Final Precursor Analysis

Accident Sequence Precursor Program—Office of Nuclear Regulatory Research

Point Beach 1 and 2	Potential Common Mode Failure of All Auxiliary Feedwater Pumps	
Event Date: 11/29/2001	**LER:** 266/01-005	**ΔCDP =** 6×10^{-4} (Unit 1) **ΔCDP =** 7×10^{-4} (Unit 2)

Condition Summary

Description. This analysis involves a design deficiency in the auxiliary feedwater (AFW) pumps air-operated minimum flow recirculation valves. The valves fail closed on loss of instrument air, which, combined with inadequacies in plant emergency operating procedures, could potentially lead to pump deadhead conditions and a common mode, non-recoverable failure of the AFW pumps (Refs. 1 and 2).

Point Beach Nuclear Plant is a two-unit site served by a shared instrument air system. Each unit has a turbine-driven AFW pump (pumps 1P29 and 2P29), which can supply water to both steam generators. Additionally, the plant has two motor-driven AFW pumps (pumps P38A and P38B), each of which can be aligned to supply water to a steam generator in each unit (see Figure 1). Each AFW pump has a recirculation line with an air-operated valve that automatically opens, as necessary, to ensure minimum flow through the pump. The recirculation valves require instrument air to open, and fail closed on loss of instrument air. Prior to the discovery of this design deficiency, there were no backup air or nitrogen accumulators associated with these recirculation valves.

During some plant upset conditions, the AFW system actuates automatically to provide feedwater flow to the steam generators for decay heat removal. Depending on the event, overfeeding of the steam generators may occur, resulting in overfilling the steam generators or overcooling the reactor coolant system (RCS). This overfeeding situation requires AFW flow to be reduced. One preferred method used at Point Beach for reducing AFW flow is to throttle or close the AFW pumps' discharge or flow control valves rather than securing the pumps. To prevent pump deadheading conditions, the pumps’ minimum flow recirculation valves provide a flow path back to the condensate storage tanks. If a pump's recirculation valve fails closed due to loss of air to the valve while the pump's discharge or injection valve is closed, the pump would experience insufficient flow, resulting in pump overheating and damage, possibly within minutes.

The pressurizer power-operated relief valves (PORVs) are air operated. The original plant design did not provide for feed and bleed capability using the pressurizer PORVs following a loss of instrument air. Nitrogen accumulators for the PORVs were strictly for low-temperature-overpressure-protection concerns during shutdown operations and have been procedurally isolated during power operations since 1979. Events involving loss of instrument air will also result in the loss of feed and bleed capability.

Condition duration. The condition has existed at both Point Beach units since original construction. Because the condition has existed for more than 1 year, the time for the condition assessment is 1 year. The period selected for the analysis is from November 30, 2000, to November 29, 2001, the date of discovery.[1]

Recovery opportunities. Recovery opportunities examined in this analysis included the following:

– Restoration of sufficient AFW flow to prevent deadhead damage to the pumps prior to damaging all AFW pumps

– Recovery of instrument air to a pressurizer PORV to permit feed and bleed cooling

– Recovery of main feedwater given recovery of instrument air pressure

Other conditions, failures, and unavailable equipment. Prior to the discovery of this condition on November 29, 2001, the utility had installed new orifices in the recirculation lines for both of the motor-driven AFW pumps (pumps P38A and P38B). Subsequently, the utility discovered that these orifices may quickly plug if service water, the alternate water supply for the AFW system, is used. (See Refs. 3 and 4.) In addition, the utility discovered that the recirculation valves for three of the four AFW pumps were supplied by a common de bus. Failure of this bus could result in only one motor-driven pump being operable (Ref. 4). Other design issues are also discussed in the inspection report (Ref. 4). A review of these design issues determined that they would not significantly change the risk results presented in this precursor analysis. Therefore, these issues are not included in this analysis but are addressed in a separate analysis.

Analysis Results

- **Importance**[2]

  For each unit, the risk significance of the design deficiency in the AFW pumps air-operated minimum flow recirculation valves is determined by subtracting the nominal core damage probability (CDP) from the conditional core damage probability (CCDP).

	Unit 1	Unit 2
conditional core damage probability (CCDP) - mean	6.3E-04	6.8E-04
nominal core damage probability (CDP) - mean	1.7E-05	1.8E-05
Importance (ΔCDP):		
95th percentile	1.3E-03	1.4E-03
point estimate	7.3E-04	7.9E-04
mean	**6.1E-4**	**6.6E-04**
5th percentile	2.1E-04	1.8E-04

1 The Accident Sequence Precursor Program limits the conditional assessment of risk to a 1-year period. For the time period selected, Unit 1 was critical for 7,680 hours and Unit 2 was critical for 8,316 hours.

2 Since this condition did not involve an actual initiating event, the parameter of interest is the measure of the incremental increase between the conditional probability for the period in which the condition existed and the nominal probability for the same period but with the condition nonexistent and plant equipment available. This incremental increase or "importance" is determined by subtracting the CDP from the CCDP.

The graph above presents the range of ΔCDP for Point Beach Units 1 and 2.[3] Based on the mean values, this is an increase of 6.1E-04 to 6.6E-04 over the nominal CDP over the calendar year selected due to the potential failure of all of the AFW pumps when the pumps' recirculation valves fail closed and failure of feed and bleed capability following loss of instrument air. (Slight differences in the results between units are attributed to the differences in operating times during the 1-year condition period.)

- **Dominant sequences**

  The four top sequences are seismic event, loss of service water, loss of offsite power, and loss of instrument air due to internal failure.

  – ***Seismic event:*** A seismic event results in non-recoverable damage to the non-seismically qualified instrument air system. The simplified event tree, shown in Figure 2, was added to the model to allow for uncertainty calculations.

  – ***Loss of service water:*** The events and important component failures in loss of service water Sequence 28 (shown in Figure 3) are:

    - postulated total loss of service water affecting both units,
    - successful reactor trip,
    - failure of the AFW system (due to damage to the AFW pumps resulting from the loss of instrument air pressure caused by loss of cooling to the air compressors), and
    - failure to recover service water cooling to the air compressors, resulting in
      – the inability to initiate feed and bleed cooling (due to loss of instrument air to the pressurizer PORVs) and
      – The inability to initiate secondary cooling via the main feedwater system (due to loss of instrument air to a steam generator atmospheric dump valve and a feedwater regulating bypass valve).

  – ***Loss of offsite power:*** The events and important component failures in loss of offsite power Sequence 18 (shown in Figure 4) are:

    - postulated loss of offsite power affecting both units,
    - successful reactor trip,
    - successful operation of the emergency power system,
    - failure of the AFW system (due to damage to the AFW pumps resulting from the initial loss of instrument air pressure caused by loss of electric power to the air

[3] A constrained non-informative prior was used to quantify industry experience on initiating event frequencies because it creates a diffuse distribution that accounts for the plant-to-plant variation in system reliability. The prior used a gamma distribution with a shape parameter (0.5), which causes the posterior mean to fall between the prior mean and the maximum likelihood estimate. Using this diffuse gamma distribution for initiating event frequencies causes the ΔCDP confidence interval to be nearly an order of magnitude and the ΔCDP point estimate to be higher than the mean.

compressors and bleed down of air pressure), and

- failure of feed and bleed cooling (due to operator failure to initiate feed and bleed cooling or the failure to recover instrument air).

