-MAN--DHE
17	1.53E-08	.46	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BOTH CCW PUMPS FAIL DUE TO LOSS OF OFFSITE POWER	3.30E-04	31-PM-CCW1AB-CM
18	1.25E-08	.38	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
19	9.11E-09	.27	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
20	8.76E-09	.26	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
21	8.14E-09	.24	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			DIESEL GENERATOR 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.75E-04	[illegible] GE-DG1B---TM
22	8.[illegible]4E-09	.24	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW HEAT EXCH 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.73E-04	31-HE--CCW1B-TM
23	5.07E-09	.15	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE

Title: F18: Fire in TU-95A Between Buses 51 and 52
File: FIRE8.WLK (File created by linking FIRE8.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.3300E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
24	4.65E-09	.14	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CHECK VALVE CC-38 FAILS TO OPEN	1.00E-04	31-CV---CC38-FO
25	4.56E-09	.14	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
26	3.92E-09	.12	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
27	2.42E-09	.07	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			ROTATING STRAINER 1B2 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1B2-PS
28	2.42E-09	.07	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			ROTATING STRAINER 1B1 MECHANICAL FAILURE	3.72E-03	02-FLR-RS1B1-PS
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
29	2.26E-09	.07	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			MOV MS 102 FAILS TO OPEN	7.50E-03	06-MV--MS102-CC
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
30	1.95E-09	.06	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SW PUMP 1B1 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B1--PS
			BREAKER FROM MCC-62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A5-FC
31	1.95E-09	.06	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM MCC-62D FAILS TO CLOSE	3.00E-03	40-CB-62D/A4-FC
			SW PUMP 1B2 MECHANICAL FAILURE	1.40E-02	02-PM-SW1B2--PS
32	1.27E-09	.04	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			CCW HEAT EXCHANGER 1B SHELL LEAK	2.74E-05	31-HE--CCW1B-HS
33	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18

TABLE 4-14 F18 ***** DOMINANT CUTSETS ***** Page: 4

Title: F18: Fire in TU-95A Between Buses 51 and 52
File: FIRE8.WLK (File created by linking FIRE8.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.3300E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C AUX LUBE OIL PUMP MECHANICAL FAILURE	3.72E-03	05BPM-ALOP1C-PS
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	3.98E-01	01-IAS-MAN---HE
34	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER FROM BUS BRB-102 TO BUS BRB-104 TRANSFERS OPEN	2.40E-05	38-CBB102-04-CO
35	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 16208 TRANSFERS OPEN	2.40E-05	40-CB--16208-CO
36	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 16201 TRANSFERS OPEN	2.40E-05	40-CB--16201-CO
37	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER SUPPLY BREAKER 1-607 TRANSFERS OPEN	2.40E-05	39-CB--1-607-CO
38	1.12E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			BREAKER 16101 TRANSFERS OPEN	2.40E-05	40-CB--16101-CO
39	1.11E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			MOV SI-4B FAILS TO OPEN	7.50E-03	33IMV---SI4B-CC
40	1.11E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			MOV SI-350B FAILS TO OPEN	7.50E-03	34RMV-SI350B-CC
41	1.11E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			MOV SI-351B FAILS TO OPEN	7.50E-03	34RMV-SI351B-CC
42	1.11E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			MOV CC-400B FAILS TO OPEN	7.50E-03	34RMV-CC400B-CC
43	1.11E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			MOV RHR-300B FAILS TO OPEN	7.50E-03	33RMVRHR300B-FO

Title: F18: Fire in TU-95A Between Buses 51 and 52
File: FIRE8.WLK (File created by linking FIRE8.IN WLINK ** Ver. 3.11 **)
Reduced Sum of Cutsets: 3.3300E-06

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME	EVENT PROB.	IDENTIFIER
44	1.08E-09	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			SOV AFW-111B FAILS TO OPEN	2.00E-03	05BSVAFW111B-CC
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
45	9.68E-10	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B UNAVAILABLE DUE TO TEST OR MAINTENANCE	1.80E-03	05BPM--AFW1B-TM
			TURBINE DRIVEN AFW PUMP 1C MECHANICAL FAILURE	1.37E-01	05BPT--AFW1C-PS
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
46	9.05E-10	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			AFW PUMP 1B MECHANICAL FAILURE	1.63E-02	05BPM--AFW1B-PS
			AFW PUMP 1C UNAVAILABLE DUE TO TEST OR MAINTENANCE	3.00E-03	05BPM--AFW1C-TM
			OPERATOR FAILS TO LOCALLY ESTABLISH INSTRUMENT AIR	2.98E-01	01-IAS-MAN--HE
47	9.02E-10	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER 1-62 FAILURE	1.94E-05	40-TR--1-62--SG
48	9.02E-10	.03	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			TRANSFORMER 1-61 FAILURE	1.94E-05	40-TR--1-61--SG
49	8.32E-10	.02	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			COMMON CAUSE FAILURES OF AFW SYSTEM	2.12E-04	05B--AFW-----CM
			COMPRESSOR 1B MECHANICAL FAILURE	8.44E-02	01-PM-SIAC1B-PS
50	8.24E-10	.02	FIRE NEAR BUSES 51 AND 52 OCCURS	4.65E-05	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	1.00E+00	FIRE-DAMAGE
			OPERATOR FAILS TO ESTABLISH AFW FLOW MANUALLY	3.18E-03	05BAF3-MAN---HE
			OPERATOR FAILS TO ESTABLISH SAFETY INJECTION MANUALLY	5.57E-03	33-HPI-MAN---HE

TABLE 4 15 ***** DOMINANT SEQUENCES ***** Page: 1

Title: Total [illegible] e Melt Due to Fires
File: FIRE [illegible] T (File created by linking firetot.in WLINK ** Ver. 3.11 **)
Reduced Su [illegible] r Sequence Probabilities: 9.8116E-05

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	4.21E-05	42.86	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB5
2	2.91E-05	29.68	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			AFW SYSTEM FAILS	SYS-AF3
3	9.65E-06	9.83	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCL
4	4.50E-06	4.59	FIRE NEAR MCC 62A IN B DIESEL GENERATOR ROOM OCCURS	IEV-F12
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			CHARGING SYSTEM FAILS	SYS-CHG
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCT
5	3.56E-06	3.63	B DIESEL GENERATOR OIL FIRE OCCURS	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			CHARGING SYSTEM FAILS	SYS-CHP
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCL
6	3.13E-06	3.19	FIRE NEAR BUSES 51 AND 52 OCCURS	IEV-F18
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCL
7	2.18E-06	2.22	B DIESEL GENERATOR OIL FIRE OCCURS	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			AFW SYSTEM FAILS	SYS-AF3
			BLEED AND FEED FAILS	SYS-OB5
8	1.70E-06	1.73	B DIESEL GENERATOR OIL FIRE OCCURS	IEV-F14
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
9	9.83E-07	[illegible]	AUXILIARY FEEDWATER PUMP A OIL FIRE OCCURS	IEV-F16
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			AFW SYSTEM FAILS	SYS-AF3
			HIGH PRESSURE RECIRCULATION FAILS	SYS-HR1
10	[illegible].76E-07	.59	AUXILIARY FEEDWATER PUMP B OIL FIRE OCCURS	IEV-F17
			EQUIPMENT FAILS DUE TO FIRE	FIRE-DAMAGE
			CHARGING SYSTEM FAILS	SYS-CHP
			COMPONENT COOLING WATER SYSTEM FAILS	SYS-CCL

Title: Total Core Melt Due to Fires
File: FIRETOT.LST (File created by linking firetot.in WLINK ** Ver. 3.11 **)
Reduced Sum of Sequence Probabilities: 9.8116E-05

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION				SEQUENCE IDENTIFIER
11	3.14E-07	.32	FIRE IN RELAY	ROOM OCCURS			IEV-FI5
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
12	1.79E-07	.18	FIRE NEAR BUSES	51 AND 52 OCCURS			IEV-FI8
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			BLEED AND FEED	FAILS			SYS-OB5
13	1.48E-07	.15	FIRE NEAR MCC	62A IN B DIESEL	GENERATOR ROOM	OCCURS	IEV-FI2
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
			BLEED AND FEED	FAILS			SYS-OB2
14	1.85E-08	.02	FIRE NEAR BUSES	51 AND 52 OCCURS			IEV-FI8
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR1
15	6.28E-09	.01	FIRE IN RELAY	ROOM OCCURS			IEV-FI5
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			CHARGING SYSTEM	FAILS			SYS-CHP
			COMPONENT	COOLING WATER	SYSTEM FAILS		SYS-CCL
16	2.52E-09	.00	FIRE NEAR MCC	62J OCCURS			IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
			BLEED AND FEED	FAILS			SYS-OB2
17	[illegible]E-09	[illegible]	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		IEV-FI3
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			BLEED AND FEED	FAILS			SYS-OB6
18	2.99E-10	[illegible]	FIRE NEAR MCC	62J OCCURS			IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			CHARGING SYSTEM	FAILS			SYS-CHG
			COMPONENT	COOLING WATER	SYSTEM FAILS		SYS-CCL
19	2.81E-11	[illegible]	FIRE NEAR MCC	62A IN B DIESEL	GENERATOR ROOM	OCCURS	IEV-FI2
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE
			AFW SYSTEM	FAILS			SYS-AF3
			MFW SYSTEM	FAILS			SYS-OM2
			HIGH PRESSURE	RECIRCULATION	FAILS		SYS-HR1
20	2.41E-10	.00	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS		IEV-FI3
			EQUIPMENT FAILS	DUE TO FIRE			FIRE-DAMAGE

TABLE 4-15 ***** DOMINANT SEQUENCES ***** Page: 3

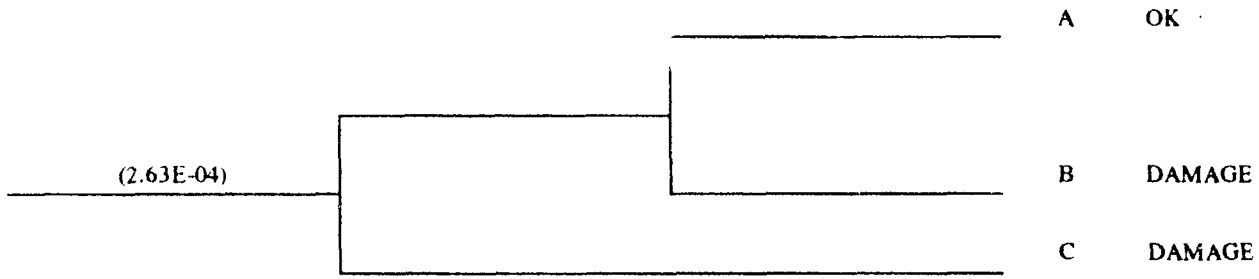
Title: Total Core Melt Due to Fires
File: FIRETOT.LST (File created by linking firetot.in ..LINK ** Ver. 3.11 **)
Reduced Sum of Sequence Probabilities: 9.8116E-05

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION			SEQUENCE IDENTIFIER
			AFW SYSTEM	FAILS		SYS-AF3
			HIGH PRESSURE	RECIRCULATION	FAILS	SYS-HR1
21	1.02E-10	.00	FIRE IN BUS 1	AND BUS 2 ROOM	OCCURS	IEV-FI3
			EQUIPMENT FAILS	DUE TO FIRE		FIRE-DAMAGE
			CHARGING SYSTEM	FAILS		SYS-CHG
			COMPONENT	. OLING WATER	SYSTEM FAILS	SYS-CCT
22	7.05E-13	.00	FIRE NEAR MCC	62J OCCURS		IEV-FI1
			EQUIPMENT FAILS	DUE TO FIRE		FIRE-DAMAGE
			AFW SYSTEM	FAILS		SYS-AF3
			MFW SYSTEM	FAILS		SYS-OM2
			HIGH PRESSURE	RECIRCULATION	FAILS	SYS-HR1

Figure 4-1: AX-23A (FI1) INITIATING EVENT FREQUENCY

AUXILIARY BLDG (MCC 1-62J ROOM, EL 642')

Fire Initiation Frequency (per year)	Automatic Detection	Automatic Suppression	End State	Frequency (per year)



Total Frequency = 2.63E-04/yr

Active Fire Protection = Manual detection and suppression

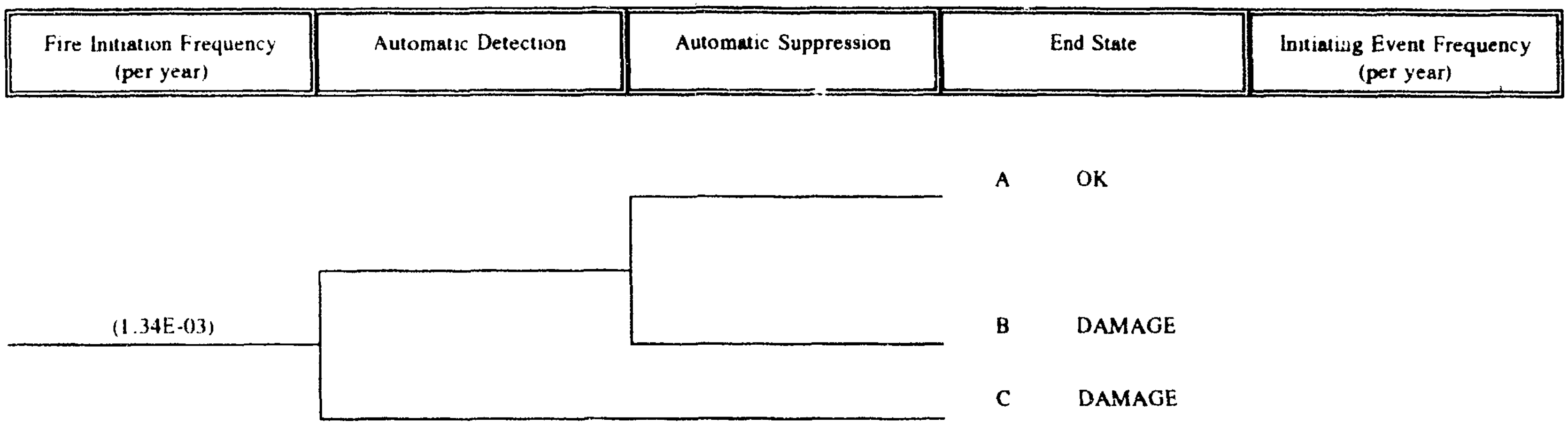
= T(manual suppression - 6.3 min.) is greater than T(damage - 2 min.); therefore, credit cannot be granted for manual fire suppression.

Fire Initiation Frequency = (compartment fire frequency) * (area of influence / room area)
= (2.63E-04) * (1) = 2.63E-04/yr

Applicable Event Sequence = Transient with main feedwater available

Figure 4-2: TU-92 (FI2) INITIATING EVENT FREQUENCY

DIESEL GENERATOR ROOM B (MCC 62A)

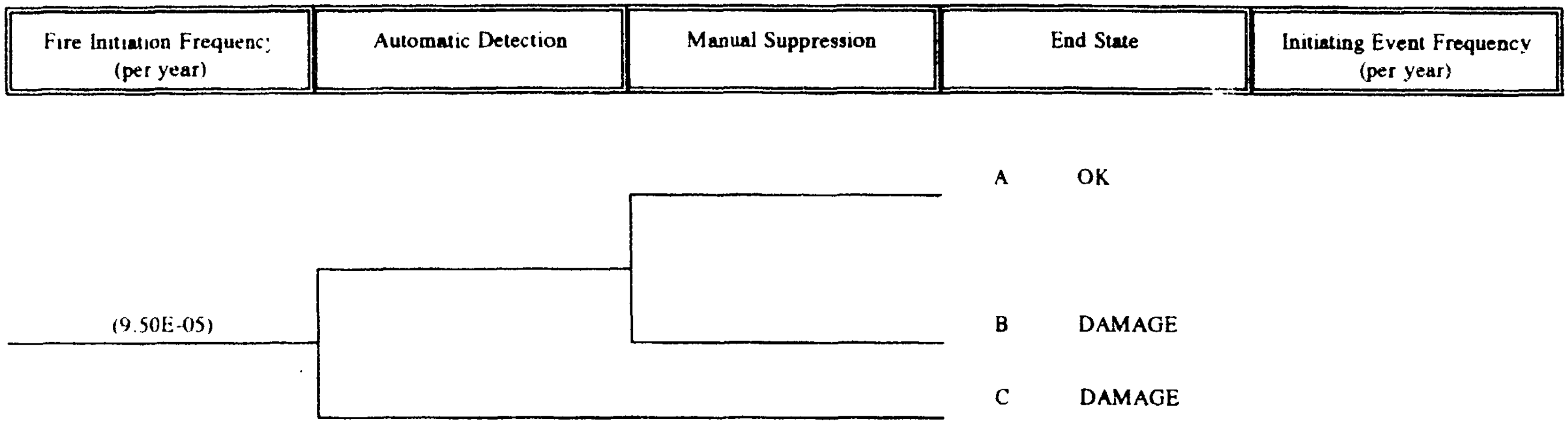


Total Frequency = 1.34E-03/yr

Active Fire Protection	=	automatic detection (fixed temperature - flame detector), automatic suppression (flooding LP CO_2 system) with time delay
Fire Initiation Frequency*	=	(compartment fire frequency) due to MCC 62A + (0.5 * frequency due to transients and ventilation)
	=	(1.084E-03) + (0.5 * 5.04E-04) = 1.34E-03/yr
Automatic Detection Failure	=	detection cannot be credited since t(damage) is less then t(detection + suppression)
Automatic Suppression Failure	=	suppression cannot be credited since t(damage) is less then t(detection + suppression - CO_2 time delay)
Applicable Event Sequence	=	Transient with main feedwater available

* The rest of the compartment fire frequency is considered in FI4.

Figure 4-3[illegible]: AX-21 (FIS) INITIATING EVENT FREQUENCY
SWGR ROOM (ROOM 16B)



Total Frequency = **9.50E-05/yr**

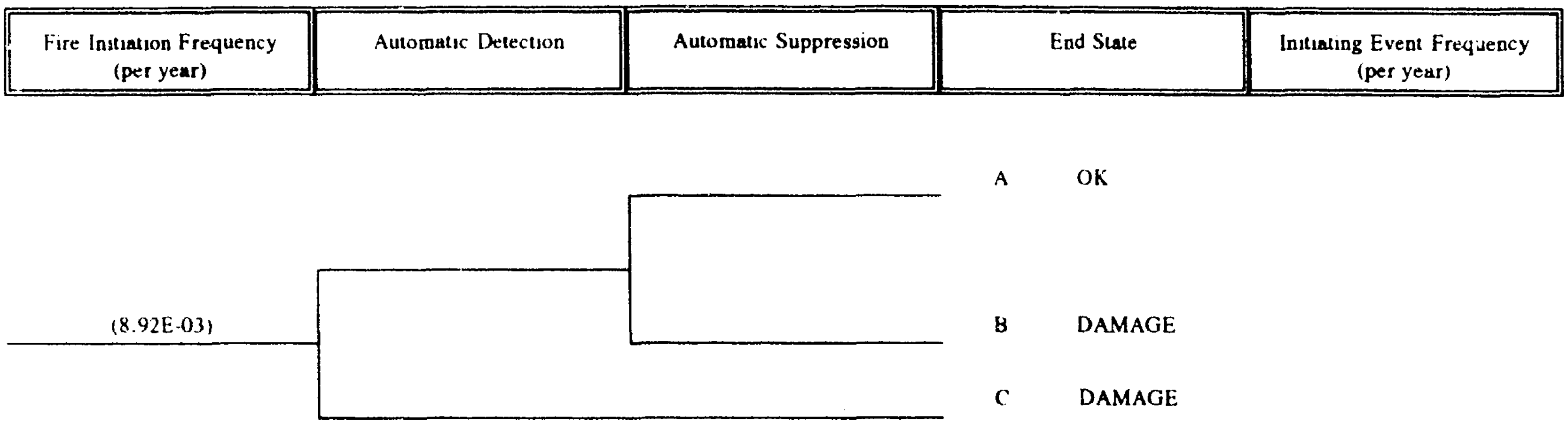
Active Fire Protection = automatic detection (ionization detector), manual suppression (CO_2 extinguisher)

= Credit is not granted for manual fire suppression because T(automatic detection) + T(manual suppression - 6.3 min.) is greater than T(damage - 1 min.); therefore, the total frequency is equal to fire initiating frequency.

Fire Initiation Frequency = (compartment fire frequency) * (area of influence / room area)
= (8.75E-04) * (12.38 m^2 / 114.07 m^2) = 9.50E-05

Applicable Event Sequence = Transient without main feedwater available

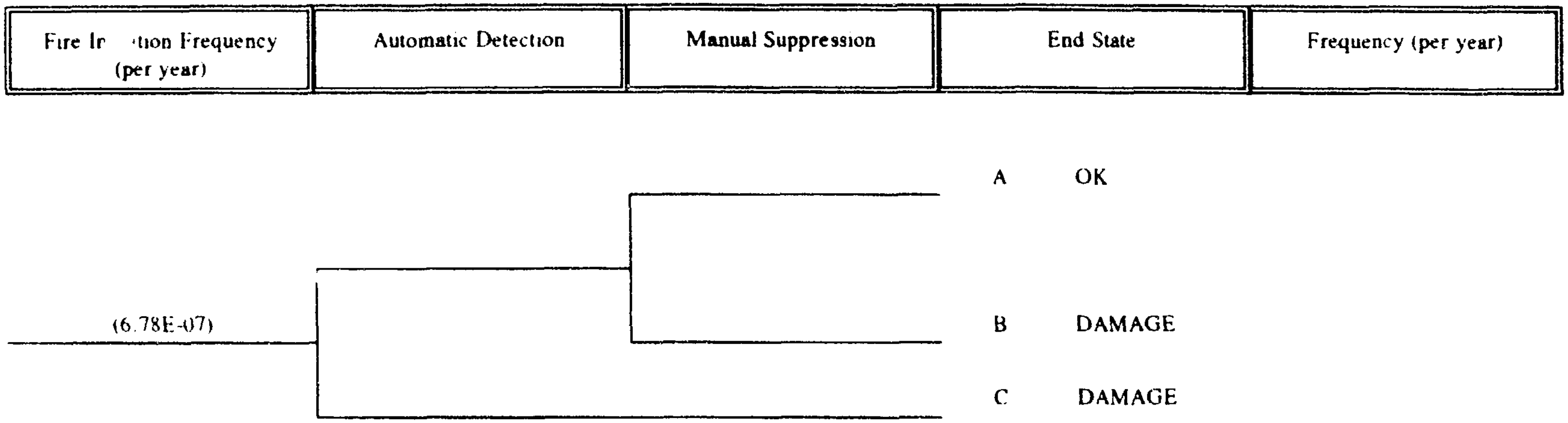
Figure 4-4: TU-92 (FI4) INITIATING EVENT FREQUENCY
DIESEL GENERATOR ROOM B (DIESEL FIRE)



Total Frequency = 8.92E-03/yr

Active Fire Protection	=	automatic detection (fixed temperature - flame detector), automatic suppression (flooding LP CO_2 system) with time delay
Fire Initiation Frequency	=	(compartment fire frequency due to diesel fire) + (0.5 * frequency due to transients and ventilation)
	=	(8.66E-03) + (0.5 * 5.04E-04) = 8.92E-03
Automatic Detection Failure	=	detection cannot be credited since t(damage) is less then t(detection + suppression)
Automatic Suppression Failure	=	suppression cannot be credited since t(damage) is less then t(detection + suppression - CO_2 time delay)
Applicable Event Sequence	=	Loss of offsite power

Figure 4-5: AX-30 (FI5) INITIATING EVENT FREQUENCY
RELAY ROOM, FIRE NEAR VERTICAL CABLE TRAYS (EL 606')



Total Frequency = 6.78E-07/yr

Active Fire Protection = Automatic detection (ionization detectors), manually actuated suppression (total flooding CO_2 system)

= T(manual suppression - 6.0 min.) is greater than T(damage - 1 min.); therefore, credit cannot be granted for fire brigade.

Fire Initiation Frequency* = (compartment fire frequency) * (area of influence / room area)
= (5.20E-05)*(3.53 m^2 / 270.8 m^2) = 6.78E-7/yr

Automatic Detection Failure = detection cannot be credited since t(damage) is less then t(detection + suppression)

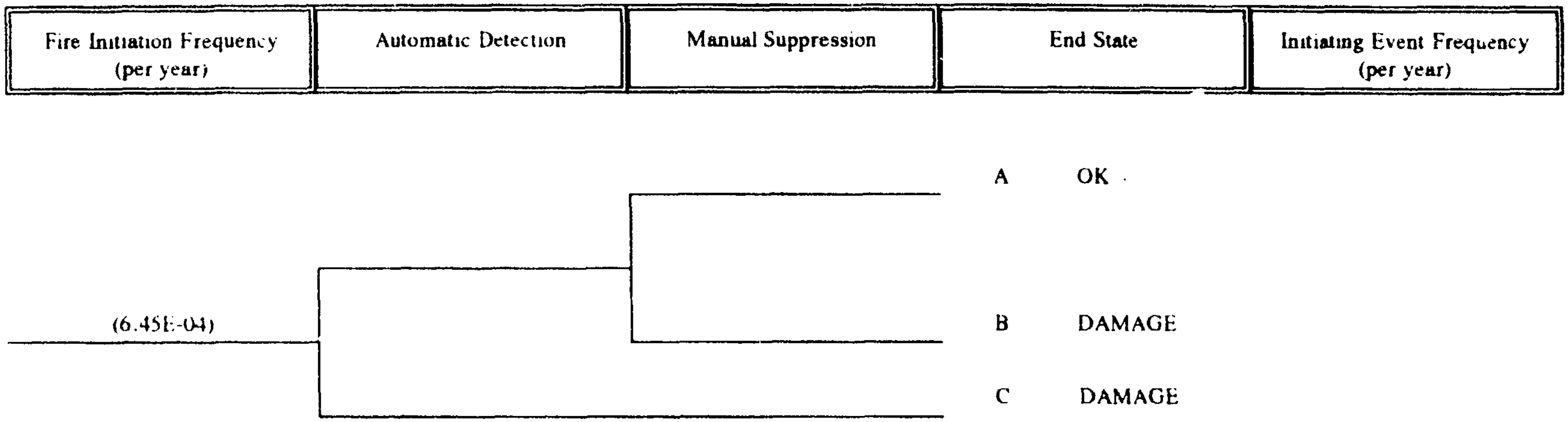
Automatic Suppression Failure = suppression cannot be credited since t(damage) is less then t(detection + suppression)

Applicable Event Sequence = Loss of offsite power

*The Fire initiation frequency given in Table 4-1 is entirely due to electrical cabinets. Since it was shown in Section 4.3.1.9b that fires in the cabinets do not propagate, these are not applicable to this scenario. Instead it was assumed that the fire is from transient combustibles. This is conservative because transient combustibles in the relay room are kept to an absolute minimum. The compartment fire frequency is then:

F(f) * WF(LS)
0.0013 * 1/25 = 5.20E-5

Figure 4-6: TU-95C (FI6) INITIATING EVENT FREQUENCY
TURBINE BLDG (AFW PUMP A ROOM)

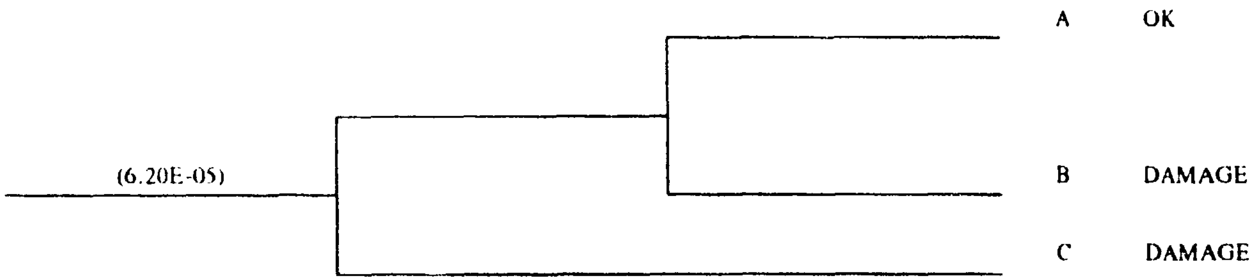


Total Frequency = 6.45E-04/yr

Active Fire Protection	=	automatic detection (ionization detector), manual suppression
	=	Credit is not granted for manual fire suppression because T(automatic detection) + T(manual suppression - 6.3 min.) is greater than T(damage - 1 min.); therefore, the total frequency is equal to the fire initiating frequency.
Fire Initiation Frequency	=	(compartment fire frequency) * (area of influence / room area) * (probability of fire damage)
	=	(1.24E-03) * (1) * (0.52) = 6.45E-04/yr
Applicable Event Sequence	=	Loss of offsite power

Figure 4-7: TU-95B (FI7) INITIATING EVENT FREQUENCY
SWGR ROOM (AFW PUMP B ROOM)

Fire Initiation Frequency (per year)	Automatic Detection	Manual Suppression	End State	Initiating Event Frequency (per year)



Total Frequency = 6.20E-05/yr

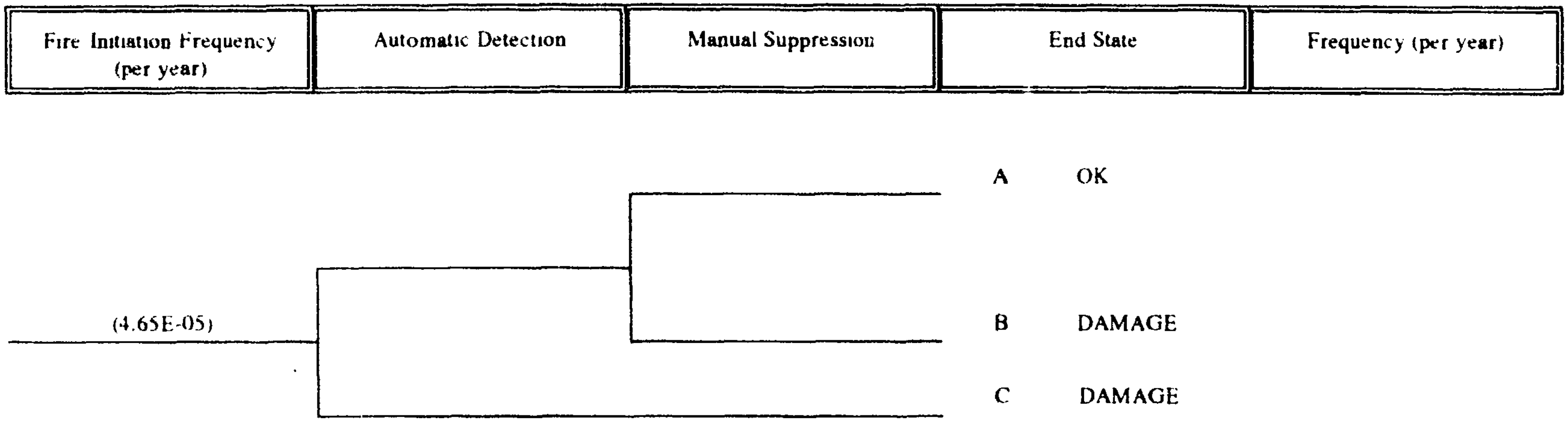
Active Fire Protection = automatic detection (ionization detector), manual suppression (CO_2 hose station)

= Credit is not granted for manual fire suppression because T(automatic detection) + T(manual suppression - 6.3 min.) is greater than T(damage - 3 min.); therefore, the total frequency is equal to fire initiating frequency.

Fire Initiation Frequency = (compartment fire frequency) * (area of influence / room area) * (probability of fire damage)
= (1.24E-03) * (1) * (0.05) = 6.20E-05/yr

Applicable Event Sequence = Loss of offsite power

Figure 4-8: TU-95A (FI8) INITIATING EVENT FREQUENCY
SWGR DEDICATED SHUTDOWN PANEL ROOM



Total Frequency = 4.65E-05/yr

Active Fire Protection = Automatic detection (ionization detector), manual suppression (fire hose station)

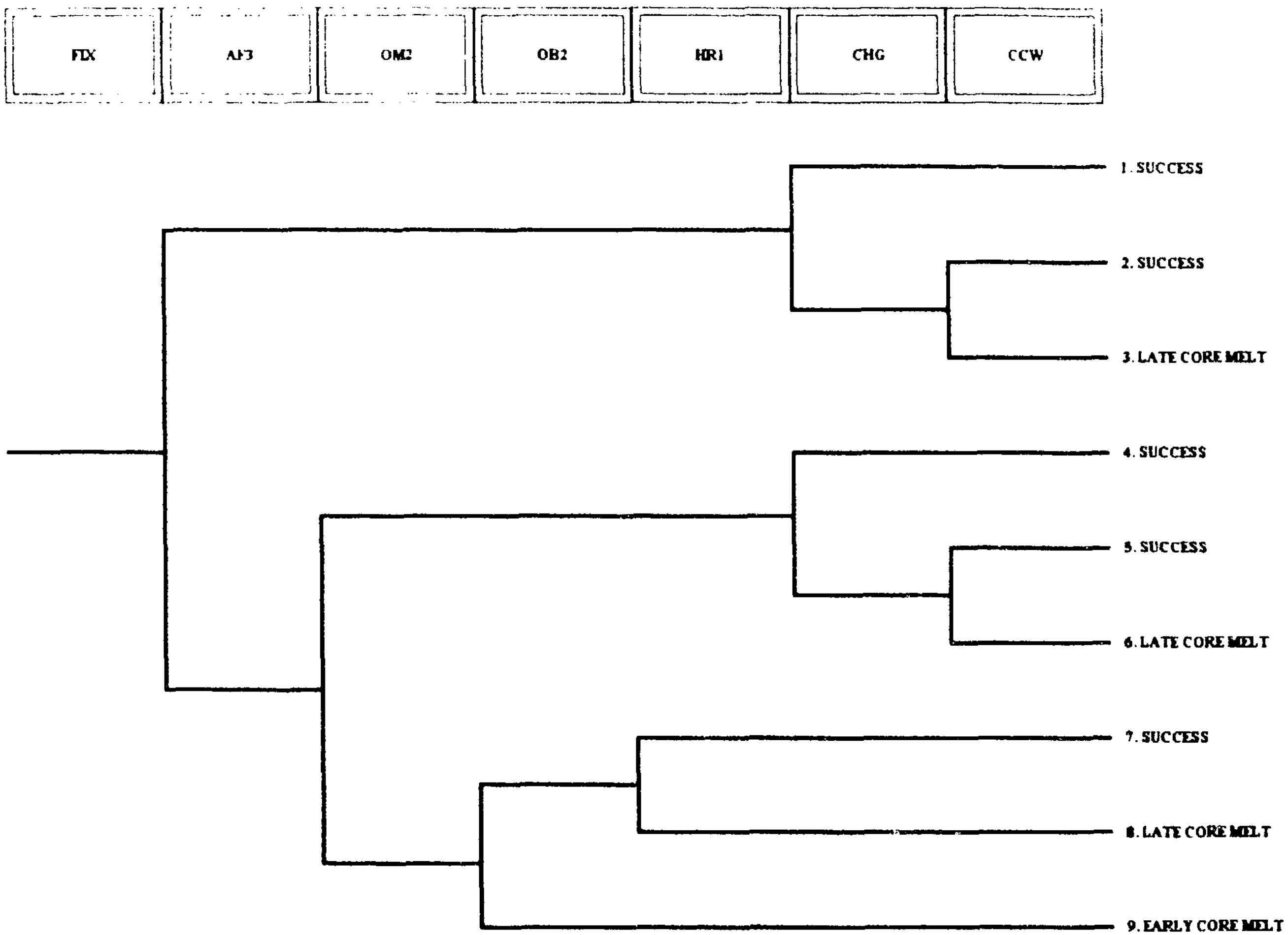
= Credit is not granted for manual fire suppression because T(automatic detection) + T(manual suppression - 6.3 min.) is greater than T(damage - 2 min.); therefore, the total frequency is equal to the fire initiating frequency.