– ***Loss of instrument air:*** The events and important component failures in loss of instrument Sequence 20 (shown in Figure 5) are:

- postulated total loss of instrument air affecting both units,

- successful reactor trip,

- failure of the AFW system (due to damage to the AFW pumps resulting from the loss of instrument air pressure), and

- failure to recover instrument air pressure, resulting in

  – the inability to initiate feed and bleed cooling (due to loss of instrument air to the pressurizer PORVs) and

  – the inability to initiate secondary cooling via the main feedwater system (due to loss of instrument air to a steam generator atmospheric dump valve and a feedwater regulating bypass valve).

- **Results Tables**

  – The conditional probabilities of the dominant sequences are shown in Table 1.
  – The event tree sequence logic for the dominant sequences is provided in Table 2a, and definitions of top events are provided in Table 2b.
  – The conditional cut sets for the dominant sequences are provided in Table 3.

- **Analysts**

  – Analysts (ABS Consulting): Michelle Johnson (lead), David Campbell, Charles Mitchell
  – NRC technical reviewers: Eli Goldfeiz, James Houghton, Gary DeMoss, Don Marksberry
  – ABS Consulting technical review: Leonard Palko

Modeling Details

- **Assessment Summary**

The design deficiency was modeled as an at-power condition assessment with all of the AFW pumps unavailable for 1 year for only those initiators that would involve loss of instrument air. These initiators include the following:

- Loss of instrument air (LOIA) caused by internal failures in the instrument air system
- Loss of offsite power (LOOP) to both units
- Loss of service water system (LOSWS) to both units
- Seismic event (LOIASEISMIC)

The Revision 3 standardized plant analysis risk (SPAR) model for Point Beach (Ref. 5) was used for this assessment. Event trees and associated fault trees, basic event probabilities, and initiating event frequencies were modified to reflect the condition being analyzed. These condition modifications include the following:

- Selection of initiators that would involve loss of instrument air, as noted above, including development of a simplified event tree for a seismic event
- Modification of the LOOP initiating event frequency that includes only dual unit LOOPs
- Accounting for the failure of all AFW pumps on the loss of instrument air

In addition, the model was modified to reflect updates to the SPAR model. These update modifications include the following:

- Updating initiating event frequencies for LOIA and LOSWS based on recent operating experience
- Modifying the LOIA and LOSWS event trees to include the following:
  - New top events to account for the opportunities to recover instrument air and service water based on the operating experience
  - New top events that credit the recovery of secondary cooling in certain sequences
- Updating uncertainty distributions for failure probabilities and initiating event frequencies so that uncertainty analysis can be performed

Modifications to the event tree and fault tree models and the bases for the changes are summarized below and discussed in detail in the attachments.

Two analyses were performed-one for Unit 1 and one for Unit 2-due to slight differences in operating time during the 12 months prior to the discovery of the condition.

- **Sequences of Interest**

  – ***Initiating events.*** Because of the vulnerability involving the fail-closed, air-operated valves on the AFW pumps' recirculation lines, loss of instrument air during AFW operations could result in total loss of the AFW system. During an initiating event in which AFW system flow is demanded, instrument air pressure could be lost due to causes that are independent of the initiating event or due to causes that share some dependency with the initiating event. It is the latter case that has the greater risk significance, which this analysis will examine.

    The initiating events that result in both a loss of instrument air pressure and reactor trip are:

    - LOIA to both units due to component failures in the instrument air system (e.g., compressor failure);
    - LOOP to both units that results in loss of electric power to both the instrument air and service air compressors and loss of air pressure due to usage or bleed down;

- LOSWS to both units that results in loss of cooling water to both the instrument air and service air compressors, trip of the compressors on loss of cooling, and loss of air pressure due to usage or bleed down; and

- a seismic event that results in non-recoverable damage to the nonqualified instrument air system (e.g., line failure).

– ***Sequence of events.*** For the four initiating events (LOIA, LOOP, LOSWS, and LOIASEISMIC) the following sequence of events leads to core damage:

- Initiating event that causes the total loss of instrument air pressure and results in a manual or automatic reactor trip

  For LOIA initiating events, the control room likely receives a low pressure annunciator alarm (89 psig) as the first alarm, based on operating experience. The alarm would be expected shortly after the occurrence of the other initiating events

- Secondary cooling lost due to low instrument air pressure to the balance of plant components (if not already lost due to loss of electric power, service water cooling, or seismic damage to non-seismic qualified structures and components)

- All AFW pumps feeding the unit (two motor-driven pumps and one turbine-driven pump) automatically start on low-low steam generator level

- Operators fail to recognize that the recirculation valves are closed upon loss of instrument air pressure. Instrument air pressure quickly degrades to the point (less than 40 psig) that the AFW pumps' recirculation valves fail closed (within 8 to 10 minutes)

- Operators choose to throttle or close the discharge valves or flow control valves for all of the AFW pumps, resulting in deadhead of the AFW pumps. Plant conditions following the trip require AFW flow to be controlled within 4 minutes (due to overcooling transient) or 13 minutes (due to steam generator overfilling)

- All AFW pumps fail within minutes due to deadhead conditions - pumps are not recoverable

- Operators fail to recover instrument air pressure in time for the initiation of feed and bleed cooling (within 30 minutes). (The PORVs, needed for feed and bleed cooling, are not available because of insufficient instrument air pressure to operate the valves. For LOIA events, operators must recover failed components in the instrument air system and recover air pressure within 30 minutes to allow initiation of feed and bleed cooling. For LOSWS events, operators must recover service water cooling to the instrument air compressors and recover instrument air pressure within 30 minutes to allow initiation of feed and bleed cooling. For LOOP (non-station blackout) events, operators must manually restore electric power to the instrument air or service air compressors and recover instrument air pressure within 30 minutes to allow initiation of feed and bleed cooling. For seismic events, damage to the instrument air system is assumed to not be recoverable)

Or operators fail to initiate short-term cooling (i.e., feed and bleed) or long-term cooling (i.e., secondary cooling, high-pressure recirculation, residual heat removal [RHR])

- Operators fail to recover instrument air pressure in time for recovery of secondary cooling via main feedwater (within 60 minutes). (Main feedwater is not available due to loss of electric power, loss of instrument air pressure, and/or loss of service water flow depending on the initiating event. For LOIA events, operators must recover failed components in the instrument air system and recover air pressure within 60 minutes to allow recovery of secondary cooling. For LOSWS events, operators must recover service water and instrument air pressure within 60 minutes to allow recovery of secondary cooling.)

  Or operators fail to initiate long-term cooling (i.e., secondary cooling, high-pressure recirculation, RHR)

- **Plant-Specific System and Operational Considerations (Facts)**

  Details of plant-specific system design and operational considerations are provided in Attachment 1. These are the facts upon which assumption and model modifications are based. Details are provided for the following:

  – AFW system design
  – Feed and bleed cooling design
  – Instrument air and service air system designs
  – Control room indications
  – Response to loss of instrument air
  – AFW flow control
  – Recovery of main feedwater

- **Important Assumptions**

  Details of these assumptions are provided in Attachment 2.