Fire Initiation Frequency = (Compartment Fire Frequency) * (Area of Influence / Room Area)
= (1.764E-03) * (2.6 m^2 / 98.65 m^2) = 4.65E-05/yr

Applicable Event Sequence = Loss of offsite power

FIGURE 4.9

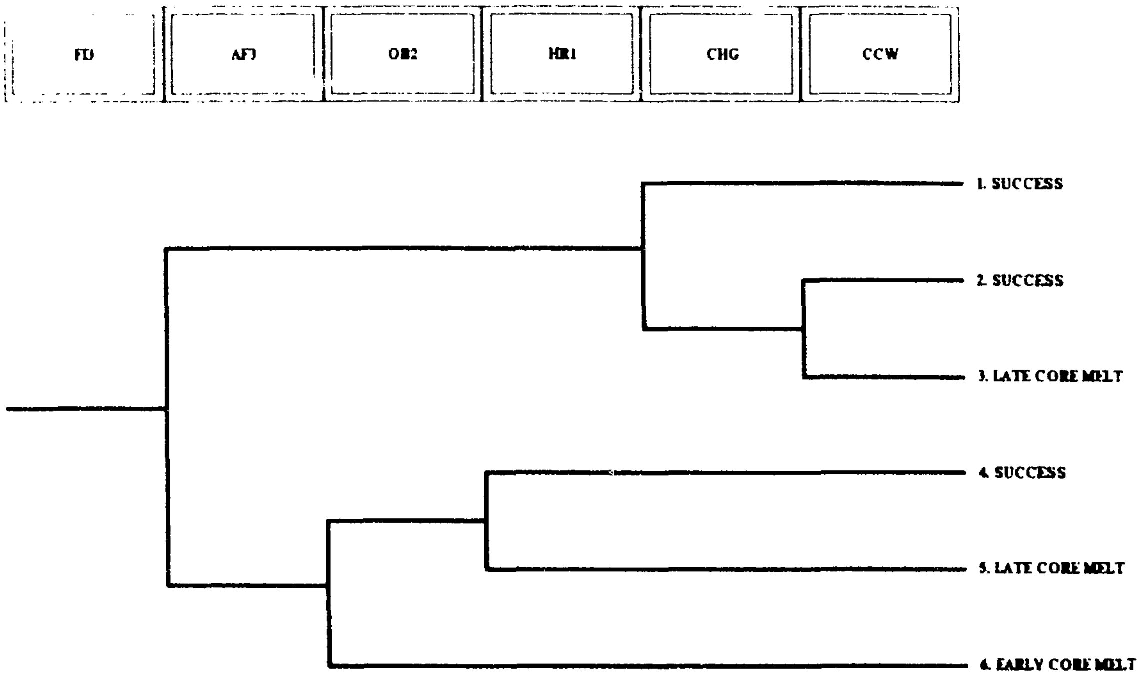
TRANSIENTS WITH MAIN FEEDWATER AVAILABLE EVENT TREE



THIS EVENT TREE IS USED FOR FIRES FI1 AND FI2

FIGURE 4-10

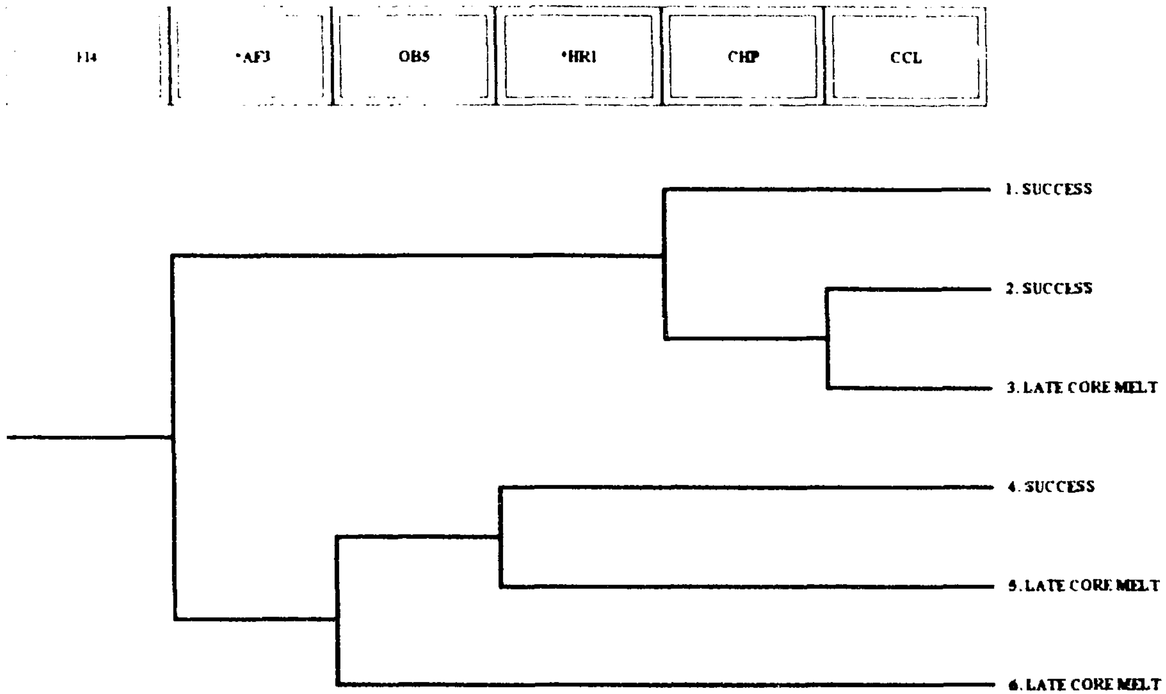
TRANSIENTS WITHOUT MAIN FEEDWATER EVENT TREE



THIS EVENT TREE IS USED FOR FIRE FD

FIGURE 4 11

LOSS OF OFFSITE POWER EVENT TREE

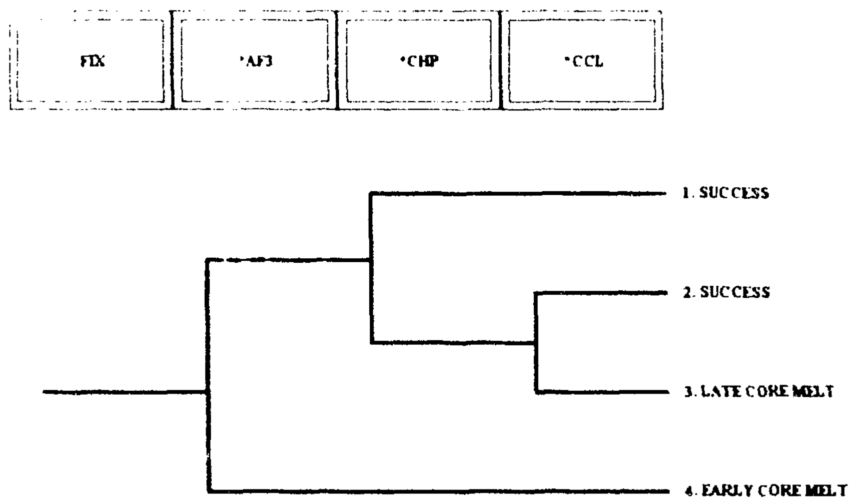


THIS EVENT TREE IS USED FOR FIRE FI4

* THESE NODES ARE USED IN OTHER EVENT TREES, THEY ARE CONDITIONAL HERE DUE TO THE INITIATOR.

FIGURE 4-12

DEDICATED SHUTDOWN SYSTEM EVENT TREE

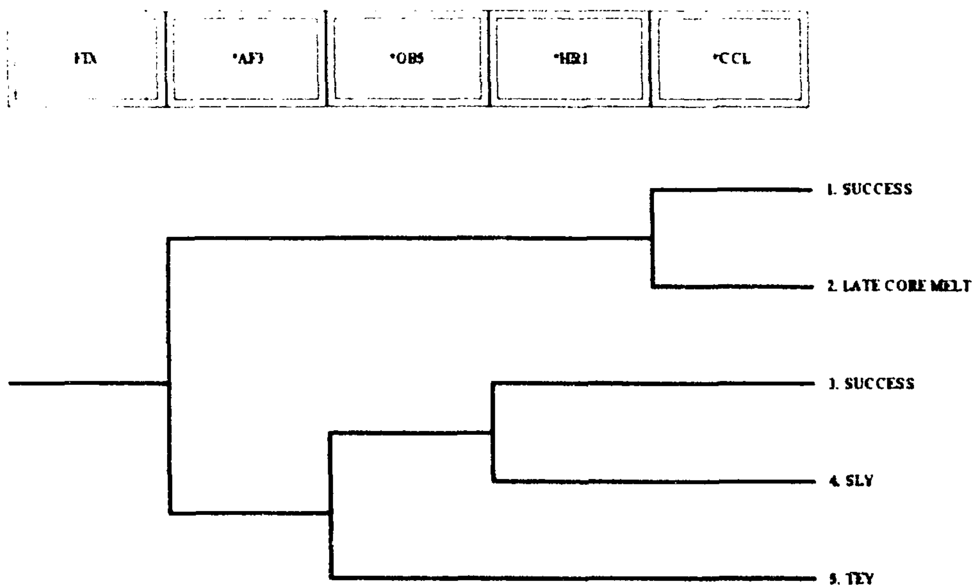


THIS EVENT TREE IS USED FOR FIRES FI5 AND FI7

*THESE NODES ARE USED IN OTHER EVENT TREES. THEY ARE CONDITIONAL HERE DUE TO THE INITIATOR.

FIGURE 4.13

ALTERNATE SHUTDOWN SYSTEM EVENT TREE



THIS EVENT TREE IS USED FOR FIRES F16 AND F18

*THESE NODES ARE USED IN OTHER EVENT TREES, THEY ARE CONDITIONAL HERE DUE TO THE INITIATOR.

5. Other External Events Analysis

SECTION 5

OTHER EXTERNAL EVENTS ANALYSIS

TABLE OF CONTENTS

SECTION **PAGE**

SECTION 5

OTHER EXTERNAL EVENTS ANALYSIS

LIST OF TABLES

SECTION 5

OTHER EXTERNAL EVENTS ANALYSIS

TABLE OF FIGURES

SECTION **PAGE**

SECTION 5

OTHER EXTERNAL EVENTS ANALYSIS

TABLE OF APPENDICES

5.0 High Winds, Floods, and Others

5.0.1 Introduction

The external events to be considered are those events whose cause is external to all systems associated with normal and emergency operations situations. A comprehensive list of external events is provided in NUREG/CR-2300, "PRA Procedures Guide" (Reference 1). Some external events listed may not pose a significant threat of a severe accident. Some external events were considered in the design of the plant and therefore have sufficiently low contribution to core damage frequency or plant risk. Some events were reviewed under ongoing programs; for instance under the individual plant examination (IPE), the significance of lightning and severe cold weather conditions that could cause loss of off-site power was assessed. Also, internal floods were included in the IPE request. The purpose of this study is to ensure that all potential external hazards are analyzed and pose no undue risk to the plant.

5.0.2 General Methodology

NUREG-1407 presents an acceptable methodology to be followed for the analysis of the "other external events." The three major steps of the general methodology used at the Kewaunee Nuclear Power Plant are:

- Establishing a List of Plant Specific Other External Events
- Progressive Screening
- Documentation

The following is a detailed description of the general methodology.

<u>Establishing a List of Plant Specific Other External Events</u>

The first step in the individual plant examination for external events (IPEEE) for "other external events" is to delineate all the events to be analyzed. In the guidelines provided in Generic Letter 88-20 (Reference 2), Supplement 4 (Reference 3), NUREG-1407 (Reference 4), NUREG/CR-2300 and NUREG/CR-5042 (Reference 5) are to be followed in developing the list of plant specific "other external events" for IPEEE analysis. The list of events for Kewaunee are presented on Table 5.1 and is discussed further in Section 5.0.4.

<u>Progressive Screening</u>

The overall approach consists of a progressive screening. The screening criterion for reporting potential severe accident sequences is consistent with that used for the IPE. The steps in the progressive screening approach represent a series of analyses in increasing levels of detail, effort, and resolution. One or more steps may be bypassed, however, as long as subsequent steps identify the vulnerabilities or demonstrate that they are insignificant. The screening approach consists of the following steps as presented in Figure 5-1.

1. Review plant-specific hazard data and licensing bases.

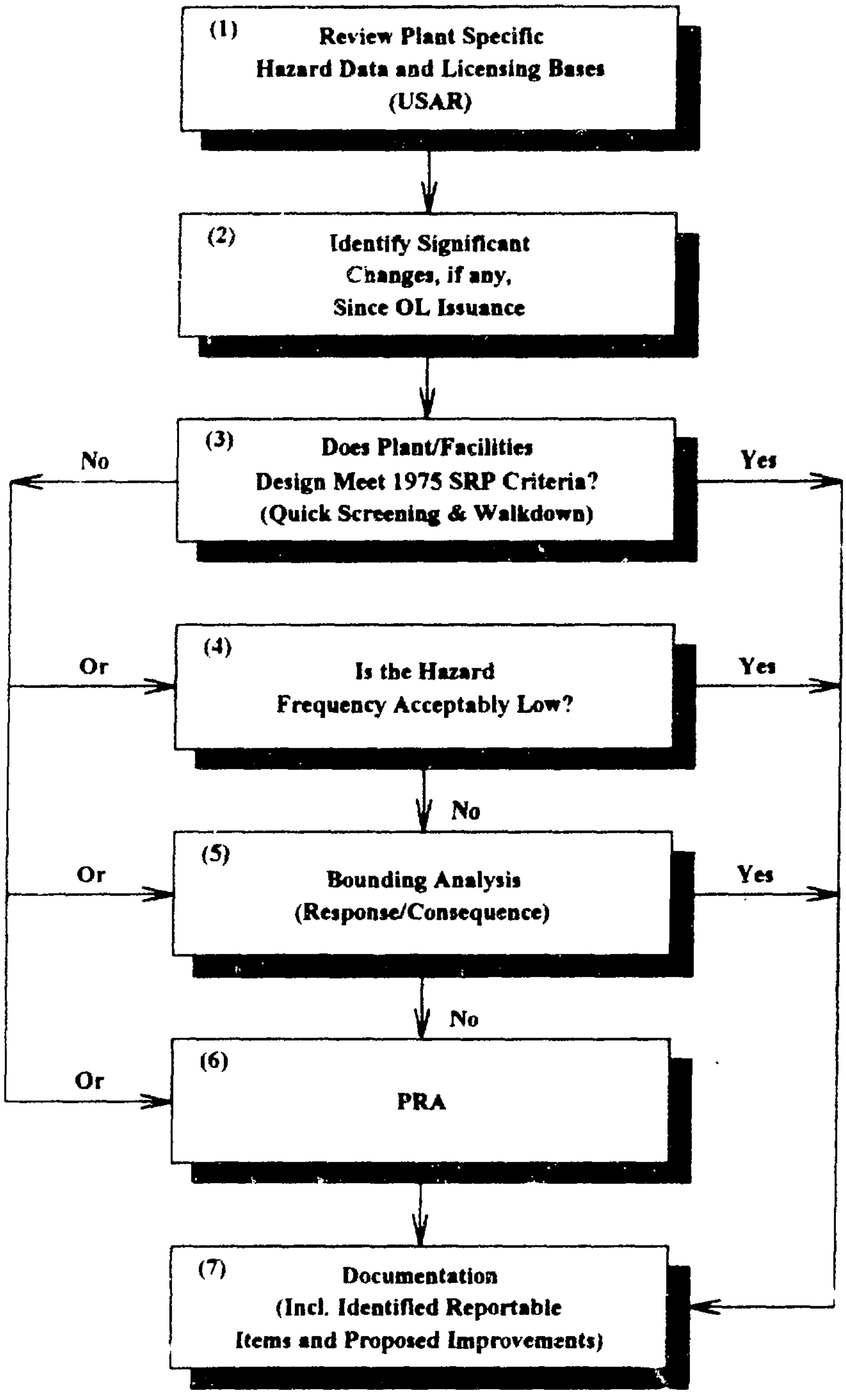
2. Identify significant changes since the operating license (OL) was issued including:

(1) military and industrial facilities within 5 miles of the site, (2) on-site storage or other activities involving hazardous materials, (3) transportation, or (4) developments that could affect the original design conditions.

3. Determine if the plant and facilities design meets the 1975 Standard Review Plan (SRP, Reference 6) criteria.

After reviewing the information obtained in the previous two steps a confirmatory walkdown of the plant is performed. The walkdown concentrates on outdoor facilities that could be affected by high winds, on-site storage of hazardous materials, and off-site developments. If the walkdown reveals that potential vulnerabilities have not changed since the original design basis analysis, it is judged that the continuation from the hazard to core damage frequency is less than 1E-06 per year (NUREG-1407) and the IPEEE screening criterion is met.

Figure 5-1: IPEEE Approach for Winds, Floods, and Other External Events



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4. Determine if the hazard frequency is acceptably low.

   If the original design basis does not meet current regulatory requirements, the next step is to demonstrate that the original design basis is sufficiently low - that is, less than 1E-05 per year, and the conditional core damage frequency is judged to be less than 1E-01.

   If the original design basis hazard combined with the conditional core damage frequency is not sufficiently low (i.e., less than the NUREG-1407 screening criterion of 1E-06 per year), additional analysis is needed.

5. Perform a bounding analysis.

   This analysis is intended to provide a conservative calculation showing that either the hazard does not result in core damage or the core damage frequency is below the reporting criterion. The level of detail is that level needed to demonstrate the point; judgement is needed for determining the proper level of detail and needed effort.

6. Perform a probabilistic risk assessment (PRA).

   A probabilistic risk assessment consists of the following key elements: hazard analysis, fragility evaluation, plant systems and accident analysis (event/fault trees), and radioactive material release analysis. The detailed procedure is described in NUREG/CR-2300. If the core damage frequency is less than 1E-06 per year, the event need not be considered further. The level of detail is that level needed to conclude that the core damage frequency is low or to find vulnerabilities.

The application of the above approach involves considerable judgment with regard to the scope and depth of the study, level of analytical sophistication, and level of effort to be expended. This judgment depends on how important the external initiators are when compared with internal initiators, and a perceived need for accurately characterizing plant capacity or core damage frequency. The detailed guidelines presented in NUREG-1407 do not preclude use of this type of judgment. Consistent with engineering practice, either expert opinions, simplified scoping studies, or bounding analyses are used, as appropriate, in forming these judgments.

Documentation

The IPEEE is documented in a traceable manner using a two-tier approach to provide the basis for the findings. The first tier consists of the results of the examination and the second tier is the documentation of the examination itself. The following information on the high winds, floods, and other portions of the IPEEE is documented in accordance with the guidance provided in NUREG-1407.

1. A description of the methodologies used in the examination.

2. Information on plant-specific hazard data and licensing bases.

3. Results of the assessment of the hazard frequency and the associated conditional core damage frequency if step 4 of Figure 5-1 is used.

4. Results of the bounding analysis if step 5 of Figure 5-1 is used.

5. If step 6 of Figure 5-1 (PRA) is required, the following is also provided:

   - All systemic event trees, including origin and method of analysis.
   - A description of each systemic sequence selected, including discussion of specific assumptions and human recovery action.
   - The estimated core damage frequency, the timing of the associated core damage, a list of analytical assumptions including their bases, and the sources of uncertainties, if applicable.

5.0.3 Methodologies for Specific Events

The approach for performing IPEEE analysis for a select group of "other" external events related to Kewaunee is presented in the following discussion.

A. High Winds and Tornadoes

For plants designed against NRC's current criteria, high winds pose no significant threat of a severe accident because the current design criteria for wind are dominated by tornadoes having a 1E-07 annual frequency of exceedance. However, Kewaunee has facilities that were designed and built prior to the NRC's current criteria and therefore a systematic examination to identify plant-specific vulnerabilities according to NUREG/CR-5042 (Reference 7) is required.

The purpose of this analysis is to determine the impact on safe plant operation due to the occurrence of high winds at the plant site. This analysis first estimates the expected frequency of exceedance of various wind speeds, and then assesses the likelihood of damage to specific plant structures and components as a result of stresses induced by those winds. The contribution of wind to plant risk depends on structures at the site susceptible to wind damage. Prior PRA experience indicates that, except in special circumstances, wind is generally not a major contributor to risk.

The specific tasks for this evaluation may include the following depending upon how the screening process proceeds.

- Wind Frequency Analysis
- Fragility Analyses
- Plant/Systems Analysis
- Core Damage Quantification

Wind Frequency Analysis

A scoping assessment of wind speeds is derived by examining wind speed distribution data from such sources as the Kewaunee Updated Safety Analysis Report (USAR) (Reference 8) and the National Weather Service. These sources provide sufficient data for use in the scoping fragility analysis task.

If this analysis indicates that wind may be a significant contributor to the core damage frequency, a more detailed wind speed frequency of exceedance curve would be developed for the site. This curve would indicate, for a given wind speed, the frequency with which winds of an equal or higher speed will occur. This curve would be based primarily on data collected over the past five years obtained from the site meteorological tower. Data on maximum historical wind speeds is obtained from National Weather Service data for locations near the site.