  – Operators fail to recognize that the recirculation valves are closed

  – Operators close the discharge valves for all of the AFW pumps, resulting in deadheading of the AFW pumps

  – No credit for operators detecting pump deadhead conditions (i.e., closed recirculation valves) and taking corrective actions to protect one or more AFW pumps

  – No credit for leakage past either the closed recirculation valves or the closed discharge valves providing adequate flow through the AFW pumps to prevent pump damage

  – No credit for the recovery of AFW pumps given failure due to deadheading conditions

  – No credit for the recovery of nitrogen air bottles to the pressurizer PORVs

  – No credit for the recovery of secondary cooling without instrument air or service water

- **Modifications to Event Trees and Fault Trees**

- *Seismic-induced loss of instrument air.* A new event tree (Figure 2) was added to the model to account for the seismic-induced loss of instrument air. This simplified tree has one top event with a single pseudo basic event set to TRUE. The tree is based on the assumption that a seismically induced loss of instrument air event is non-recoverable and, therefore, would lead directly to core damage (hence the pseudo-event set to TRUE).

- *Loss of instrument air and loss of service water.* The LOIA and LOSWS event trees were modified to include recovery of instrument air, service water, and main feedwater. Recovery-related changes were made to related event and fault trees, and basic event non-recovery probabilities. These changes are refinements to the SPAR model; therefore, these modifications are applied to the base case and change case. Details of these modifications are provided in the addendum to the Point Beach SPAR Manual (Attachment 5).

- **Modifications to Basic Event Failure Probabilities**

  Table 4 provides the basic event probabilities that were modified for this analysis. Changes are summarized below.

  - *Probability of common-cause failure of all AFW pumps (AFW-PMP-CF-ALL).* The common-cause failure probability that all the AFW pumps would fail was set to TRUE for those sequences in which instrument air would be lost (i.e., LOOP, LOSWS, and LOIA). This reflects the fact that given a LOOP, LOSWS, or LOIA initiating event, the AFW pumps would be damaged fairly quickly into the event and would not be recoverable.

  - *New basic events.* New basic events were created for the new top events added to the LOIA and LOSWS event trees. Details of the failure probability estimates are provided in the addendum to the Point Beach SPAR Manual (See Attachment 5). These basic events are also listed in Table 4.

- **Modifications to Initiating Event Frequencies**

  Table 41iststhe initiating event frequencies that were modified for this analysis. Changes are summarized below.

  - *Loss of offsite power initiating event (IE-LOOP).* Because of the design of the instrument air and service air systems at Point Beach (redundant compressors and diverse power sources), loss of all four compressors due to a LOOP would only occur if power were lost to both units. Types of LOOPs that would involve both units include dual- unit, plant-centered LOOP; grid-related LOOP; and severe weather-related LOOP. Operating experience data were reviewed to determine the frequency of plant-centered, dual-unit LOOP; grid-related LOOP; and severe weather-related LOOP. The mean frequency for IE-LOOP used in both the base case and change case is 8.8E-3/year (1.0E-6/hour). Details of the frequency calculation and the data used in the estimate are provided in Attachment 3.

  - *Seismically induced loss of instrument air (IE-SEISMIC).* A seismically induced loss of instrument air was also considered as a contributor to core damage. A simplified event tree was created for this purpose (see Figure 2). The safe shutdown earthquake for Point Beach is 0.12 g and the operating basis earthquake is 0.06 g. Because the instrument air system piping design is less robust than ANSI B31.1 piping design, the instrument air system cannot be assumed to withstand any seismic event greater than 0.06 g, without either performing a seismic analysis of the piping design or conducting visual inspections of the piping to determine seismic tolerance. Therefore, the return frequency for seismic events that would result in a loss of instrument air is conservatively estimated at 3.1E-4/yr (3.5E-8/hour) based on the lowest estimated ground acceleration value (50 cm/sec^2 or 0.05 g) at Point Beach from NUREG/CR-1488, *Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains* (Ref. 6). The mean frequency for this acceleration is 3.1E-4/yr. As the lowest site-specific value found in NUREG/CR-1488, this frequency is more appropriate than arbitrarily selecting the design basis earthquake, because no design value exists for instrument air system piping. The frequency of seismically induced core damage events for the base case is taken from the IPEEE (Ref. 7). The base case frequency is 1.5E-05/yr.

– ***Loss of instrument air (IE-LOIA) and loss of service water (IE-LOSWS) initiating events.*** The initiating event frequencies for IE-LOIA and IE-LOSWS were updated using recent operating experience. In both cases, total losses (to both units) are of interest because both systems are shared between units. Details of the frequency updates and the data used in the estimate are provided in the addendum to the Point Beach SPAR Manual (See Attachment 5).

– ***Initiating event frequency changes to eliminate unaffected sequences.*** Initiating events IE-LDC01, IE-LLOCA, IE-MLOCA, IE-SLOCA, IE-LOCCW, IE-RHR-DIS-V, IE-RHR-SUC-V, IE-STGR, IE-SI-CLDIS-V, IE-SI-HLDIS-V, and IE-TRANS were set to FALSE in the base case and the change case to reflect the condition being analyzed. The sequences associated with these initiating events have no shared dependencies with loss of instrument air; therefore, including them in the CDP and CCDP calculations with the common-cause failure of all AFW pumps (AFW-PMP-CF-ALL) to TRUE is not appropriate.

  Furthermore, because the condition being analyzed does not impact the SPAR change case for these other initiating events (i.e., the base case and change case are identical), there is no contribution to the delta CDP importance measure from these initiating events. Therefore, all accident sequences associated with these initiating events were removed from the GEM calculations by setting the frequencies for these initiators to FALSE in both the base and change cases.

- **Sensitivity Study- Potential Common-Mode Failure of All Auxiliary Feedwater Pumps**

  Several sensitivity studies were performed to determine the effects of key assumptions on the ΔCDP. These studies included the following cases: (1) varying likelihood that all AFW pumps would be failed on loss of recirculation flow, (2) varying the initiating event frequencies, and (3) reducing the likelihood that operators would fail to initiate feed and bleed cooling. As the results show (point estimate values for Unit 1), these sensitivity studies did not cause the ΔCDP to fall outside the bounds of the 5th and 95th percentile of the best estimate.

  – ***Failure of the AFW pumps.*** In the condition assessment, it was assumed that, early in the event, operators would throttle the discharge flow for all of the AFW pumps, resulting in the pumps operating in deadhead conditions. This would quickly lead to pump failure. Several sensitivity cases were run varying the likelihood that operators would detect pump deadhead conditions and respond quickly enough to save at least one AFW pump.

– ***Initiation of feed and bleed cooling.*** For LOOP sequences, failure of the operator to initiate feed and bleed cooling is an important event. The probability for this event was derived using the human error worksheet. In the worksheet, the nominal failure probability for human action is 1.0E-3. Several sensitivity cases were run varying the failure probability for human action.

– ***Loss of instrument air and loss of service water.*** The initiating event frequencies for loss of instrument air and loss of service water were estimated using operating experience data (see Attachment 5). For both events, a gamma distribution was assumed and the mean value was used in the condition assessment for the event's frequency. Sensitivity cases were run using the 5th and 95th percentile values for the distributions as the event's frequency.