Fragility Analysis

The fragility analysis estimates the conditional probability that specific plant components or structures will fail when subjected to winds of a given intensity. A scoping analysis is performed to identify those components and structures whose failure could have a significant impact on core damage frequency. For the initial screening, conservative assumptions are made as to the likelihood of failure of various structures subjected to the scoping wind speed/frequency data. If necessary for more detailed analysis, estimates are made for the key items of the wind loading at which failure can be expected to occur.

Plant/Systems Analysis

The plant/systems analysis consists of developing logic models (e.g., event or fault trees) to determine the impact of wind speeds of a given intensity distribution. System availabilities are calculated by taking into account system component failures or fragilities as derived in the fragility analysis task.

Core Damage Quantification

The results of the wind analysis and plant systems analysis are combined to determine the frequency of degraded core states as a result of wind. The core damage frequency is then calculated as in the IPE. If, based on the screening analysis, the core damage frequency due to wind is significant with respect to that due to other external events, a more detailed wind analysis is required.

B. External Flood

For plants designed in accordance with Regulatory Guide 1.59 and applicable SRP sections (particularly Section 2.4), floods pose no significant threat of a severe accident. This is because the exceedance frequency of the design basis flood, (excluding floods due to failure of upstream dams) being judged to be less than 1E-05 per year, and the conditional core damage frequency for a design basis flood being judged less than 1E-01. Thus, core damage frequencies are estimated to be less than 1E-06 per year for a plant designed against NRC's current criteria. However, the latest probable maximum precipitation (PMP) criteria published by the National Weather Service (NWS) calls for higher rainfall intensities over shorter time intervals and smaller areas than have previously been considered. This change could result in higher site flooding levels than have been used in previous design bases. The effects of applying these new criteria in terms of on-site flooding is assessed. Also, systematic examination for plant-specific vulnerabilities are performed if needed.

An external flooding analysis has been performed to account for any significant contribution to core damage frequency resulting from plant damage caused by storms, seiche, dam failure, flash floods, etc., is accounted for.

The basic steps involved in an external flooding analysis are similar to those described for internal flooding. However, the focus of attention is on areas that, due to their location and grading, may be susceptible to external flood damage. Thus, information on such items as dikes, surface grading, locations of structures, and locations of equipment within the structures is required. Information such as meteorological data for the site, historical flood height and frequency data is needed. For the scoping analysis, data available from the USAR and from PRAs for plants in similar areas are used to estimate the frequency and severity of external floods. Once the data is collected, a compilation of potential flooding sources and a matrix of flooding areas versus critical components is developed.

Flooding areas are then evaluated for adequacy of flood protection. The consequences of postulated flooding in those areas are determined, in order to identify those areas in which external flooding could lead to initiating events, loss of critical system functions, etc. Electrical power supply equipment, and in particular, off-site power supplies, receive special attention.

Finally, if required, the probabilities of flood-induced initiating events are determined using event trees to account for mitigation and detection actions. These probabilities are then combined with the estimated external flooding frequency for each potential flood area. The core damage frequency is then calculated in a manner similar to that described for the internal flooding analysis

C. Transportation and Nearby Facility Accidents

These events consist of accidents related to either transportation accidents or accidents at industrial and military facilities. Plants designed against NRC's current criteria (NUREG/CR-5042) should have no significant vulnerability to severe accidents from these events because the initiators considered in the design should have a recurrence frequency less than 1E-06 or have been shown through a bounding analysis not to affect the plant. Changes may have occurred since the original design, however, so there may be exceptions that need systematic examination. Since Kewaunee was designed and built prior to the NRC's current criteria, a systematic examination for plant-specific vulnerabilities is required.

Any transportation of hazardous materials or nearby facility accident has associated with it the potential for hazards. Such hazards can include the accidental release of toxic material, explosion, fire, and so forth. If such a hazard occurs near the plant site, the possibility exists that sufficient damage could occur to the plant, and thus lead to scenarios that result in core damage.

The hazardous material transportation analysis first identifies the types and frequencies of such shipments in the vicinity of the plant. This information is obtained from a variety of sources, including the U.S. Coast Guard and Department of Transportation. Next, an evaluation is made of the types of hazards that could occur near the plant, and then an estimated frequency of occurrence is assigned. Records from sources such as those listed above are used to assess these hazard frequencies. The generic data is then updated with any available local data. For each type of hazard identified, the consequence at the site boundary is calculated based on the type of material involved, the anticipated quantities being shipped, and so forth, using appropriate simplified dispersion models and atmospheric data.

Finally, an assessment is made of the vulnerability of the plant (i.e., the probability of reaching a core damage state) to the consequences of any such event. As a first approximation, an assignment of plant system or structural damage is made, based on the type and magnitude of the postulated event. Then, if such events are determined to be significant contributors to core damage frequency, a more detailed analysis (e.g., a consequence-specific fragility analysis) is performed to determine the true contribution. The internal event plant logic is modified as appropriate for use with the external hazards analysis, and then combined with the vulnerability assessment to determine the conditional annual frequency of each core damage state resulting from hazardous shipping.

The potential for hazards resulting from transportation incidents other than those described above are also evaluated. Truck or rail accidents involving hazardous materials near the plant site can pose a threat to safe operation of the plant as well. The magnitude of risk from such incidents depends on the types and frequencies of shipments, proximity to the plant, and so forth. The end result is an estimate of the annual frequency of core damage due to other transportation hazards. Nearby facility accidents are similarly analyzed.

D. Hazardous Material

These events consist of accidents related to the release of hazardous materials on site. Such hazards would include the release of any toxic material located in or near the plant. If such a hazard occurs, the possibility exists that sufficient damage could occur to the plant or creates an environment which prevents the safe operation of plant equipment, and thus increases the probability of scenarios that could result in core damage.

This analysis first identified the types and quantities of such materials located in or near the plant. This information is obtained from a variety of sources, including current programs involving hazardous chemical control, and plant walkdowns. Next an evaluation is made of the various hazards to determine their effect on plant equipment or personnel.

Finally, an assessment is made of the vulnerability of the plant (i.e, increased probability of core damage) to the consequences of any such event.

5.0.4 Determination of Need to Analyze Additional Events

In order to determine if any other external event needs to be analyzed, a set of screening criteria is used. This is done to identify those external hazards that can be screened from further consideration based on the screening criteria outlined in NUREG/CR-2300.

An external event is to be excluded from further consideration if:

Criterion 1 The event is of equal or lesser damage potential than the events for which the plant is designed. This requires an evaluation of the plant design bases in order to estimate the resistance of plant structures and systems to a particular external event.

Criterion 2 The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and cannot result in worse consequences than those events.

Criterion 3 The event can not occur close enough to the plant to affect it. This is also a function of the magnitude of the event.

Criterion 4 The event is included in the definition of another event.

Criterion 5 The event is slow in developing and there is sufficient time to eliminate the source of the threat or to provide an adequate response.

The use of these criteria minimizes the possibility of omitting any significant risk contributors while at the same time reducing the amount of detailed bounding analysis required. A summary of the screening analysis based on the Kewaunee USAR is included in Table 5-1.

Table 5-1: Screening of External Events for KNPP

Event	Screening Criterion	Remarks
Aircraft Impact	–	A bounding analysis is performed for this event.
Avalanche	3	Topography is such that no avalanche is possible.
Biological Events	5	The only biological event that may affect Kewaunee is zebra mussel blockage of the Circulating Water System intakes. This event is not further considered because flow blockage would be a gradual event to which the plant could easily respond. This is considered in the loss of service water event in the IPE.
Coastal Erosion	5	The shoreline at Kewaunee is not subject to rapid coastal erosion due to the extensive application of rip-rap. The long-time periods required to produce sufficient coastal erosion to endanger the plant preclude further analysis.
Drought	2	The depth of the intake cribs at Kewaunee (about 10 feet below the record low lake level) precludes further analysis.
External Flooding	–	A bounding analysis is performed for this event.
Extreme Winds/Tornados	–	A bounding analysis is performed for this event.

Event	Screening Criterion	Remarks
Fog	4	Fog can increase the frequency of occurrence of accidents. Fog is implicitly included in aircraft and other transportation accident frequency data. Fog, in combination with substation component failures, could result in off-site power line outages. This is considered in the IPE study.
Frost	1	Snow and ice govern.
Hail	1	Tornado and turbine missiles govern.
Nearby Facility Accidents	–	A bounding analysis is performed
High Lake Level	4	Included under external flooding.
High Summer Temperature	1	The main problem would probably be loss of heat sink, which is included in drought.
Hurricane	4	Included under high winds.
Ice Cover	1	Ice or snow loading is considered in the plant design.
Industrial or Military Facility Accidents	4	Included under hazardous materials.
Internal Fire	–	A detailed plant specific analysis is performed in Section 4 of this report.
Internal Flooding	–	A detailed plant specific analysis was performed in the IPE.
Landslide	3	Topography is such that a landslide is not possible.
Lightning	4	Included in loss of off-site power initiating event frequency and external fire analysis.
Low Lake Level	4	Included under drought.

Event	Screening Criterion	Remarks
Low Winter Temperature	1	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design; there would also be ample warning time with respect to freezing of the heat sink.
Meteorite	2	Extremely low frequency according to K. A. Soloman, et. al., "Estimate of the Hazards to a Nuclear Reactor from the Random Impact of Meteorites," UCLA-ENG-7426, March 1974.
Pipeline Accident	4	Included under hazardous materials.
Intense Precipitation	4	Included under external flooding.
River Diversion	4	Included under external flooding; also heat sink is Lake Michigan.
RWST/CST Rupture	2	The likelihood of such a random failure is extremely low and inlcuded in the IPE. Seismically induced failure of these tanks is analyzed in Section 3 of this report.
Sandstorm	3	This is not relevant for this region.
Seiche	4	Included under external flooding.
Seismic Activity	–	A detailed site and plant specific analysis is performed in Section 3 of this report.
Snow	1	Ice or snow loading is considered in the plant design.
Soil Shrink-Swell Consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
Storm Surge	4	Included under external flooding.
Transportation Accidents	-	Included under transportation and nearby facilities.

Event	Screening Criterion	Remarks
Tsunami	3	This is not relevant for this region.
Volcanic Activity	3	This is not relevant for this region.
Waves	4	Included under external flooding.
Hazardous Materials	-	Control room habitability study includes an evaluation of hazardous materials. A review is performed to ensure that no changes have occurred to invalidate the results of this study.

5.1 High Winds and Tornadoes

This section examines potential severe wind events which might initiate an accident sequence leading to core damage. Potential severe wind sources include: high winds, tornadoes, and hurricanes.

5.1.1 Severe Wind Source Screening

Hurricane wind speeds tend to diminish as they pass over large land masses. According to the Kewaunee USAR, due to the geographic location of Kewaunee, the probability of core damage as a direct result of wind from a hurricane is negligible.

5.1.2 Screening Summary

The only severe wind sources that could potentially impact Kewaunee are strong winds and tornadoes.

5.1.3 Analysis

Severe winds can affect safety related structures at the plant site in at least two ways. If the wind forces exceed the load capacity of a building or other external facility, the incident walls or framing may collapse or the structure may overturn. If strong enough, the winds may be capable of lifting objects and hurling them against some of these structures. If a wind-induced missile breaches a building wall, critical components or other equipment inside the building may be damaged or disabled.

The first step in a severe wind analysis is to determine the frequency of the wind speeds associated with the various wind sources for the area in question. If the hazard frequency is acceptably low (≤ 1E-06), no further analysis is needed. Otherwise, the allowable wind loading of each pertinent plant structure is coupled with the wind speed frequency data to determine the expected failure frequency of the structure and ultimately the core damage frequency.

A. High Wind Analysis

The site-specific data is obtained from NUREG/CR-2890 (Reference 9) for the years 1887 through 1979. Data for the years 1980 through 1988 is obtained from the National Oceanic and Atmospheric Administration (NOAA). Appendix A provides the annual extreme wind data collected for the years 1887-1988. All the data is obtained from the weather station at Green Bay, Wisconsin, which is approximately 30 miles from the Kewaunee site. The following justification is provided in NUREG/CR-4458 (Reference 10) Appendix G for using the Green Bay weather station data. NUREG/CR-4458 was developed for the Point Beach Nuclear Plant which is located three miles from the Kewaunee site (Reference 11 and 12).

> The possible uncertainty due to the proximity of the weather station to the site and the differences in terrain roughness between the site and station were considered. If a station is located far from a site, then other adjacent stations should also be included in the analysis. In addition, if the terrain roughness is different, then a wind speed correction factor, which is uncertain, should be included.
>
> For Point Beach, the closest weather station is only 30 miles away and the terrain in between is essentially flat. Thus, no uncertainty for station distance or differences in terrain roughness were considered.

Since the Kewaunee site is closer to Green Bay than Point Beach is, and the terrain between Kewaunee and Green Bay is essentially flat, the Green Bay weather station data is also applicable to the Kewaunee site.

In NUREG/CR-4458, a family of mean wind hazard curves for straight winds were developed using the data base from NUREG/CR-2890. The "Straight Wind Hazard Data" from NUREG/CR-4458 is presented in Table 5-2.

Table 5-2: Straight Wind Hazard Data

Windspeed (mph)	Mean Value*	Frequency of Exceedance Per Year / Probability Value 0.20	0.20	0.20	0.20	0.20
95	1.00E-02	5.67E-04	3.53E-03	7.37E-03	1.29E-02	2.56E-02
101	5.00E-03	1.79E-04	1.45E-03	3.34E-03	6.33E-03	1.37E-02
110	2.00E-03	3.64E-05	4.27E-04	1.14E-03	2.41E-03	5.99E-03
116	1.00E-03	1.15E-05	1.73E-04	4.08E-04	1.16E-03	3.15E-03
123	5.00E-04	3.35E-06	6.68E-05	2.20E-04	5.50E-04	1.66E-03
132	2.00E-04	6.62E-07	1.91E-05	7.29E-05	2.04E-04	7.04E-04
138	1.00E-04	2.06E-07	7.58E-06	3.20E-05	9.61E-05	3.64E-04
153	2.00E-05	1.24E-08	8.42E-07	4.53E-06	1.65E-05	7.82E-05
160	1.00E-05	3.51E-09	3.18E-07	1.92E-06	7.60E-06	4.02E-05
175	2.00E-06	2.06E-10	3.45E-08	2.66E-07	1.27E-06	8.43E-06
182	1.00E-06	5.79E-11	1.29E-08	1.12E-07	5.83E-07	4.29E-06

* Equal to US NRC mean values

Based on the data of Table 5-2, the frequency of exceedance of a wind speed of 182 MPH is 1E-06 per year. Hence, the frequency of exceedance per year for the design wind speed of 300 MPH, from the Kewaunee USAR, is less than 1E-06. Thus, based on step 4 of Figure 5-1, the hazard frequency is acceptably low (≤ 1E-06) and as a result, the high winds do not cause any significant threat to the Kewaunee site.

B. Tornado Analysis

According to the Kewaunee USAR, Wisconsin lies to the northeast of the principal tornado belt in the United States. During the ten-year period 1960-1969, 161 tornadoes were reported in the state. Only six of these tornadoes occurred in the four counties closest to the Kewaunee site (Brown, Door, Kewaunee, and Manitowoc Counties). During the period 1916-1969, only one tornado caused injury to people or major property damage within these four counties. This one occurred in Green Bay, 27 miles WNW of the site, on May 10, 1959, at 8:50 p.m. Three persons were injured and property damage ranged from $500,000 to $5,000,000. The tornado path was 6 miles long and 600 yards wide. The region north of Sheboygan, along the Lake Michigan shore, appears to be relatively free of tornadoes. Approximately six tornadoes occurred in the Green Bay-Kewaunee area on April 22, 1970. Damages were estimated at approximately $500,000 and four to five people were injured.

Tornadoes tend to advance from the west with most of the tracks from the southwest to northeast. Maximum occurrence during the year is in May, with 90 percent reported in May through September. According to statistical methods proposed by Thom, (Reference 14) the probability of a tornado striking a point within a given area may be estimated as follows:

$$P = \frac{zt}{A}$$

P is the mean probability per year, z is the mean tornado path area, t is the mean number of tornadoes per year in area A. The value of t is 16.7 for Wisconsin and 1.2 for the four counties surrounding the Kewaunee site, if the April 22, 1970 tornadoes are included. The average path length and width for tornadoes occurring in the state are 7 miles and 200 yards, respectively, and yield a value of z equal to 0.80 square miles. Using a value of A equivalent to the total area of Brown, Door, Kewaunee, and Manitowoc counties yields:

$$P = 4.86E\text{-}04/yr$$

An equivalent value of 2.45E-04 per year is obtained using data based on the entire state.

At a 95 percent confidence interval Thom's formula becomes:

$$P' = P\left[1 \pm \frac{1.96}{(N)^{.5}}\right]$$

N is the total number of tornadoes in the area of concern during the ten years of record, 1960-1969 (the tornadoes of April 1970 are also included for conservatism).

The 95 percent confidence limits in the four counties around the site are 7.65E-04/yr and 2.09E-04/yr. The mean recurrence interval, R = 1/P', is 2060 years, and at the 95 percent confidence limits, the recurrence intervals R = 1/P', range between 1310 and 4770 years. The danger from tornadoes is therefore very slight.

Damage caused by tornadoes result from three principal effects:

1. The dynamic forces resulting from the high velocity vortex winds;
2. The bursting forces caused by differential static pressure resulting from the sharp pressure reduction in the immediate vicinity of a tornado funnel;
3. The impact of missiles generated by (1) and (2) above.

The most widely accepted values of maximum wind speed in a tornado appear to be about 300 mph (References 15, 16, and 17) for a very severe tornado at the peak of its intensity. Some sources mention values as high as 500-600 mph, (References 15 and 18) but these estimates appear to be based on indirect observations of phenomena such as straw driven into trees, etc., and are not regarded as authoritative.

The highest directly observed wind velocities were derived from motion pictures of debris in the Dallas tornado of April 2, 1957 (Reference 16 and 19). These velocities ranged up to 170 mph and resulted in a maximum wind vector of 227 mph. If higher velocities were present, they must have been very localized and not typical of the average wind on large bodies and structures.

The design wind speed of 300 mph with a forward progression of 60 mph is about 36 percent greater than that of the Dallas tornado and is thought to be conservative in view of the Kewaunee location. The structural design criteria used to assure adequate design to accommodate the most severe storm conditions are discussed in Appendix B of the Kewaunee USAR.

Kewaunee is located in Region I of the NRC tornado risk regionalization scheme given in WASH-1300. This region has the highest tornado hazard of the three NRC regions. The mean values of wind speeds and frequencies of exceedance are tabulated below. Please note that similar values were used in the extreme wind analysis of Point Beach, located 3 miles from Kewaunee.

Wind Speed (mph)	Frequency of Exceedance
74	5.38E-04
93	3.20E-04
134	1.04E-04
182	2.19E-05
234	3.95E-06
290	6.30E-07
349	7.33E-08

Hence, based on the design wind speed of 300 mph, the tornadoes pose no significant threat to Kewaunee.

5.1.4 Results, Recommendations, and Conclusions

Due to the low frequency of high winds and tornadoes at the Kewaunee site, it is concluded that the contribution to plant risk from severe wind events is insignificant.

The severe wind protective measures and design features instituted at Kewaunee are consistent with a highly safe plant design with very low risk contribution. Therefore, no design changes are currently recommended to protect the plant from severe winds.

5.2 External Floods

This section examines potential external flooding events that might initiate an accident sequence leading to core damage. The potential flooding events considered include: dam failures, lake flooding, river flooding, and intense precipitation.

5.2.1 External Flooding Source Screening

As reported in NUREG-0965 (Reference 20), there are no on- or off-site dams associated with, or in the proximity of Kewaunee. Also, based on the Kewaunee USAR, local topography precludes any flooding from the landward side of the site. For these reasons, dam failure and flooding from inland lakes and streams are not applicable to Kewaunee.

5.2.2 Screening Summary

The only sources of external flooding that could potentially impact Kewaunee are Lake Michigan and intense precipitation.

5.2.3 Analysis

External flooding methodology involves the determination of the maximum possible flooding levels, and the effect these flooding levels have on the plant. If the plant elevation precludes any flooding from these maximum flooding levels, the analysis is complete; if the elevation is insufficient to preclude flooding, further analysis is required.

A. Flooding from Lake Michigan

Provisions were made in the plant design to protect safety-related plant structures and equipment from flooding, waves, storms, and other phenomena generated in the lake.

According to U.S. Geological Survey figures from the Kewaunee USAR, the normal water datum of Lake Michigan is 577.5 feet above mean sea level (MSL). The lowest recorded level of the lake was 575.4 feet above MSL in 1964; the highest recorded level was 582.5 feet above MSL in 1986. The current (May 1994) lake level at the Kewaunee site is 579.9 feet above MSL. The lowest plant access doors, the two screenhouse doors, are 586 feet above MSL. Even these doors, however, are 3.5 feet above the highest recorded lake level. In the unlikely event that the lake level would rise above this level, the plant would be protected by solid steel normally closed and locked doors that open toward the outside and would therefore be pushed closed by the flood waters. It is assumed here, as in the internal flooding study, that these doors would remain intact. The next higher doors are at 605 feet above MSL, 22.5 feet higher than the highest recorded lake level.

Although the Kewaunee site is well above the normal lake levels, external flooding from abnormal lake levels is possible. Seiches are phenomena capable of producing large temporary deviations in lake water levels.

Seiches are oscillations in the level of lakes and similar bodies of water caused by the passage of squall lines across the body of water. In Lake Michigan, these squalls have their fronts oriented NE to SW and are accompanied by an abrupt increase in barometric pressure and local high winds. There have been a number of seiches recorded in the Great Lakes, the great majority of which were of only a few inches amplitude and, therefore, of no consequence. A few, however, have caused considerable flooding damage, and even loss of life. The most severe of the large seiches occurred on June 26, 1954 and caused water level increases of up to 10 feet at North Avenue in Chicago, Illinois. The greatest level increase recorded on the lake's eastern shore was 6 feet at Michigan City, Indiana, according to the Kewaunee USAR and the D. C. Cook USAR (Reference 21).

Seiches do not have the rapidity or damaging power of a wind-wave of equal height. Instead, the rise of water is continuous over several minutes, and damage is primarily due to flooding.

Within the bounds of seiche-causing conditions, the most severe initiating meteorological condition may be assumed to be a squall line traveling the entire lake from a direction west of northwest with a progress velocity sufficient to match the natural oscillation mode of the lake's southern sub-basin and producing a seiche front so shaped as to trap against the shore at the plant site.

The infrequency of seiches of significant size on Lake Michigan restricts to some degree the volume of recorded data from which future seiche characteristics may be predicted. The great quantity of information available concerning other large bodies of water, including measurements and observations of actual seiches, the characteristics of the shoreline at the plant site, historical meteorological conditions, computations based upon mathematical models, etc., confirm that no water level increase of as much as 8 feet should ever be experienced at the plant site (Kewaunee and D. C. Cook USARs). In conclusion, such an increase in water level is of no concern to plant safety since the screenhouse doors would prevent water from entering the plant even if such a seiche would occur.

B. Flooding from Intense Precipitation

Precipitation flooding analysis consists of first determining the maximum amount of precipitation an area can receive followed by an analysis of water removal for the area (i.e., runoff analysis).

The rain water concerning plant safety usually comes from local, convective type heavy rainstorms, which are characterized by high rain intensity over a relatively short duration, normally less than an hour (Reference 22).

Probable Maximum Precipitation

Rainfall frequency analysis is best performed using area-specific data from rain gauges or if no rain gauge data is available then the results from nearby gauged sites can be meteorologically transferred to the area in question.

One such frequency analysis performed on recorded point rainfall data at gauged stations and generalized to ungauged stations is the U.S. National Weather Bureau Technical Paper No. 40 (Reference 23).

However, because of the potential seriousness of external flooding induced nuclear plant failure, the use of probable maximum precipitation (PMP) is usually recommended for plant design. The PMP is defined in hte U.S. National Weather Services, NOAA Hydrometeorological Report No. 52 (Reference 25) as "the theoretically greatest depth of precipitation for a given duration that is physically possible over a particular drainage area at a certain time of year". The derivation of the PMP estimate only produces the theoretical maximum precipitation produced by the combination of reasonably conceivable worst ydrometeorological conditions occurring concurrently; there is no frequency of occurrence implied. Therefore, the PMP is useful only as a guide in plant flood prevention design. Since there is no frequency implied by the PMP, the plant must be designed to withstand the possible flooding effects from the PMP event.

PMP estimates for a 10 mi^2 area about the Kewaunee site for durations from 0.5 to 24 hours are given in Table 5-3, which is based on NOAA Hydrometeorological Report No. 52 and the Standard Handbook for Civil Engineers (Reference 26). For comparison, Table 5-3 also contains the 100 year recurrence interval extreme rainfall estimates calculated in the U.S. National Weather Bureau Technical Paper No. 40, and the PMP to 100 year recurrence extreme rainfall depth ratios.

Although the PMP estimates are all at least five times as great as the respective 100 year recurrence depth for the same storm duration, the SRP Section 2.4.2 requires that plant designs satisfy the PMP flooding criteria. Therefore, it should be demonstrated that safety-related equipment is not prone to flooding from the probable maximum precipitation.

Table 5-3: PMP and 100 Year Recurrence Rainfall Depths for Storms with Durations from 0.5 To 24 Hours

Duration [hours]	*PMP Estimate [inches]	**100 Year Recurrence Depth [inches]	***PMP to 100 Year Depth Ratio
0.5	10.0	1.8	5.6
1.0	13.0	2.3	5.7
2.0	18.0	2.7	6.7
3.0	20.0	3.0	6.7
6.0	23.0	3 5	6.6
12.0	27.0	4.2	6.4
24.0	29.0	5.0	5.8

* PMP estimates based on [illegible] square mile drainage area.

PMP estimates for:
- 0.5 and 1 hour durations from Reference 24
- 6, 12 and 24 hour durations from Reference 23
- 2 and 3 hour durations obtained from interpolation.

** From Reference 20

*** This column underscores the conservatism of PMP criteria.

As another comparison the results of average, minimum, and maximum annual precipitation for the years 1971-1980 are provided in Table 5-4. These rates fall within the values used for the analysis.

Table 5-4: Precipitation*

Location	Ten Year Average (1971 - 1980)	Max Annual	Year	Min Annual	Year
Kenosha	32.92	46.12	1972	25.07	1975
Milwaukee	33.39	40.74	1978	26.45	1971
Port Washington	30.24	37.34	1978	21.51	1976
Manitowoc	30.25	36.08	1978	25.20	1976
Two Rivers	30.20	34.6[illegible]	1973	24.81	1976
Kewaunee	30.35	34.69	1977	21.68	1976
Green Bay	29.36	35.47	1975	17.85	1976

*Data obtained from Wisconsin State Climatologist and the National Oceanic and Atmospheric Administration

Runoff Analysis

The Standard Handbook for Civil Engineers describes a method to determine the volume of water from a rainstorm that must be removed (i.e., water that does not evapotranspire or seep into the ground). The peak discharge runoff, Q, is defined in the "Rational Formula" as:

$$Q = CIA; \text{ where}$$

Q = peak discharge (ft^3/s)
C = runoff coefficient (% of rain that appears as direct runoff)
I = rainfall intensity (in/hr)
A = drainage area (acres)

A value of 0.15 is used for the runoff coefficient, C, defined in the Handbook based on the soil in the vicinity of the Kewaunee site.

From Table 5-3, the greatest hourly rainfall intensity (13.0 inches) occurs during a one-hour storm. Since the Rational Formula is defined for an hourly rainfall intensity, the more intense half-hour storm (10.0 in/half-hour) is not used.

As previously stated, the PMP estimates are based on a 10 mi^2 area or 6400 acres (Note: Larger areas tend to produce lower PMP values). However, realizing that water depth and not water volume is the issue here, we will define the runoff depth rate, H, as:

$$H = Q/A = CIA/A = CI \text{ (in/hr)}$$

Using the above values for "C" and "I," 0.15 and 13.0 inches/hour, the runoff depth, or height of standing water left by the PMP storm that did not evapotranspire or seep into the soil is 2.1 in/hr (see Appendix B for calculation).

For storms greater than one hour the following runoff depth rates are expected (see Appendix B for calculation):

Duration of Storm (hr)	**Runoff Depth Rate** (in/hr)
2	1.4
3	1.0
6	0.6
12	0.3
24	0.2

The above results are expected since storm intensity tends to decay logarithmically with storm duration. Figures 5-2 and 5-3 show this logarithmic decay for both PMP and 100 year recurrence estimates.

The Kewaunee freeboard protects the plant from rainfall-induced external floods up to 6 inches. General runoff is toward the east to Lake Michigan. However, due to the immense size of Lake Michigan and its normal water level (approximately 23 feet below the plant elevation) no flooding of Lake Michigan from a combination of rain collection and runoff will ever endanger Kewaunee.

Figure 5-2: Rainfall PMP Estimates

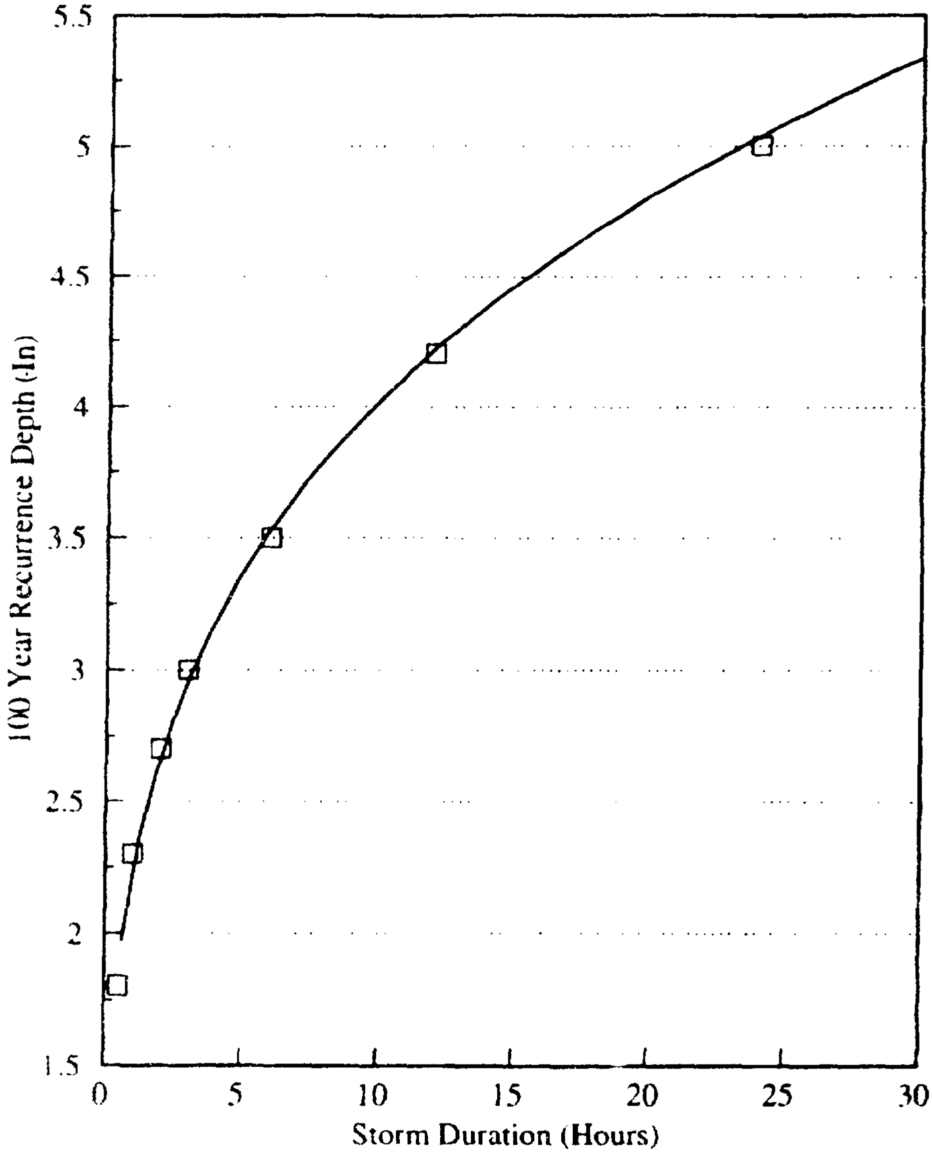
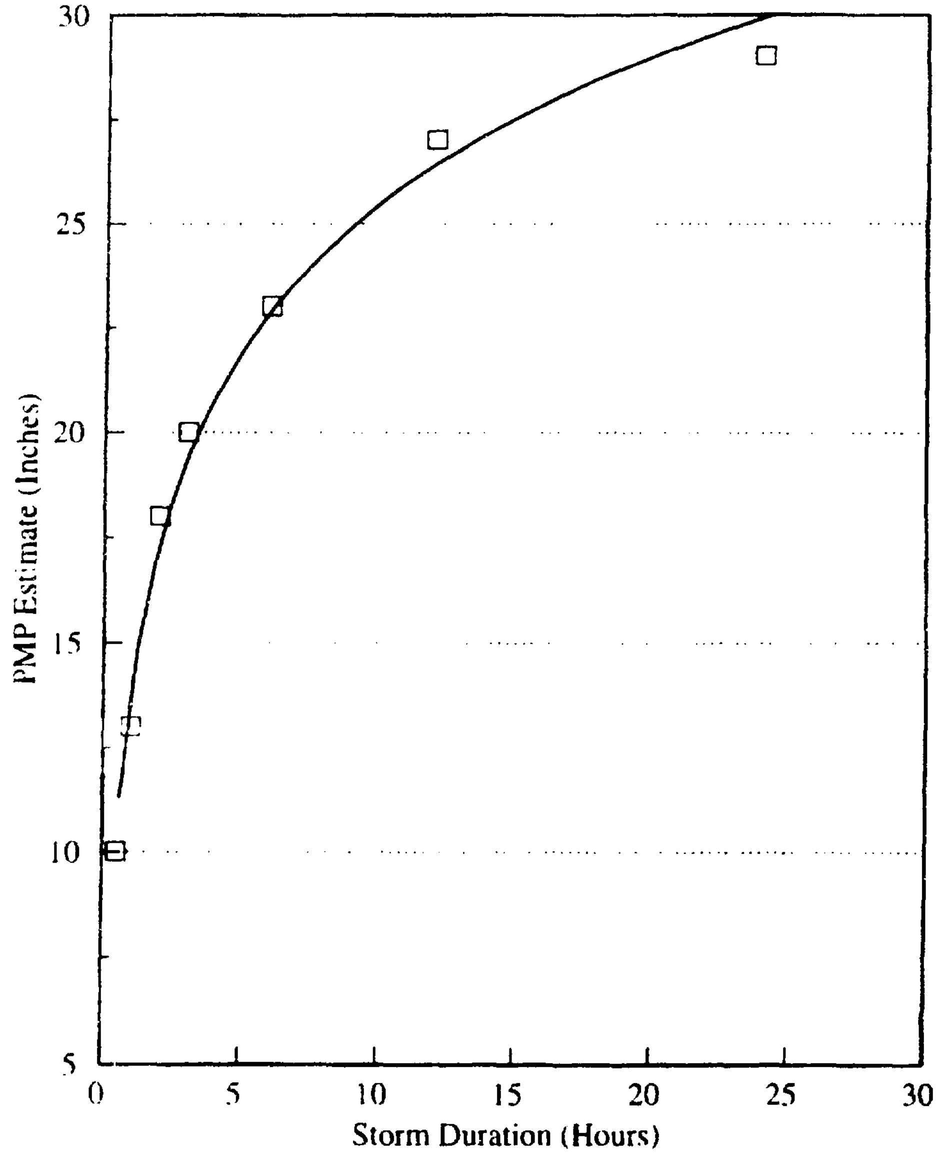


Figure 5-3: Rainstorm 100 Year Recurrence Depth



5.2.4 Results, Recommendations, and Conclusions

In view of the low frequencies and maximum flood levels, as well as the plant elevation, the topographical layout of the site, and the elevation of plant penetrations relative to safety-related equipment, it is concluded that the contribution to plant risk from external flooding is greatly dominated by other risk contributors.

An analysis was also performed to evaluate flooding based on probable maximum precipitation (PMP) criteria. It is concluded that Kewaunee is not endangered by the flooding based on the PMP criteria.

In fact, in 1993 when flood levels across the Midwest were at all time highs, there were no problems experienced at Kewaunee.

The external flooding protective measures and design features instituted at Kewaunee are consistent with a highly safe plant design with very low risk contribution. Therefore, no design changes related to external flooding prevention are currently recommended.

5.3 Transportation and Nearby Facility Accidents

This section examines transportation accidents including accidents from other industrial facilities near the plant that may initiate a sequence leading to core damage. The potential transportation events include those caused by aircraft, ship impact, roadway, and railroad accidents. Only nearby facility accidents that have an impact on the core damage frequency are considered. This analysis made extensive use of the control room habitability study performed in response to NUREG-0737 (Reference 33), which includes transportation and nearby facility accidents.

5.3.1 Transportation and Nearby Facility Accidents Source Screening

A. Nearby Facilities

The Kewaunee site is surrounded by sprawling farmland that does not contain any large industrial facilities (chemical, etc.) (Reference 27). This was confirmed by plant walk-down, driving in the vicinity of the plant, and by aerial photographs. Hence, accidents associated with nearby facilities do not contribute to core damage frequency and are not evaluated further.

B. Ground Transportation

Kewaunee receives no hazardous materials via pipeline, air, railway, ship or barge. Furthermore, there are no military installations, missile sites, or industrial facilities located beyond the Kewaunee site boundary at which an accident might cause interaction with the plant affecting public health and safety. Therefore, the only potential source of damage from off-site hazardous materials accidents is ground transportation accidents via road or rail.

The nearest major highway, Interstate 43, is more than 10 miles from the plant. Even though other roads are in the vicinity of the Kewaunee site, including Wisconsin State Highway 42, which crosses the site exclusion area, the volume of chemicals in a single truck shipment is very small. The nearest railway is approximately 10 miles from the plant.