– ***Seismic event.*** For the condition assessment, the instrument air system was assumed to be unable to withstand any seismic event greater than the operating basis earthquake (0.06 g). The return frequency for seismic events that would result in a loss of instrument air is conservatively estimated at 3.1E-4/yr (3.5E-8/hour), based on a ground acceleration value of 50 cm/sec2 or 0.05 gat Point Beach from NUREG/CR-1488 (Ref. 6). The IPEEE for Point Beach identified the plant high confidence of a low probability of failure (HCLPF) capacity to be 0.16 g (Ref. 7). A sensitivity case was run using the return frequency for 0.15 g, nearest peak ground acceleration given in NUREG/CR-1488 (Ref. 6).

Basic Event	Value	Importance[1] (ΔCDP)	Description
Failure of AFW Pumps			
AFW-PMP-CF-ALL - Common cause failure of all AFW pumps	1.0	7.3E-04	Value used in condition assessment. Assumes all AFW pumps fail when run in deadhead conditions
	0.75	6.1E-04	25% chance at least one AFW pump survives
	0.5	5.0E-04	50% chance at least one AFW pump survives
	0.1	3.1E-04	90% chance at least one AFW pump survives
Initiation of Feed and Bleed Cooling			
HPI-XHE-XM-FB - Operator fails to initiate feed and bleed cooling	2.0E-02	7.3E-04	Probability derived using human error worksheet. (Nominal failure probability for human action is1.0E-3.) Value used in condition assessment
	2.0E-03	5.8E-04	Assumes nominal failure probability for human action is 1.0E-4
	2.0E-04	5.7E-04	Assumes nominal failure probability for human action is 1.0E-5

Basic Event	Value	Importance[1] (ΔCDP)	Description
Loss of Instrument Air			
IE-LOIA	9.0E-07/hr	7.3E-04	Mean. Value used in condition assessment
	3.54E-09/hr	6.0E-04	Lower bound (5%)
	3.4E-06/hr	1.1E-03	Upper bound (95%)
Loss of Service Water			
IE-LOSWS	4.5E-08/hr	7.3E-04	Mean. Value used in condition assessment
	1.83E-10/hr	5.6E-04	Lower bound (5%)
	1.71E-07/hr	1.2E-03	Upper bound (95%)
Loss *of Electric Power*			
IE-LOOP	1.0E-06/hr	7.3E-04	Mean. Value used in condition assessment
	4.0E-09/hr	5.6E-04	Lower bound (5%)
	3.9E-06/hr	1.2E-03	Upper bound (95%)
Seismic Event			
IE-SEISMIC	3.5E-08/hr	7.3E-04	Return frequency for 0.05 g (Ref. 7). Value used for change case in condition assessment
	7.5E-09/hr	5.1E-04	Return frequency for 0.15 g (Ref. 7)

Note:

1. Values given for importance (ΔCDP) are point estimate values for Point Beach Unit 1.

- **Other Considerations**

  In addition to the vulnerability identified in this event, another licensee event report (LER) identifies a potential for complete loss of AFW flow to Unit 1 due to fires in the AFW pump room (LER 266/01-006, Ref. 8) that also existed during the same time period. The likelihood of a damaging fire in the AFW pump room in conjunction with a failure of the fire sprinkler system is small in comparison to the loss of instrument air initiators. Therefore, fire effects were not included in this analysis.

References

1. LER 266/01-005, *PRA Assessment of Auxiliary Feedwater System Reveals Procedural Vulnerability Related to Loss of Instrument Air,* November 29, 2001 (ADAMS Accession No. ML020560352).

2. NRC Inspection Reports No. 50-266/01-17 and No. 50-301/01-17, February21, 2002 (ADAMS Accession No. ML020950889).

3. LER 266/02-003, Possible *Common Mode Failure of AFW due to Partial Clogging of Recirculation Orifices,* December 29, 2002 (ADAMS Accession No. ML030080291).

4. NRC Inspection Report No. 50-266/02-015 and 50-301/02-015, April 2, 2003 (ADAMS Accession No. ML03092011280).

5. Scott T. Beck and Robert F. Buell, *Standardized Plant Analysis Risk Model for Point Beach Units 1 and 2,* Revision 3, Idaho National Engineering and Environmental Laboratory, September 2001.

6. Wisconsin Electric Power Company, *Point Beach Nuclear Plant, Individual Plant Examination of External Events for Sever Accident Vulnerabilities, Summary Report,* 1997.

7. LER 266/01-006, *Appendix R Requirements Not Satisfied for Unanalyzed Fire Induced Damage to the Auxiliary Feedwater System, December* 5, 2001 (ADAMS Accession No. ML020580395).

8. Wisconsin Electric Power Company, Point Beach Nuclear Plant- Units 1 and 2, Individual Plant Examination, revised December 1997.

9. Updated Final Safety Analysis Report for Point Beach.

10. Wisconsin Electric Power Company, Point Beach Nuclear Plant Critical Safety Procedure, CSP-H.1, "Response to Loss of Secondary Heat Sink," Rev. 21 (Unit 1), Rev. 22 (Unit 2), 4/26/2001.

11. C. L. Atwood, et al., *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996,* NUREG/CR-5496, U.S. Nuclear Regulatory Commission, Washington, DG, November 1998.

12. C. L. Atwood, "Constrained Non-informative Priors in Risk Assessment," *Journal of Reliability Engineering and System Safety,* Vol. 53, Issue 1, pp. 37-46, 1996.

13. *Dockets 50-266 and 50-301, Monthly Operating Reports, Point Beach Nuclear Power Plant, Units 1 and 2 (for the calendar month of December 2000),* January 8, 2001 (ADAMS Accession No. ML010180121).

14. *Dockets 50-266 and 50-301, Operating Licenses DPR-24 and DPR-27, Point Beach Nuclear Power Plant, Units 1 and 2, Monthly Operating Reports (for the calendar month of November 2001),* December 7, 2001 (ADAMS Accession No. ML020150475).

15. *Dockets 50-266 and 50-301, Point Beach Nuclear Plant, Units 1 and2, Review of Preliminary Accident Sequence Precursor Analysis of November 2001 Operational Condition (TAC No. MB7832),* letter from A J. Cayia, Site Vice President, Nuclear Management Company, LLC, to the U.S. Nuclear Regulatory Commission, dated May 19, 2003 (ADAMS Accession No. ML0314902530).

Table 1. Conditional probabilities associated with the dominant sequences (Unit 2).[1]

Event Tree Name	Sequence No.	Conditional Core Damage Probability (CCDP)[2]	Core Damage Probability (CDP)[2]	Importance (ΔCDP)[3]
LOIASEISMIC	2	2.9E-04	1.4E-05	-
LOSWS	28	1.9E-04	4.8E-06	-
LOOP	18	1.7E-04	6.2E-07	-
LOIA	20	1.0E-04	1.4E-07	-
Total (all sequences)[4]				
Point Estimate		8.1E-04	2.1E-05	7.9E-04
Means		6.8E-04	1.8E-05	6.6E-04
95th Percentile[5]		1.5E-03	4.7E-05	1.4E-03
5th Percentile[5]		2.2E-05	4.7E-06	1.8E-05

Note:
1. File names: GEM 266-01-005 U1 1-7-2003 174000.wpd (for Unit 1 results) and GEM 266-01-005 U2 1-7-2003 173758.wpd (for Unit 2 results).
2. Core damage probabilities calculated using sequences for only those initiators having a shared dependency with the loss of instrument air. Core damage probabilities for sequences associated with initiators not having a shared dependency were not included.
3. Importance is calculated using the total CCDP and total CDP from all sequences. Sequence level importance measures are not additive.
4. Total CCDP and CDP includes all sequences (including those not shown in this table).
5. Values generated using the uncertainty analysis option in Saphire. Uncertainty method used was Monte Carlo with 8,000 histories.