The results of the analyses performed in NUREG-2462 (Reference 28) are equations relating the minimum standoff distance from an accident site to the hazardous material's equivalent TNT yield and static wall capacity of the structures.

The minimum standoff distance R (feet), is predicted by:

R = $f_\mu(W \times p_s^{-2})^{1/3}$; where,

W = TNT equivalent yield (lbs) for solid explosives,

p_s = static wall capacity (psi), and

f_μ = factor related to the permissible ductility μ as given by:

μ	f_μ
1.0	87
3.0	54
5.0	51

NUREG-2462 recommends that μ be set to 3.0, therefore f_μ equals 54. For Kewaunee, p_s is 3.0 psi, which corresponds to the minimum static lateral load design capacity of walls within tornado zone I, defined in Regulatory Guide 1.76 (Reference 29). Setting the distance to the constant R = 52,800 feet (10.0 miles), which is a conservative estimate for the perpendicular distance from the Kewaunee site to the railroad, the maximum TNT equivalent is found to be 8.413 billion pounds or about 4,207,000 tons.

The TNT equivalent for fuel-air mixtures is calculated from NUREG-2462 by the equation $W = 2W_F^{1.07}$; where, W_F = weight of the hydrocarbon fuel (lbs). Using the above equation, the maximum amount of hydrocarbon fuel at the standoff distance of 52,800 feet is about 811,507 tons.

Based on the quantity of the chemicals required to cause any significant risk at the plant, the railroad and truck accidents pose no threat to Kewaunee.

C. Water Transportation

Due to the physical location of the Kewaunee buildings and structures, the only danger to the plant is from run-aground ships or barges collapsing the circulating water intake structure and ultimately causing flow obstruction of circulating water system intake lines. In the unlikely event of a loss of intake structure the unit would be shut down. However, the intake structure is designed so that flow can be maintained to remove heat from the component cooling water system and other service water system loads.

The circulating water inlet structure starts with three 22 foot diameter vertical inlet cones which discharge through 6 foot diameter outlet pipes to the 10 foot diameter intake conduit. The cones are located with their tops one foot above the lake bottom with approximately 15 feet of normal water depth and approximately 1600 feet from shore. A steel grid with 12 inch square openings and a hinged manway serves as a trash screen. The three cones are reduced to 6 foot diameter pipes which join at the 10 foot diameter steel pipe. The 120 inch diameter intake conduit is buried a minimum of 3 feet below the lake floor to ensure the necessary depth to provide the minimum net positive suction head (NPSH) for the circulating water pumps.

Two 30 inch auxiliary inlets in the top of the 120 inch intake pipe provide auxiliary supplies for service water. The two auxiliary inlets are located 50 ft and 100 ft shoreward from the intake cones. Each auxiliary inlet tee rises vertically to one foot above the lake bottom. Special screened cover plates are suspended 12 inches above the intake openings to prevent the entrance of debris. Each auxiliary water intake can supply in excess of 24,000 gpm. Spacing of the three inlet cones and the auxiliary inlets is such that the largest lake barge cannot directly cover all water inlets.

Therefore, no shipping docks in the vicinity of the plant, and no ships come close to the plant. Only small watercraft come close to the plant, and based on the elevation of the plant embankment adjoining Lake Michigan, they can cause no damage to the plant. Since the plant decay heat load can be removed by the service water system even if a shipping accident causes a loss of the circulating water system, the only credible shipping accident affecting the Kewaunee site cannot cause further plant damage leading to core damage or a radiological release. Therefore, an in-depth analysis is not required.

D. Air Transportation

The aircraft accident events that might initiate an accident sequence leading to core damage were evaluated. All private, commercial, and military aircraft and flight paths are examined.

The effect of an aircraft of sufficient weight, traveling at sufficient speed, crashing at a nuclear power plant site may result in physical damage such that a release of radioactive material from the reactor core may result. Only physical damage to the plant is considered because aircraft carry insufficient hazardous material.

The NRC SRP identifies acceptance criteria for siting nuclear power plants near airports and/or airways. The probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 (Reference 30) exposure guidelines is considered to be less than 1E-07 per year if the plant meets the criteria listed below:

(a) The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2.

(b) The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.

(c) The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

If the above proximity criteria are not met, or if sufficiently hazardous military activities are identified (see item b above), a detailed review of aircraft hazards must be performed.

The Federal Aviation Administration (FAA), in compiling airport use statistics, defines an aircraft operation as the airborne movement of aircraft in controlled or noncontrolled airport terminal areas and about given enroute fixes or at other points where counts can be made. There are two types of operations - local and itinerant. These are defined in the FAA Statistical Handbook of Aviation (Reference 31). Local operations are performed by aircraft that: (1) operate in the local traffic pattern or within sight of the airport, (2) are known to be departing for, or arriving from, flight in local practice areas within a 20 mile radius of the airport, and (3) execute simulated instrument approaches or low passes at the airport. Itinerant operations are all aircraft operations other than local operations.

Although the FAA defines local aircraft operations as those within a 20 mile radius of an airport, a 25 mile radius about the Kewaunee site is used for conservatism. This extra five miles will in fact, increase the area reviewed ($25^2/20^2 = 1.56$) by more than 50 percent.

Table 5-5 contains the names, distances and approximate number of operations per year of all airports within a 25 mile radius surrounding the Kewaunee site.

Table 5-5: Aircraft Accident Acceptance Criteria Calculations & Comparisons

Facility	Distance to Kewaunee Nuclear Plant – D (statute miles)	*No. of Operations per Year – N	**Number-Distance Criterion – C	***Acceptance Criterion Met? (Yes or No)
Austin Straubel Airport	25	125,000	625,000	Yes
Manitowoc County Airport	17	22,000	289,000	Yes
Military (Restricted Area) R-6903	17	145	289,000	Yes
Minnow MOA: Military Operations Area	10	35	50,000	Yes

* No. of Operations per Year data found in Appendix C

** Number-Distance Criterion is calculated by:

$C = 500 * D^2$ for $D < 10$ statue miles, and

$C = 1000 * D^2$ for $D > 10$ statute miles.

*** Acceptance criteria met if: $N < C$.

Using the criterion in (a) above, the probability of radiological consequences greater than 10 CFR Part 100 exposure guidelines from aircraft operations associated with those airports listed in Table 5-5 is considered less than 1E-07 per year (See Appendix C for data and calculations).

The nearest military training route is more than 5 miles from the Kewaunee site (Reference 32 and Appendix C). However, to add conservatism, all military airports within 25 miles from the plant were considered in the analysis. Therefore, criterion (b) concerning military training routes is satisfied, and again probability of exceedance of the radiological exposure guidelines set in 10 CFR Part 100 is considered less than 1E-07.

The distances from the Kewaunee site of local airports preclude danger from aircraft in approach or holding patterns over the airports.

5.3.2 Screening Summary

Based on the screening analysis performed in Section 5.3.1, railroad, roadways, shipping, and nearby facility accidents do not pose any significant threat to the safety of Kewaunee. Using the screening criteria found in the SRP it is determined that commercial and military flight cause no relevant safety hazard to the plant.

5.3.3 Analysis

The methodology used to determine the core damage frequency from in-flight crashes begins by first determining the frequency of an in-flight crash into the "effective plant area" for all types of aircraft using the flight path in question.

The total core damage frequency from in-flight crashes is the sum of all the individual core damage frequencies for all types of aircraft.

A core damage frequency less than 1E-07 per year precludes further analysis; otherwise, a more detailed plant specific analysis is required.

5.3.4 Results, Recommendations and Conclusions

Based on the location of Kewaunee with respect to the major roads, rail transportation, air traffic and nearby industrial facilities, there is no threat to the plant safety from any of these sources.

Unless flight patterns change, the air or ground traffic significantly increases, or any new industrial facility is opened in the vicinity of the plant, no plant modifications are recommended due to these events.

5.4 Hazardous Materials

This analysis began with a review of the 1989 Updated Control Room Habitability Report (Reference 34). This report was the result of a study performed in response to NUREG-0737 and includes an assessment of hazardous materials on-site as well as off-site.

A plant walkdown and a review of plant records was performed to verify that the control room habitability study assumptions were still valid. It was determined that the results and conclusions stated in the study are still valid, and that there is no threat to control room personnel from hazardous spills or releases.

The analysis was further expanded to consider the effects of a release of hazardous materials on safety-related equipment or the local operation of plant systems during emergencies. It was determined that a release of hazardous material would have no effect on safety related equipment, and furthermore, no hazardous materials were located near safety-related equipment. It was further determined that there are no credible hazardous material releases that would prevent an operator from locally operating plant equipment during plant emergencies.

5.5 References

1. USNRC NUREG/CR-2300, "PRA Procedures Guide," American Nuclear Society and Institute of Electrical and Electronic Engineers, January 1983.

2. USNRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)," June 28, 1991.

3. USNRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f)," November 23, 1988.

4. USNRC NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," May 1991.

5. USNRC NUREG/CR-5042, Supplement 2, "Evaluation of External Hazards to Nuclear Power Plants in the United States, Other External Events," Lawrence Livermore National Laboratory, February 1989.

6. "NRC Standard Review Plan," NUREG-0800, Revision 3, 1990.

7. USNRC NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987.

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24. L. C. Schreiner and J. T. Riedel, "Probable Maximum Precipitation Estimates, United States East of the 105th Meridian," United States National Weather Service, Hydrometeorological Report No. 51, June 1978.

25. E. M. Hansen, et al., "Application of Probable Maximum Precipitation Estimates -United States East of the 105th Meridian," United States National Weather Service, NOAA Hydrometeorological Report No. 52, August 1982.

26. F. S. Merritt, ed. "Standard Handbook for Civil Engineers," Third Edition, Sections 21-53 to 21-57, McGraw-Hill, 1983.

27. Map of Local Facilities from the Chamber of Commerce. Supplemented by the plant walk-down and drive-down.

28. R. P. Kennedy, et. al., "Capacity of Nuclear Power Plant Structures to Resist Blast Loadings," NUREG/CR-2462, also SAND83-1250 R4. RP, September 1983.

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32. Air Traffic Data near Kewaunee Nuclear Power Plant.

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APPENDIX A: Wind Data

STATION DATA AND RETURN PERIOD VALUES

This attachment provides the annual extreme wind data, anemometer data and selected return period values. The following is a description of the information provided:

1. STATION NAME - Includes 3-letter station identifier and 5-digit identification number where applicable.

2. EXPOSURE TYPE - WBO $z_o = .7$
   WBO** $z_o = .4$
   WBO* $z_o = .05$
   APT $z_o = .05$

3. PERIOD OF RECORD - 1887 - 1988

4. LATITUDE, LONGITUDE - In degrees-minutes.

5. MEASURED SPEED - Uncorrected speed extracted from register charts. An "E" preceding indicates estimated speeds. MISSING is self-explanatory.

6. TRUE SPEED - Measured speed corrected to true using appropriate tables for each anemometer type (Item 11).

7. SPEED AT STANDARD HEIGHT - Speed standardized to 10 meters (WBO*, WBO**, APT) or 30 meters (WBO) using appropriate roughness lengths.

8. DIRECTION - To eight compass points. UNK indicates unknown directions.

9. ANEM TYPE - 4c = 4 cup anemometer
   4c-b = 4 cup beaded anemometer
   4c-x = 4 cup experimental anemometer
   3c = 3 cup anemometer

10. ANEM HT - Numbers indicate height of the instrument above ground level. Preceding "E" indicates an estimated height based on:

    a. The first known instrument height and assuming this height existed from the beginning of the period or record, or

    b. A 50-foot elevation for locations with no available data. Most nonurban sites were instrumented at approximately this height early in their record history.

Following "R" indicates instrument mast is roof mounted. Following "G" indicates instrument mast is ground mounted.

11. REMARKS - Consecutive numbers for each year necessitating remarks with self-explanatory notes listed after the data set.

GREEN BAY, WISCONSIN WBO $Z_o = .7$ 1887-1948 44 31 88 01
WBAN #14831

Date	Measured Speed	True Speed	Speed at Standard Ht (30 M)	Direction	Anem Type	Anem Ht	Remarks
10/12/1887	60	47	73	SW	4c	57R	
08/08/1888	45	36	36	SW	"	"	
11/28/1889	60	47	73	N	"	"	
03/28/1890	60	47	73	N	"	"	
07/13/1891	60	47	73	SW	"	"	
04/02/1892	60	47	73	W	"	"	
02/19/1893	60	47	73	N	"	"	
05/18/1894	60	47	73	N	"	"	
09/22/1895	56	44	69	SW	"	"	
05/17/1896	50	40	62	SW	"	"	
04/18/1897	40	32	50	SW	"	"	
07/19/1898	50	40	62	SW	"	"	
06/04/1899	42	34	53	SW	"	"	
07/07/1900	50	40	62	S	"	"	
03/03/1901	46	37	58	SW	"	"	
07/30/1902	52	41	64	NW	"	"	
07/01/1903	67	52	81	NW	"	"	
04/15/1904	75	58	91	NE	"	"	
10/19/1905	60	47	73	NE	"	"	
11/21/1906	58	46	72	N	"	"	
03/19/1907	60	47	73	NW	"	"	
06/22/1908	66	51	80	W	"	"	
04/07/1909	70	54	84	NW	"	"	
04/23/1910	66	51	80	N	"	"	
05/01/1911	76	59	69	N	"	123R	
04/16/1912	60	48	55	SW	"	"	
11/09/1913	60	47	55	N	"	"	
06/24/1914	60	47	55	SW	"	"	
09/08/1915	60	47	55	SW	"	"	
05/08/1916	64	50	59	W	"	"	
01/21/1917	60	47	55	N	"	"	
03/09/1918	72	56	66	NE	"	"	
04/07/1919	66	51	60	NE	"	"	
12/14/1920	67	52	61	W	"	"	
02/16/1921	60	47	55	SW	"	"	
07/30/1922	72	56	66	NW	"	"	

Date	Measured Speed	True Speed	Speed at Standa-- Ht (30 M)	Direction	Anem Type	Anem Ht	Remarks
03/12/1923	64	50	59	N	"	"	
03/29/1924	56	44	52	NE	"	"	
04/18/1925	48	38	45	NE	"	"	
04/24/1926	58	46	54	N	"	"	
06/09/1927	50	40	47	S	"	"	
04/13/1928	46	43	50	NE	3c	"	
04/01/1929	51	48	56	NE	"	"	
05/01/1930	42	49	57	W	"	"	
03/28/1931	60	56	66	NE	"	"	
05/16/1932	58	46	54	W	4c	"	
03/19/1933	62	49	57	NE	"	"	
06/23/1934	62	49	57	W	"	"	
09/25/1935	62	49	57	SW	"	"	
08/15/1936	70	54	63	NW	"	"	
04/21/1937	56	44	52	NE	"	"	
08/16/1938	60	47	55	NE	"	"	
02/10/1939	60	47	55	SW	"	"	
11/11/1940	70	54	63	S	"	"	
08/29/1941	58	46	54	NW	"	"	
01/01/1942	58	46	54	N	"	"	
05/16/1943	44	42	49	SW	3c	"	
08/15/1944	60	56	66	SW	"	"	
04/05/1945	57	53	62	S	"	"	
11/21/1946	45	42	60	SW	"	30R	1.
01/30/1947	44	42	60	NE	"	"	"
02/19/1948	48	45	65	SW	"	"	"

1. Standardized to 30 M using $Z_o = .05$ due to exposure at temporary location.

Probability	Return Period (years)	30 M Wind Speed (mph) Type I Distribution
.50	2	61
.80	5	69
.90	10	73
.95	20	78
.96	25	79
.98	50	84
.99	100	89
.995	200	93
.998	500	99
.999	1000	103

GREEN BAY, WISCONSIN (GRB) APT $Z_o = .05$ 1949-1979 44 29 88 08
WBAN #14898

Date	Measured Speed	True Speed	Speed at Standard Ht (30 M)	Direction	Anem Type	Anem Ht	Remarks
10/10/1949	72	66	62	SW	3c	47R	
05/05/1950	96	88	83	SW	"	"	
03/03/1951	70	65	61	W	"	"	
04/13/1952	61	57	54	NE	"	"	
06/04/1953	80	73	69	SW	"	"	
03/25/1954	65	60	57	SW	"	"	
11/16/1955	73	67	63	W	"	"	
07/01/1956	68	63	60	NW	"	"	
07/29/1957	76	70	66	NE	"	"	
11/18/1958	64	59	56	SW	"	"	
07/08/1959	52	49	46	SW	"	"	
04/11/1960	60	56	53	W	"	"	
10/11/1961	55	51	48	SW	"	"	
11/20/1962	50	47	51	S	"	20G	
04/03/1963	54	50	55	SW	"	"	
04/13/1964	64	59	65	SW	"	"	
06/20/1965	48	45	49	W	"	"	
10/22/1966	42	40	44	SW	"	"	
05/18/1967	52	49	54	SW	"	"	
05/08/1968	58	54	59	SW	"	"	
08/13/1969	39	37	41	S	"	"	
06/17/1970	65	60	66	W	"	"	
02/27/1971	54	50	55	SW	"	"	
01/25/1972	45	42	46	W	"	"	
0[illegible]09/1973	50	47	51	NE	"	"	
[illegible]1/1974	38	36	39	S	"	"	
01/11/1975	64	59	65	SW	"	"	
06/15/1976	45	42	46	SW	"	"	
03/29/1977	54	50	55	SW	"	"	
01/26/1978	50	47	51	NW	"	"	
04/05/1979	43	41	45	N	"	"	

Probability	Return Period (years)	10 M Wind Speed (mph) Type I Distribution
.50	2	54
.80	5	62
.90	10	67
.95	20	72
.96	25	73
.98	50	78
.99	100	83
.995	200	88
.998	500	94
.999	1000	99

The data for the years 1980 through 1988 was obtained from the National Oceanic and Atmospheric Administration (NOAA) annual summaries.

Green Bay, Wisconsin

Table - Wind Speeds 1980 - 1988

Date	Measured Speed	Direction
05/30/80	40	SW
08/06/81	39	SW
04/03/82	48	NW
02/02/83	45	NE
12/16/84	52*	NW
08/06/85	53*	W
07/27/86	44*	NW
07/09/87	46*	SW
07/09/88	56*	NW

*Maximum Gust Wind Speed

APPENDIX B: Derivation Of Depth Rate

DERIVATION OF DEPTH RATE OF RUNOFF FOLLOWING A ONE HOUR PMP STORM

$Q = CIA$, where

- Q = peak coefficient (ft^3/s)
- C = runoff coefficient (ft^3 hr)/(acre in s)
- I = rainfall intensity (in/hr), and
- A = drainage area (acres)

If $C = 0.15$, $I = 13.0$ and $A = 6400$ acres, then $Q = 12{,}480\ ft^3/s$, or $44{,}928{,}000\ ft^3/hr$ spread over the 10 square miles drainage basin.

The average depth rate of runoff over the basin, H, would be:

$$H = 44{,}928{,}000/(10 \times 5280^2)$$
$$= 1.9 \text{ in/hour}$$

Note: The definition of "C" is such that it also equals the % of rain that appears as direct runoff; therefore, we can also say:

$$H = CI, \text{ then}$$

$$H = 0.15 \times 13.0 \text{ in/hr} = 1.9 \text{ in/hr}$$

DERIVATION OF DEPTH RATE OF RUNOFF FOR PMP STORMS OF VARIOUS DURATIONS

If C = 0.15, and H = IC;

Duration (Hrs)	PMP Estimate (In)	Hourly Intensity (in/hr)	H_{ave} (in/hr)
0.5	10.0	20.0	3.0
1.0	13.0	13.0	1.9
2.0	18.0	9.0	1.4
3.0	20.0	6.7	1.0
6.0	23.0	3.8	0.6
12.0	27.0	2.3	0.3
24.0	29.0	1.2	0.2

Note that the 0.5 PMP storm has a greater value for H_{ave} than the one hour PMP storm, i.e., 3.0 in/hr > 1.9 in/hr. However, the rainfall intensity associated with the 0.5 hour PMP storm does not persist the full one hour. The maximum height for the 0.5 hour PMP storm is 3.0 in/hr X 0.5 hr = 1.7 inches.

Note also that the depth rate of runoff is called H_{ave} since these storms are not of one hour duration and the Rational Formula is defined for hourly rainfall intensities. Recall that the brunt of the storm usually occurs in the first few hours.

APPENDIX C: Air Traffic Data

Kewaunee site Location 87° - 32'/44° - 20'

A. Austin Straubel Airport - Green Bay

1. 25 statute miles NW of plant by air
2. Rick Remely - Austin Staubel - Operations
   - Airways are 8 miles wide
   - Flights that could have impact on plant

     IFR flights = 100/day J106 (114°)

     VFR flights = 10/day V26-55 (115°)
3. 125,000 operations/year
4. Phone: (414) 431-5751

B. Door Co. Cherryland Airport - Sturgeon Bay

1. 36 statute miles NE of plant by air
2. Travel is N&S along shoreline, traffic expected during summer months
3. Phone: (414) 743-6952
4. = 50 operations/day during June, July and August - = 1/4 to 1/3 are N-S bound

C. Manitowoc Co. Airport - Manitowoc

1. 17 statue miles SW of plant by air
2. 22,000 operations/year for private and industrial. (No commercial flights into or out of the Manitowoc Airport from Manitowoc operations.)
3. Phone: (414) 683-4594

D. Military Training Flights

1. Nearest military training flight path is 27 statue miles NE of plant out over Lake Michigan. (IR609-610).

2. Military (Restricted area) R-6903
   - 17 miles from plant near Manitowoc
   - Altitude = (surface to 45,000 ft)
   - 145 operations/year = 47 total hours during 25 days [(608) 427-1445]
3. Minnow MOA (Military Ops area) = 10 miles SE of plant. Altitude = 10,000 to 18,000 ft., 35 operations/year
4. Phone: (608) 427-1445

E. Ephraim-Fish Creek

1. 50 miles from plant in Door County
2. Travel is N-S along shore line; traffic is expected during summer months
3. Phone: (414) 854-9711 - No estimates could be given

6. Licensee Participation and Internal Review Team

7. Plant Improvements and Unique Safety Features

8. Summary and Conclusions

SECTIONS 6, 7, 8

TABLE OF CONTENTS

6. Licensee Participation and Internal Review Team

6.1 IPEEE Program Organization

Wisconsin Public Service Corporation (WPSC) has committed substantial personnel power financial resources to its Individual Plant Examination for External Events (IPEEE) program. Due to the magnitude of the Kewaunee Nuclear Plant IPEEE Program, WPSC engaged the services of consultants (namely Westinghouse, J. R. Benjamin and Associates, and Stevenson and Associates) to support and direct efforts on the IPEEE. WPSC created a Kewaunee IPEEE team from the interval events PRA team which effectively used its personnel resources and provided WPSC with complete control and involvement in the IPEEE analyses. In the organizational structure, contractor personnel provided the overall task leadership while both the contractor and the WPSC team jointly performed all the analyses. Interactions between WPSC personnel and the contractor were conducted on a continual basis to resolve issues and incorporate plant specific knowledge. In addition to the IPEEE personnel, other WPSC engineering and support staff provided design and operational information, as well as internal review.

WPSC established an IPEEE Project Manager who was responsible for the overall performance of the IPE project and served as the primary point of contact for the Kewaunee IPEEE. For the Kewaunee IPEEE, an Independent Review Team of WPSC middle level management actively reviewed all results and insights.

The WPSC IPEEE team members were trained and involved in all aspects of the IPEEE project. This included taking part in the IPEEE plant walkdowns, becoming familiar with analyses performed by consultants, and authoring/reviewing sections of the various IPEEE analyses.

The IPEEE project was directed by Westinghouse and supported by WPSC in the Kewaunee IPEEE project with a core of experienced IPEEE personnel, led by a Project Coordinator. The Project Coordinator was responsible to the Project Manager for coordinating project activities and maintaining the project schedule. The Project Coordinator was the primary interface between the contractor personnel and the WPSC IPEEE Project Manager.

A task-by-task participation of the WPSC IPEEE team engineers in the development of the Kewaunee IPEEE is presented in Section 2.3 of this report.

6.2 Composition of Independent Review Team

Although the Kewaunee IPEEE program satisfied the requirements of 10 CFR 50, Appendix B, an additional Independent Review Team was organized to review the various IPEEE analyses. This team generally consisted of middle level managers from applicable engineering and operations organizations as indicated in Table 6-1. The team conducted reviews and commented on all aspects of the IPEEE analyses.

6.3 Areas of Review and Major Comments

All areas of the IPEEE were subject to independent review through either the 10 CFR 50, Appendix B process or through consultant support. WPSC engineers were directly involved in a majority of the analysis or review tasks associated with the IPEEE. This approach assured WPSC's involvement in the IPEEE. Although consultants did solely develop and review certain inputs to the IPEEE, WPSC engineers became familiar with these efforts and ensured that the IPEEE properly employed these inputs.

6.4 Resolution of Comments

All comments were formally documented and resolved. Any resolution items were dispositioned through immediate changes to the IPEEE models if the effects were anticipated to be significant to the results.

TABLE 6-1

INDEPENDENT REVIEW TEAM COMPOSITION

Title	IPEEE Area Reviewed
Nuclear Engineer (SQUG)	Seismic PRA
Asst. to Licensing Superintendent	All
Safety Systems Inspection Supervisor	Seismic PRA
Quality Assurance Auditor	Fire PRA*
Shift Supervisor	Fire PRA
Risk Assessment Engineering Supervisor [3]	All
Risk Assessment Supervisor [1]	Fire PRA
Senior Risk Assessment Engineer [2]	Seismic PRA Other External Events
Shift Supervisor	Seismic PRA
Structural Engineer - Nuclear	Seismic PRA

* Had recently assumed new position. Formerly Fire Protection Operations Supervisor.

[1] Responsible for SPRA and Other External Events Analyses

[2] Responsible for Fire PRA Analysis

[3] IPEEE Project Manager

7. Plant Improvements and Unique Safety Features

7.1 Level 1 Unique Safety Features

Based on performance of the Level 1 PRA analysis, several features of the Kewaunee design have been identified that reduce the likelihood of core damage. These include:

- High head safety injection pumps deliver flow if Reactor Coolant System pressure is less than 2200 psig, which is significantly higher than typical Westinghouse plants designated as low pressure plants.
- Containment sump recirculation can be aligned to the high head safety injection, low head safety injection and containment spray pumps from the control room.
- Three auxiliary feedwater (AFW) pumps (two motor-driven and one turbine driven for diversity), which are independent of external cooling water systems as the pumps are cooled by the fluid being pumped. The Service Water System serves as a backup suction supply to the three AFW pumps.
- Separate eight hour batteries for safeguards and non-safeguards equipment.
- Four safety-related service water pumps for a single unit site.
- The Chemical Volume and Control System has three positive displacement charging pumps which are independent of cooling water systems. One of the pumps is driven by a variable speed DC motor for speed control and is not dependent on instrument air for attaining maximum pump output. Two charging pumps have four hour air supply for speed control. All pumps are capable of being powered by an emergency diesel. Two of the pumps have an alternate AC diesel generator to supply power.
- Two independent methods for maintaining reactor coolant pump seal integrity, seal injection from the charging pumps and thermal barrier cooling via the Component Cooling Water System.

7.2 Level 2 Unique Safety Features

Based on performance of the Level 2 PRA analysis, several features of the Kewaunee design have been identified that reduce the likelihood of containment failure.

The first of these features is the Kewaunee containment heat removal capability. The plant is designed with four containment fan cooling units (FCUs) and two internal containment spray (ICS) trains. Only one FCU or ICS train is needed to preclude containment failure on overpressure. The FCU discharge piping is at high enough elevation to preclude the discharge from being submerged following a loss of coolant accident and refueling water storage tank (RWST) injection.

The Kewaunee containment free volume is such that complete oxidation of the fuel cladding does not produce enough hydrogen to challenge the containment structure. The open design promotes good communication between compartments, precluding hydrogen pocketing.

The geometry of the cavity and instrument tunnel is such that deentrainment of debris following high pressure melt ejections will occur, precluding direct containment heating (DCH) as a concern. The cavity floor is large enough to allow the debris to spread into a thin layer, allowing coolability through an ablated vessel. This minimizes the likelihood of non-volatile fission product release.

In general the containment design is capable of handling severe accidents. The conservative 95% confidence containment ultimate pressure is 2.7 times the design pressure. The containment penetrations are capable of withstanding high temperature conditions for extended periods of time. These safety features, inherent in the design of containment, allow the containment structure to respond to severe accidents.

7.3 Plant Improvements

No major plant changes have been deemed necessary based on the results of the Kewaunee IPEEE. Some equipment outliers were identified during both the seismic IPEEE/USI A-46 and the evaluation of relay chatter walkdowns. Table 7-1 lists the walkdown findings including the specific resolutions and schedules for completion.

Kewaunee is classified as a focused-scope plant for the purposes of performing relay chatter evaluation. Since the plant is included in the USI A-46 program, relays were evaluated in accordance with Section 6 of the GIP. A focused scope plant that is an USI A-46 plant thus needs only to conduct a "bad actor" relay review if such bad actors were found in the USI A-46 scope of review.

The USI A-46 review found 12 installations of the Westinghouse, Model SC relay in the 4160V switchgear. (Relays in Breakers 502 to 508 and 604 to 609.) The scope of the bad actor review was expanded to include the IPEEE equipment with no additional bad actor relays identified.

WPSC initiated Engineering Support Request (ESR) 94-005 to consider outright replacement of the 12 Westinghouse relays, or alternatively, a reworking of the system circuitry to exclude the Westinghouse relays once emergency AC power has initiated. As such, further consideration of the relay capacities in the SPRA model is not required. Details of the recommended solutions will be provided at a later date in the USI A-46 submittal.

Table 7-1
EQUIPMENT OUTLIERS
SEISMIC WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
Motor Control Centers MCC52F & MCC52FEXT.	Adjacent MCC's not bolted together, which may pose an interaction hazard based on relay chatter concerns.	Cabinet displacements during a design basis seismic event were determined. The evaluation concluded that the cabinets will not impact.
Diesel Generator Excitation & Control Cabinets DR101 & DR111.	Several fasteners on cast-in-place anchors were found missing. An overhead emergency light posed an interaction hazard to DR101.	Missing fasteners were installed during the 1992 refueling outage and a restraint was installed on the emergency light during 1993 refueling outage.
Station Service Transformers 51, 52, 61 & 62.	Transformer cabinets were found anchored to the floor with friction clips, which are considered undesirable according to USI A-46 walkdown guidelines.	A design change was initiated to have the transformer cabinet bases welded to embedded floor channels. Transformers 51 and 52 were modified during the 1994 refueling outage. Transformers 61 and 62 are scheduled for modification during a later refueling outage.
Relay Racks RR186 & RR187.	The relay racks are not bolted to adjacent panels, which may pose an interaction hazard based on relay chatter concerns.	An engineering support request was initiated to have the racks bolted together to eliminate the concern. A schedule for completion has not been determined.
Reactor Trip Breaker Cabinet RD106.	Several anchor bolts which connect cabinet to embedded channel were found missing.	A design change was initiated to have the cabinet sections welded to the embedded channel in lieu of installing bolts. Work was completed during the 1993 refueling outage.

Table 7-1
EQUIPMENT OUTLIERS
SEISMIC WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
SI Pump B Suction Isolation Valve SI5B.	One leg of a Unistrut frame is within 1/2" of the valve motor, which may present an interaction hazard.	IE Bulletin 79-14 pipe stress evaluation determined that displacement of the pipe and valve is approximately 1/8". Issue considered resolved.
Main Steam Header A Controlled Relief Valve SD3A.	Valve actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
SI Pump Makeup Valve SI101B to Accumulator.	Actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
Aux Feedwater Pump Lube Oil Pressure Switches 16016, 16019 & 16085.	All three switches identified as Mercoids, which are considered outliers for the USI A-46 program.	A design modification was previously initiated to have the switches replaced for other reasons. Modification completed in 1993.
Flux Mapping Transfer Cart (GI-131 Issue).	Two concerns identified; (1) lateral restraints for the 10-path assembly frame were never installed, and (2) chain hoist on overhead rail identified as a possible interaction hazard to 10-path assembly.	It was determined by analysis that lateral restraints are not required to support 10-path assembly under seismic loads. Administrative controls were implemented to restrain hoist at the fixed end of crane rail when not in use.
Overhead Fluorescent Lights.	Generic problem throughout safety-related areas of the plant. S-hooks on the chains supporting the lights are not closed, presenting a possible interaction hazard to equipment below.	A plant walkdown was conducted during the 1994 refueling outage to pinch the S-hooks closed.
Emergency Lights.	Some of the lights were found to not have seismic restraints installed, presenting a possible interaction hazard to equipment below.	Lights and battery units strapped to supports as required during 1994 refueling outage.

Table 7-1
EQUIPMENT OUTLIERS
SEISMIC WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
480V Switchgear Bus 62.	An empty spare breaker cabinet was used for parts storage, presenting a possible interaction hazard on the basis of relay chatter concerns.	Maintenance department notified of problem. Spare parts were removed and all other spare breaker cabinets were inspected for similar problems during 1993 refueling outage.
Control Room Ceiling.	Aluminum ceiling diffuser panels were considered as a possible hazard to operators if the diffusers were to dislodge from T-bar supports.	An engineering support request was initiated to have the diffuser panels tie-wrapped to the T-bar supports. A schedule for completion has not been determined.
Control Room Vertical Panel C.	Rear doors on panel could not be latched shut due to interference with cables that extend from rear of cabinet. Unlatched doors present possible interaction hazard on basis of relay chatter.	It was determined that the cables were temporarily in place to support radiation monitoring modifications. Doors could not impact with cabinet because of cable interference. Operations department agreed to latch doors shut following completion of work during 1994 refueling outage.
All Equipment	Possible interaction hazards due to loose or unrestrained portable equipment.	Plant procedure GNP 1.31.1 drafted to provide guidelines for control of portable equipment. Full implementation occurred June 1, 1994.

8. Summary and Conclusions

WPSC has performed a complete IPEEE including all credible external events at Kewaunee. This study was performed using a seismic and fire PRA with a fault tree linking methodology and a screening approach for other external events that meets the intent of NUREG-1407. The Kewaunee IPEEE documents the computer models and the results of the analysis that together comply with 10 CFR 50, Appendix B. While contract personnel were used for the Kewaunee IPEEE, WPSC personnel were involved in every aspect of this analysis through either detailed review of contract work or actual performance of the analysis. The agreement with the contractor includes a complete transfer of technology upon their completion of services. This technology transfer allows WPSC to update the Kewaunee IPEEE in-house with minimal additional contract work.

It is the intention of WPSC to use the Kewaunee IPEEE as a decision-making tool in many aspects of engineering support and plant operations. Since the IPEEE is a highly technical document and uncertainties do exist in the analysis, the use and interpretation of IPEEE results and conclusions is currently limited to those individuals who have been intimately involved with its development. This approach avoids the problems that might arise from misinterpretation of the study.

Seismic

The seismic portion of the IPEEE is a Level 1 effort with both a qualitative and quantitative containment performance analysis, using a seismic PRA (SPRA) approach using guidance described in NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150." In keeping with the requirements of NUREG-1407, both Electric Power Research Institute (EPRI) seismic hazard curves and hazard curves developed by the Lawrence Livermore National Laboratory (LLNL) are used in the analysis. Plant walkdowns provided field information for the component fragility analysis.

Seismic Core Damage Summary

In general, no significant seismic concerns were discovered during the seismic IPEEE. The core damage frequency based upon the EPRI Kewaunee site-specific seismic hazard curve is 1.10E-05/year, whereas core damage frequency based upon the 1993 LLNL seismic hazard curve is 1.15E-05/year. Rankings of the dominant contributors to seismic core damage frequency remain the same regardless of the seismic hazard curve. Approximately 76 percent of the CDF is contributed by peak ground acceleration (PGA) values in the range 0.25g to 0.65g.

Seismic Containment Performance Summary

As part of the seismic containment walkdowns, containment mechanical penetrations and the containment isolation valves were analyzed for the ability to withstand seismic events. The penetrations and isolation valves form both inside and outside of containment were analyzed. Based upon these plant walkdowns, no significant seismic hazards were found to exist and it was determined that these components possess a high capability to withstand seismic events.

The seismic containment failure frequency is 6.24E-06 which is about the same as the containment failure frequency due to internal events (8.03E-06). The seismic containment failure frequency is 57% of the seismic core damage frequency. The containment failure median capacity and HCLPF for Kewaunee are 0.51 and 0.30g PGA, respectively. The median capacity is about four times the SSE and the HCLPF is two and a half times the SSE. The HCLPF is based on an evaluation of the seismic containment systems model that includes both random and seismic failures. No additional Level 2 vulnerabilities were discovered.

The results of the evaluations performed indicated that the containment as well as the systems designed to ensure containment integrity are seismically sound and no vulnerabilities could be identified.

Relay Chatter Issue

The relay chatter issue involved interfacing with the USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants" program at Kewaunee. USI A-46 "bad actors" relays identified as part of the USI A-46 program were also found within systems modeled for the SPRA. Plans have been developed to replace USI A-46 bad actor relays at Kewaunee that affect operability of safety-related equipment.

Fire

The internal fires analysis of the IPEEE is performed for Kewaunee using a Level 1 PRA and a qualitative and quantitative containment performance evaluation. A screening study based on the plant walkdowns and the EPRI Fire-induced Vulnerability Examination (FIVE) Methodology is used to screen out the less important fire areas, while a full PRA is performed for the remaining areas. This analysis is a new fire PRA and follows the guidance identified in NUREG-1407. The deficiencies of past fire PRAs identified in NUREG/CR-5088 "Fire Risk Scoping Study" are addressed in the Kewaunee Fire PRA. The Westinghouse WLINK code is used for fault tree and core melt quantification.