Table 2a. Event tree sequence logic for the dominant sequences.

Event Tree Name	Sequence No.	Logic ("/" denotes success; see Table 2b for top event names)
LOIASEISMIC	2	SEISMIC
LOSWS	28	AIR-REC-SW-ST, /RT, /RCPSL-SWS, AFW, AIR-REC-SW-MT, AIR-REC-SW-LT
LOOP	18	/RT-L, /EP, AFW, BLEED
LOIA	20	AIR-REC-ST, /RT, AFW, AIR-REC-MT, AIR-REC-LT

Table 2b. Definitions of fault trees listed in Table 2a.

AIR-REC-LT	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN LONG TERM
AIR-REC-MT	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN MEDIUM TERM
AIR-REC-ST	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN SHORT TERM
AIR-REC-SW-	OPERATOR FAILS TO RECOVER SERVICE WATER (SW) TO INSTRUMENT AIR IN LT LONG TERM
AIR-REC-SW-	OPERATOR FAILS TO RECOVER SW TO INSTRUMENT AIR IN MEDIUM TERM MT
AIR-REC-SW-	OPERATOR FAILS TO RECOVER SW TO INSTRUMENT AIR IN SHORT TERM ST
AFW	NO OR INSUFFICIENT AUXILIARY FEEDWATER FLOW
BLEED	FAILURE TO PROVIDE BLEED PORTION OF FEED AND BLEED COOLING SEISMIC EVENT
EP	EMERGENCY POWER SYSTEM FAILURES
RCPSL-SWS	REACTOR COOLANT PUMP SEALS INTACT GIVEN LOSS OF SERVICE WATER
RT REACTOR FAILS TO TRIP DURING TRANSIENT	
RT-L	REACTOR FAILS TO TRIP DURING LOSS OF OFFSITE POWER

Table 3. Conditional cut sets for dominant sequences (Unit 2).

CCDP	Percent contribution	Minimal cut sets1
Event Tree: LOIASEISMIC, Sequence 2		
2.9E-04	100	SEISMIC
Event Tree: LOSWS, Sequence 28		
1.9E-04	100	AIR-XHE-RECOVERY-SW-ST AIR-XHE-RECOVERY-SW-MT AIR-XHE-RECOVERY-SW-LT
Event Tree: LOOP, Sequence 18		
1.7E-04	98.9	HPI-XHE-XM-FB
Event Tree: LOIA, Sequence 20		
1.0E-04	100	AIR-XHE-RECOVERY-ST AIR-XHE-RECOVERY-MT AIR-XHE-RECOVERY-LT

Note:
1. See Table 4 for definitions and probabilities for the basic events.

Table 4. Definitions and probabilities for modified or dominant basic events.

Event Name	Description	Probability/ Frequency	Modified
ACP-XHE-NOREC-BD	OPERATOR FAILS TO RECOVER OFFSITE POWER BEFORE BATTERY DELETION	0.36	YES[1]
AFW-PMP-CF-ALL	COMMON-CAUSE FAILURE OF AFW PUMPS	TRUE	YES[2]
AIR-XHE-RECOVERY-LT	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN LONG TERM GIVEN FAILURE TO RECOVER IN MEDIUM TERM	0.14	NEW[3,4]
AIR-XHE-RECOVERY-MT	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN MEDIUM TERM GIVEN FAILURE TO RECOVER IN SHORT TERM	0.17	NEW[3,4]
AIR-XHE-RECOVERY-ST	OPERATOR FAILS TO RECOVER INSTRUMENT AIR IN SHORT TERM	0.58	NEW[3]
AIR-XHE-RECOVERY-SW-LT	OPERATOR FAILS TO RECOVER SW TO INSTRUMENT AIR IN LONG TERM GIVEN FAILURE TO RECOVER IN MEDIUM TERM	0.68	NEW[3,4]
AIR-XHE-RECOVERY-SW-MT	OPERATOR FAILS TO RECOVER SW TO INSTRUMENT AIR IN MEDIUM TERM GIVEN FAILURE TO RECOVER IN SHORT TERM	0.83	NEW[3,4]
AIR-XHE-RECOVERY-SW-ST	OPERATOR FAILS TO RECOVER SW TO INSTRUMENT AIR IN SHORT TERM	0.88	NEW[3]
HPI-XHE-XM-FB	OPERATOR FAILS TO INITIATE FEED AND BLEED COOLING	2.0E-02	NO
IE-LDC01	LOSS OF DC BUS INITIATING EVENT	FALSE	YES[5]
IE-LLOCA	LARGE LOSS-OF-COOLANT ACCIDENT (LOCA) INITIATING EVENT	FALSE	YES[5]
IE-LOCCW	LOSS OF COMPONENT COOLING WATER (LOCCW) INITIATING EVENT	FALSE	YES[5]
IE-LOIA	LOSS OF INSTRUMENT AIR INITIATING EVENT	9.0E-07/hr	YES[1]
IE-LOOP	LOSS OF OFFSITE POWER INITIATING EVENT	1.0E-06/hr	YES[6]
IE-LOSWS	LOSS OF SERVICE WATER INITIATING EVENT	4.5E-08/hr	YES[1]
IE-MLOCA	MEDIUM LOCA INITIATING EVENT	FALSE	YES[5]
IE-RHR-DIS-V	RHR DISCHARGE VALVE INTERSYSTEM LOCA (ISLOCA) INITIATING EVENT	FALSE	YES[5]
IE-RHR-SUC-V	RHR SUCTION VALVE ISLOCA INITIATING EVENT	FALSE	YES[5]
IE-SI-CLDIS-V	SAFETY INJECTION (SI) COLD LEG ISLOCA INITIATING EVENT	FALSE	YES[5]
IE-SI-HLDIS-V	SI HOT LEG ISLOCA INITIATING EVENT	FALSE	YES[5]
IE-SEISMIC	SEISMICALLY INDUCED LOSS OF INSTRUMENT AIR	1.7E-09/hr (base) 3.5E-08/hr (change)	YES[7]
IE-SLOCA	SMALL LOCA INITIATING EVENT	FALSE	YES[5]
IE-STGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	FALSE	YES5

Event Name	Description	Probability/ Frequency	Modified
IE-TRANS	TRANSIENT INITIATING EVENT	FALSE	YES[5]
RCS-MDP-SEALS	REACTOR COOLANT PUMP (RCP) SEALS FAIL W/O COOLING AND INJECTION	0.22	YES[1]
RCS-MDP-SEALS2	RCP SEALS FAIL W/O COOLING AND INJECTION GIVEN LOSWS OR LOCCW	0.22	YES[1]

Notes:
1. Basic event/initiating event frequency to the base case model updated. See Attachment 5 for details.
2. Event changed to reflect event being analyzed.
3. Basic event added to update base case model. See Attachment 5 for details.
4. Conditional probability.
5. Initiating event frequencies were set to FALSE in the base case and the change case to reflect the condition being analyzed. The sequences associated with these initiating events have no shared dependencies with loss of instrument air; therefore, including them in the CDP and CCDP calculations with the common-cause failure of all AFW pumps (AFW-PMP-CF-ALL) to TRUE is not appropriate.
6. Initiating event frequency updated for event being analyzed. See Attachment 3 for event analysis and frequency calculation.
7. Initiating event frequency updated for event being analyzed. The return frequency for seismic events that would result in a loss of instrument air is conservatively estimated at 3.1E-4/yr based on the lowest estimated ground acceleration value (50 cm/sec^2 at Point Beach from NUREG/CR-1488, *Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains* (Ref. 6). Base case value of 1E-09/hr (1.5E-05/yr) was taken from Point Beach Units 1 and 2 Individual Plant Examination (Ref. 9).