Fire Core Damage Summary

In general, no significant fire concerns were discovered in the Fire PRA. The core damage frequency due to fire is 9.8E-05/year. This is dominated by fires in the A and B auxiliary feedwater (AFW) pump rooms, which contribute 84% to the total fire core melt frequency. These areas each contain cabling for one train of safe shutdown equipment and cabling for numerous non-safety related equipment, such as transformers supplying offsite power. Kewaunee meets all the requirements of 10 CFR 50, Appendix R, (other than exemptions approved by the NRC) and an additional equipment failure or human error in addition to the fire is necessary for core melt to occur.

Fire Human Error Summary

Due to manual actions necessary to respond to a fire, human error is an important contributor to fire core damage frequency, contributing 56% (based on Fussel Vesely importance) to the fire core damage frequency. Of these, the largest contributor is failure to locally establish power to the dedicated (A) train of safe shutdown equipment, contributing 17% to the fire core damage frequency.

Fire Containment Performance Summary

Due to the robust design of Kewaunee's large dry containment, no containment failure due to overpressurization are produced by any fire-initiated core damage sequence within the 48 hour containment mission time. The containment failure frequency of 3.64E-05/year is due exclusively to failure to isolate containment. No additional containment failure modes unique to internal fires were identified.

High Winds, Floods, and Others

This analysis examines all credible external events other than seismic events and internal fires. Specifically examined in the other external events analysis are external flooding, aircraft accidents, severe winds, ship impact accidents, off-site and on-site hazardous materials accidents, and external fires. No vulnerabilities were identified that require detailed quantification of any accident events. It is, therefore, concluded that the effects from any of the other external events described here are not a significant concern at Kewaunee.

Plant Improvements

No major plant changes have been deemed necessary based on the results of the Kewaunee IPEEE. Some equipment outliers were identified during both the seismic IPEEE/USI A-46 and the evaluation of relay chatter walkdowns. These include 12 installations of the Westinghouse, Model SC relay in the 4160V switchgear. (Relays in Breakers 502 to 508 and 604 to 609.) WPSC initiated Engineering Support Request (ESR) 94-005 to consider outright replacement of the 12 Westinghouse relays, or alternatively, a reworking of the system circuitry to exclude the Westinghouse relays once emergency AC power has initiated. Details of the recommended solutions will be provided at a later date in the USI A-46 submittal.

Licensee Response/NRC Response/NRC Question Closure

Id	**2991**
NRC Question Number	**ALK-015**
Select Application	**Licensee Response**
Response Date/Time	**5/14/2010 7:50 AM**
Closure Statement	
Response Statement	**During an informal phone conversation on May 13, 2010, the NRC requested the Powerpoint™ presentation from a March 30, 2010 discussion related to snubbers be added to the responses for this RAI. Attached is the Powerpoint™ presentation. KPS notes that a more specific snubber accounting exists in correspondence related to a request to identify specific snubbers. Specifics about the presentation were discussed in detail during the phone conversation on March 30, 2010.**
Question Closure Date	
Attachment 1	**ALK-015 info.pdf** (414KB)
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Victor Cusumano** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**5/14/2010 7:52 AM**
Modified By	
Date Modified	

Kewaunee Snubber Presentation ITS 3.0.8

Tom Schneider
Cognizant Engineer

Overview

- Hot shutdown plant
- Piping configuration issues
- Snubber design analysis issues
- Maintenance considerations
- Safety conclusion

Hot Shutdown Plant

- Discussed in the LAR
- Designed and licensed for hot shutdown as safe shutdown
- Piping configurations are not consistently separated by train
- Examples:
  - RHR, SI and CC

Piping configuration issues

- ~98 safety related within the scope of 3.0.8
- Differing cylinder manufacturers
- Unique rod end attachments
  - Male versus female
  - sizes1 ½, 2 ½, 3 ¼ and 4 inch diameters with 5, 6 and 10 inch stroke lengths in various styles
- Snubbers are not easily interchangeable


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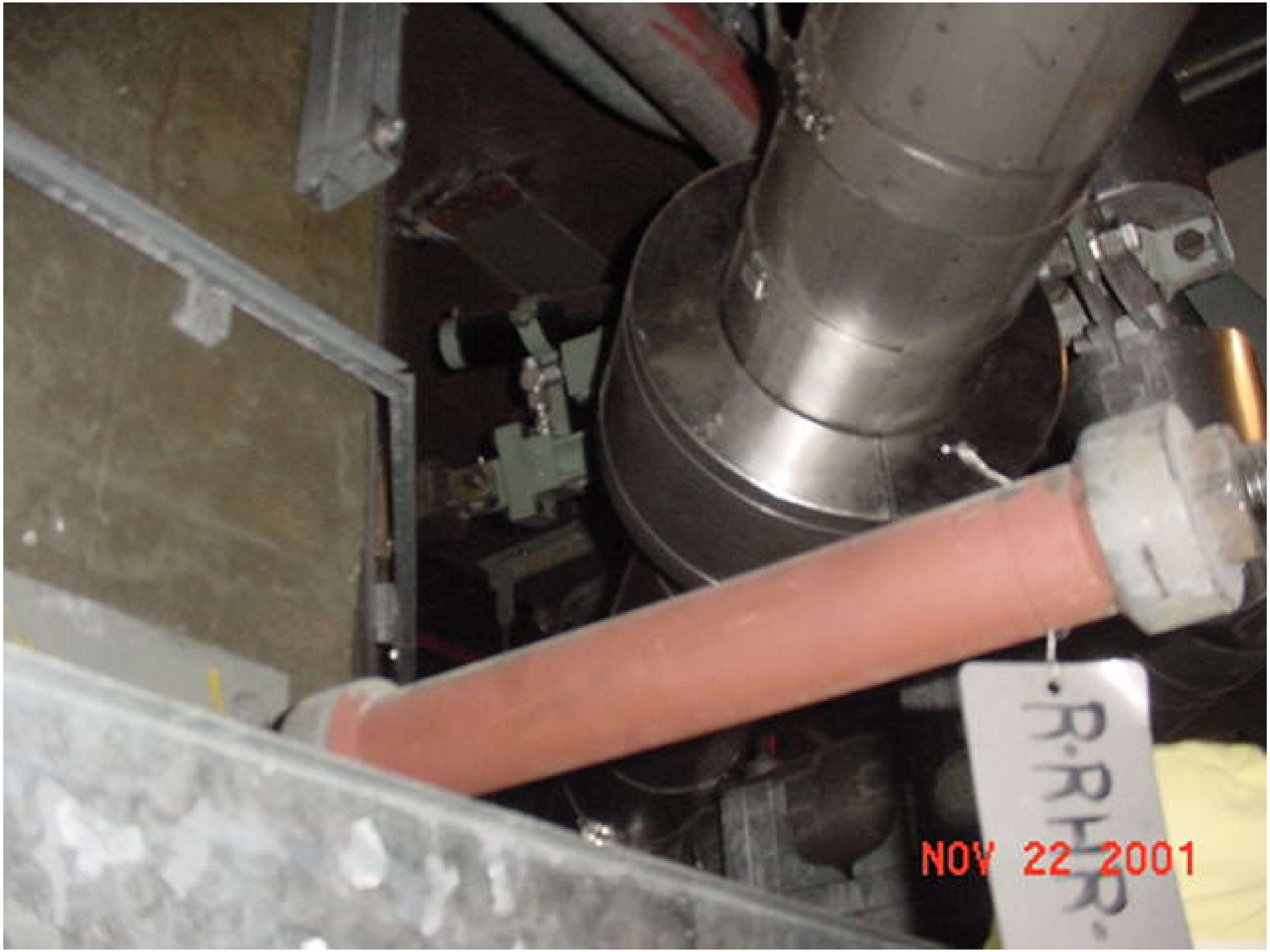
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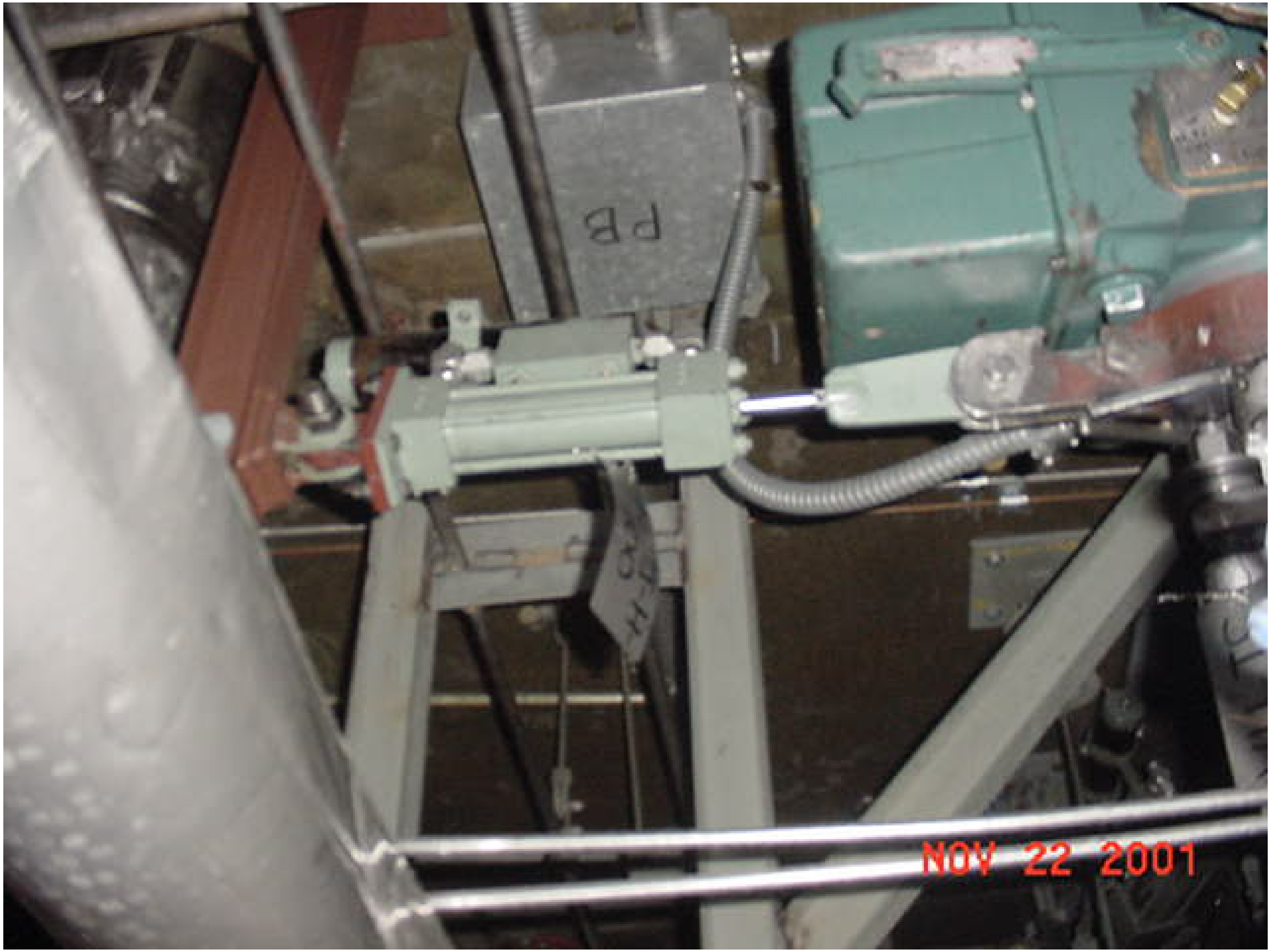
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Snubber Analysis

- Not analyzed for individual train designation per 3.0.8
  - A failed Snubber typically impacts both trains

Maintenance considerations

- Scaffold
- Spare parts or repair
- Physical location for some
- Confined space
- ALARA
- Off shift hours/Weather

Safety Conclusions

- 30+ years with 72 hours to repair
- Historically low failure rate
  - Maintenance and Materials
- Outage snubber performance
- Replacement time versus shutdown
- Repair is safer than a plant shutdown

Licensee Response/NRC Response/NRC Question Closure

Id	**3011**
NRC Question Number	**ALK-015**
Select Application	**NRC Response**
Response Date/Time	**5/17/2010 6:00 PM**
Closure Statement	
Response Statement	**The presentation slides provided on 5/14/10 mentioned that the snubbers had a historically low failure rate. Please provide a more detailed historical perspective on how often snubbers are found or declared inoperable in any given year.**
Question Closure Date	
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision Victor Cusumano Audrey Klett Carl Schulten**
Added By	**Audrey Klett**
Date Added	**5/17/2010 8:25 AM**
Modified By	
Date Modified	

Licensee Response/NRC Response/NRC Question Closure

Id	**3201**
NRC Question Number	**ALK-015**
Select Application	**Licensee Response**
Response Date/Time	**5/25/2010 10:00 AM**
Closure Statement	
Response Statement	**The NRC requested a more detailed historical perspective on how often snubbers are found or declared inoperable in any given year.**

Since 2000, the following were the most significant snubber deviations identified:

In May 2000, during performance of a snubber visual inspection surveillance, a snubber was identified with a missing pipe clamp nut (i.e., not per the applicable drawing), and the reservoir fluid level appeared to be low. Follow on inspection concluded that the reservoir fluid level was normal (i.e., within the inspection acceptance criterion) with the ram extended. No leakage was identified. The snubber was reinstalled with the missing nut replaced. In this case the snubber was concluded to have been operable.

In November 2001, during routine performance of snubber functional testing surveillance during a refueling outage, a single snubber in test group 4 was found out of the acceptance criterion range for bleed rate. The plant was in a refueling outage with the supported system out of service and the snubber was not required, therefore expedited repair was not required. This snubber failure resulted in testing an additional snubber test group (group 5). This snubber was rebuilt, tested satisfactorily, and reinstalled. In this case the snubber was determined to have been operable while installed.

In January 2006, during performance of a snubber visual inspection surveillance, the same snubber discussed above (i.e., May 2000) was again identified with a missing pipe clamp nut (i.e., not per the applicable drawing). The missing nut was replaced. In this case the snubber was initially declared inoperable, and later, following engineering evaluation, was concluded to have been operable.

In summary, during the last 10 years no snubbers were found or ultimately determined to be inoperable. However, note that in the one case above a snubber was initially declared inoperable, and later, following engineering evaluation, was concluded to have been operable.

Question Closure Date	
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**5/25/2010 9:58 AM**
Modified By	
Date Modified	

Licensee Response/NRC Response/NRC Question Closure

Id	**3431**
NRC Question Number	**ALK-015**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**6/7/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Victor Cusumano** **Audrey Klett** **Carl Schulten**
Added By	**Audrey Klett**
Date Added	**6/7/2010 9:04 AM**
Modified By	
Date Modified	

ITS NRC Questions

Id	**1841**
NRC Question Number	**ALK-016**
Category	**Editorial**
ITS Section	**3.0**
ITS Number	**3.0**
DOC Number	
JFD Number	
JFD Bases Number	**7**
Page Number (s)	**Vol. 5, page 52 of 63**
NRC Reviewer Supervisor	**Carl Schulten**
Technical Branch POC	**Add Name**
Conf Call Requested	**N**
NRC Question	**JFD #7 for ITS 3.0 Bases, page 52 of 63 of Volume 5, states: "Furthermore, these changes are consistent with industry approved TSTF-494T." This TSTF has not been formally resolved by the NRC; therefore, the NRC staff should not reference this TSTF as a basis for accepting the proposed change. Please delete this sentence from the JFD.**
Attach File 1	
Attach File 2	
Issue Date	**3/5/2010**
Added By	**Audrey Klett**
Date Modified	
Modified By	
Date Added	**3/5/2010 3:25 PM**
Notification	**NRC/LICENSEE Supervision** **Audrey Klett**

Licensee Response/NRC Response/NRC Question Closure

Id	**2591**
NRC Question Number	**ALK-016**
Select Application	**Licensee Response**
Response Date/Time	**3/15/2010 3:40 AM**
Closure Statement	
Response Statement	**KPS included the reference to TSTF-494T in order to be consistent with the exact same Justification for Deviation in the Davis-Besse ITS conversion. The Davis-Besse JFD (ITS 3.0 Bases, JFD 16) included the same last sentence concerning TSTF-494T, and the NRC did not question its inclusion. KPS maintained these words so that the JFD would be the same as one already approved by the NRC. However, KPS has no problem removing the sentence from the JFD, since the JFD itself provides adequate justification for the change. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS conversion amendment.**
Question Closure Date	
Attachment 1	**ALK-016 Markup.pdf** (268KB)
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Robert Hanley** **Jerry Jones** **Bryan Kays** **Audrey Klett** **Ray Schiele**
Added By	**Robert Hanley**
Date Added	**3/15/2010 3:36 PM**
Modified By	
Date Modified	

JUSTIFICATION FOR DEVIATIONS
ITS 3.0 BASES, LCO AND SR APPLICABILITY

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 5.1.3.

2. The ISTS contains bracketed information and/or values that are generic to all Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is provided. This is acceptable since the ISTS generic information/value is revised to reflect the current plant design.

3. The ISTS Figure B 3.0-1 is located within the text portion of LCO 3.0.6. The typical configuration for locations of figures is at the end of the applicable section. The figure is relocated to the end of the LCO section for 3.0. This is acceptable since the relocation of the figure to the end of the section is in accordance with the format of the ITS.

4. Typographical/grammatical error corrected.

5. These changes are made to be consistent with changes made to LCO 3.0.4.

6. Changes made for enhanced clarity.

7. The following fixes to incorrect statements for the examples in the LCO 3.0.6 Bases have been made. Specifically:

   1) Example B 3.0.6-1 is changed from "If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5" to "If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11." Examining Figure B 3.0-1, it is clear that if there is a loss of safety function in System 5, there is also a loss of safety function in the systems supported by System 5, i.e., Systems 10 and 11. This relationship is explicitly listed in Example B 3.0.6-3 and to not do so here is inconsistent and confusing as it leads the reader to believe that Systems 10 and 11 do not have a loss of safety function. Furthermore, System 5 of Train B is not a supported System of System 2 of Train A, since they are in different trains. Thus, the word “supported” has been deleted.

   2) Example B 3.0.6-2 is changed from "If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5" to "If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11." The phrase "which in turn is supported by System 5" is confusing. System 5 is not inoperable and does not lead to the loss of safety function. Examples B 3.0.6-1 and B 3.0.6-3 do not discuss OPERABLE support systems. This phrase adds no value and leads the reader to believe there is some special relationship with System 5 which does not exist.

   Kewaunee Power Station needs to make these changes since the current wording in the ISTS 3.0.6 Bases are not correct. ~~Furthermore, these changes are consistent with industry approved TSTF-494T.~~

Licensee Response/NRC Response/NRC Question Closure

Id	**2601**
NRC Question Number	**ALK-016**
Select Application	**NRC Question Closure**
Response Date/Time	
Closure Statement	**This question is closed, and no further information is required at this time to draft the Safety Evaluation.**
Response Statement	
Question Closure Date	**3/15/2010**
Attachment 1	
Attachment 2	
Notification	**NRC/LICENSEE Supervision** **Victor Cusumano** **Audrey Klett**
Added By	**Audrey Klett**
Date Added	**3/15/2010 3:40 PM**
Modified By	
Date Modified	