Figure removed during SUNSI review

Figure 1 AFW System Simplified Diagram

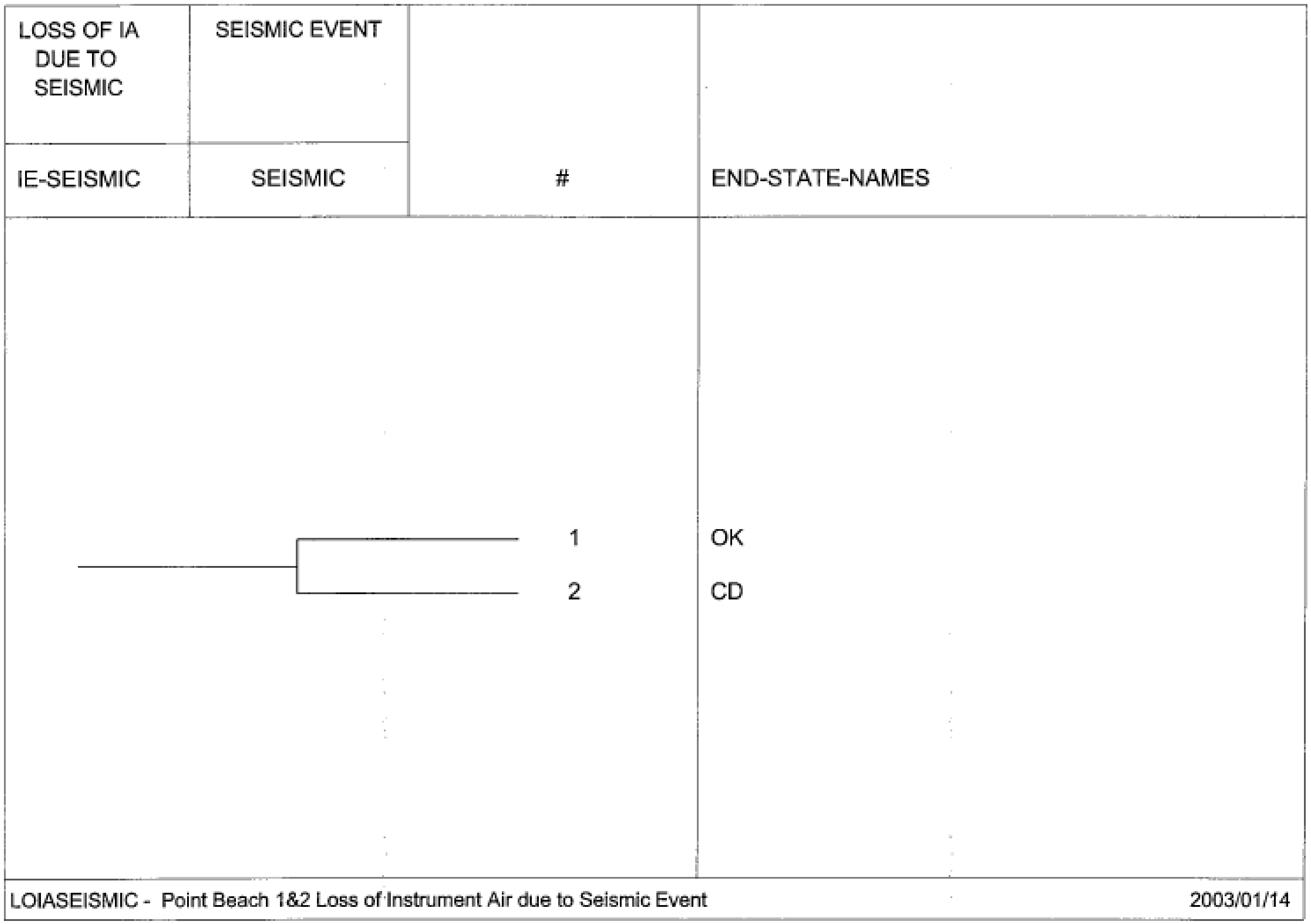

Figure 2 Seismically Induced Loss of Instrument Air Event Tree

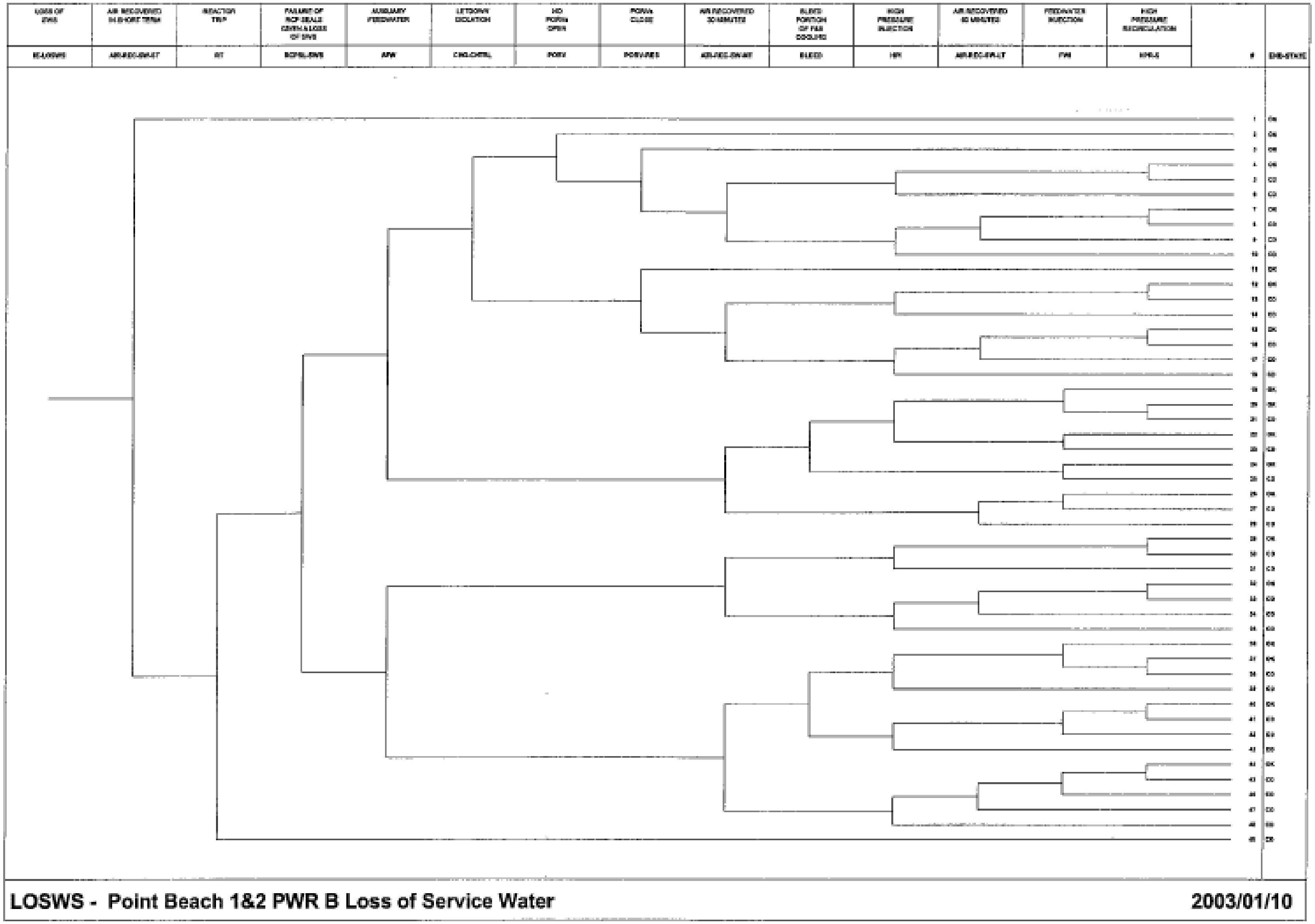

Figure 3 Loss of Service Water Event Tree

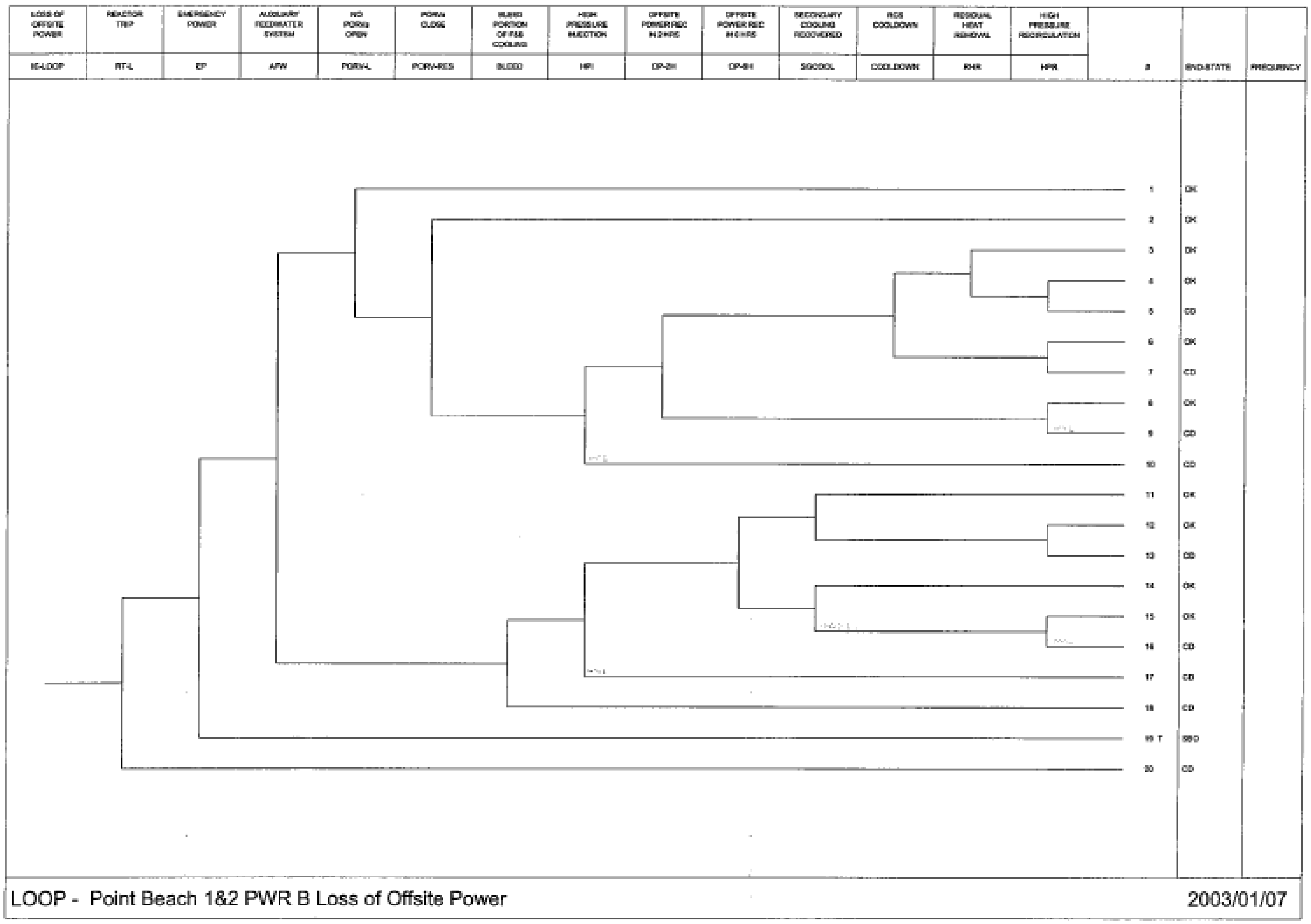

Figure 4 Loss of Offsite Power Event Tree

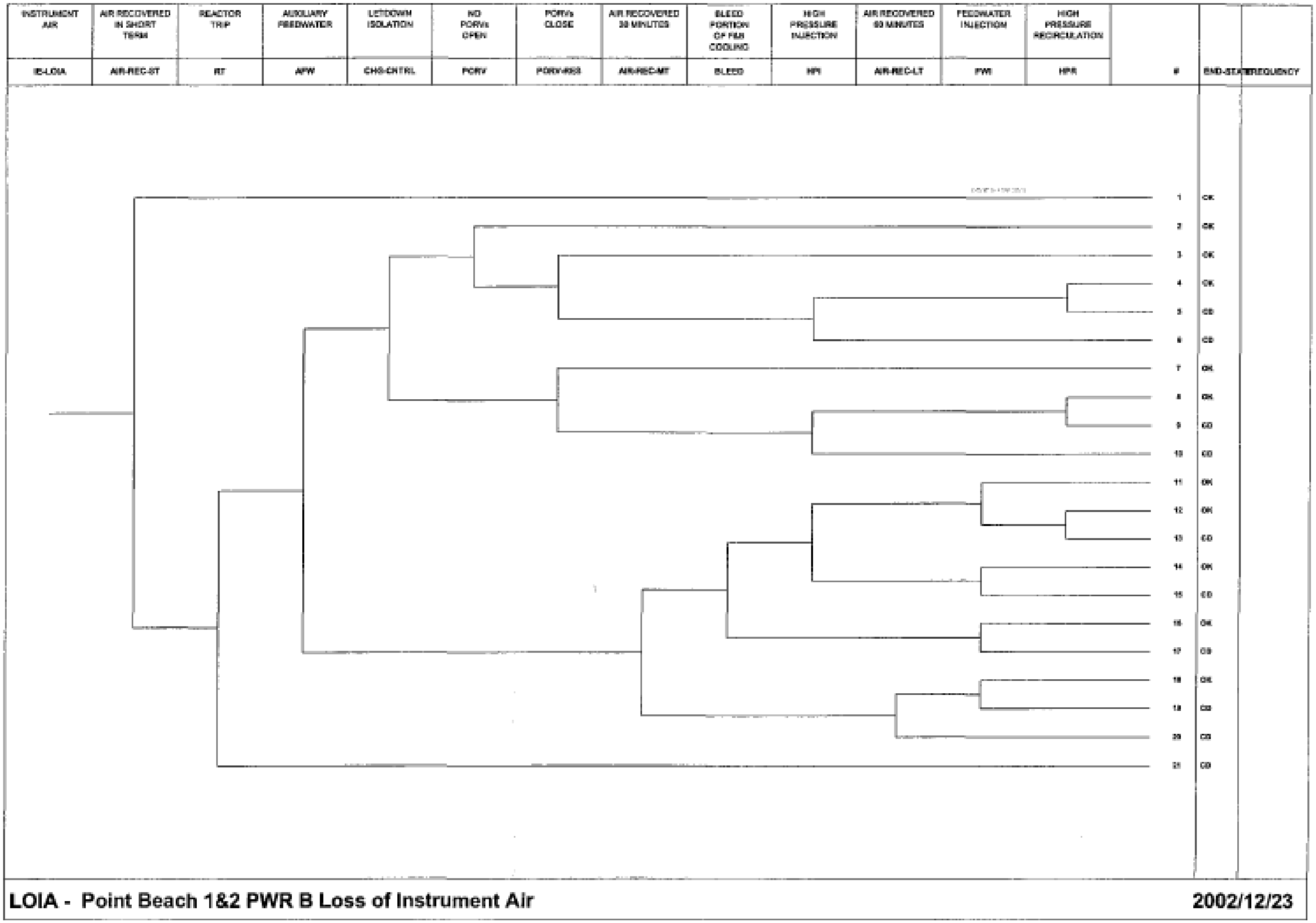

Figure 5 Loss of Instrument Air Event Tree

Figure removed during SUNSI Review

Figure 6 Instrument Air Simplified Flow Diagram

Figure removed during SUNSI Review

Figure 7 Service Air Simplified Flow Diagram

Attachment 1 - Plant-specific System and Operational Considerations

These are the facts upon which assumption and model modifications are based. Details are provided for the following:

- AFW system design
- Feed and bleed cooling design
- Instrument air and service air system designs
- Control room indications
- Plant response to loss of instrument air
- AFW flow control
- Recovery of main feedwater

Information removed during SUNSI Review

Information removed during SUNSI Review

Information removed during SUNSI Review

Attachment 2- Details of Important Assumptions

This analysis includes several important assumptions. The assumptions and the bases for making these assumptions are described below.

- ***Operators fail to recognize that the recirculation valves are closed.*** This assumptions is based on the following:

  – EOP-0.1, Reactor *Trip Response,* does not provide any steps to caution the operator about the damage to AFW pumps during deadheading conditions-pump's minimum flow recirculating valves closed due to the loss of instrument air and the closure of the AFW flow control valves.

  EOP-0.1, *Reactor Trip Response,* directed operators to control feedwater flow early in the procedure. EOP-0.1 was the procedure that operators would use for most transients. Response not obtained (RNO) column (Step 1.c of the procedure) directed operators to reduce feed flow if RCS temperatures were less than 547 degrees C) Fahrenheit (F) and trending lower. Step 4.b directed operators to control feed flow to maintain steam generator levels between 29% and 69%. RNO (Step 4.b) directed operators to stop feed flow to intact steam generators if level continued to rise. If instrument air had been lost, damage would occur to the AFW pumps by these operator actions to control feedwater flow due to the low-flow conditions created. The team noted that procedure OM 4.3.1, *AOP and EOP Writers' Guide, Step* 5.4.2, stated, "A caution is used to present information regarding potential hazards to personnel or equipment associated with the subsequent step(s)." The emergency operating procedures steps did not provide any such cautions prior to November 30, 2001.

  – The time that the AFW recirculation valves would fail closed due to the loss of instrument air could vary; however, time is on the order of 10 minutes or less.

  Based on discussions with licensee engineering staff, the team determined that the time that the AFW recirculation valves would fail closed due to loss of instrument air could vary. The engineering staff had determined that the recirculation valves would begin to drift shut when instrument air header pressure was reduced to 40 psig and would be fully closed at 25 psig. The instrument air header pressure was nominally maintained at 100 psig with some variation due to cycling of air compressors. Based on observations of instrument air header pressure drop between cycling of air compressors, the engineering staff determined that the instrument air head pressure would drop approximately 13.5 pounds per square inch in 1 minute under normal loads. The engineering staff estimated that the AFW recirculation valves would begin to drift shut approximately 6 to 8 minutes after loss of all air compressors with complete valve closure 1 to 2 minutes thereafter. A loss of instrument air due to a leak in an airline versus a loss of air compressors would result in different bleed down rates, depending on the size of the break. Additionally, the instrument air bleed down rate could be faster due to greater demands on the instrument air system in response to the transient.

  – The AFW recirculation valves could reposition at a time when an operator's attention would not be directly focused on the AFW pumps.

  Based on discussions between the NRC and licensee personnel, as documented in the inspection report (Ref. 2), the preferred method for controlling AFW flow was by throttling or closing the AFW flow control valves (for the motor-driven AFW pumps) or discharge valves (for the turbine-driven AFW pumps) rather than securing the pumps. Section 14.1.12, "Loss of All AC Power to the Station Auxiliaries," of the original Final Facility Description and Safety Analysis Report (FFDSAR)" stated, "The reactor operator in the control room can monitor the steam generator water level and control the feedwater flow with remote-operated AFW control valves." The FFDSAR did not discuss securing AFW pumps as a means to control steam generator levels. In some loss of instrument air scenarios (e.g., those involving RCS overcooling), the recirculation valves could remain open at the time that operators throttle or close flow control and discharge valves due to remaining air header pressure. However, the recirculation valves would subsequently close due to decreasing air pressure. Consequently, the valves could reposition at a time when an operator's attention would not be directly focused on the AFW pumps (Ref. 2).

– Abnormal Operating Procedure (AOP)-58, *Loss of Instrument Air,* had steps that addressed the recirculating valves; however, the guidance appeared deep into the procedure.

  Procedure AOP-58 provided operators with guidance for loss of instrument air. However, during these transients, operators would typically be using emergency operating procedures, such as EOP-0.1, in their initial response to a transient. After plant conditions stabilized, abnormal operating procedures, such as AOP-58, would be used to restore equipment. AOP-58 has steps to secure open the AFW pump recirculation valves. However, guidance on securing open the valves does not appear until Step 1 of Attachment R, "Auxiliary Feed," located on page 36 of the procedure. Operators were directed to Attachment R by Step 26 (located on page 14) of the procedure. Step 26 simply directed operators to check plant systems status per attachments A through Z. Consequently, although procedure AOP-58 had steps that addressed the failed closed recirculation valves, operators would likely not reopen the recirculation valves before damage occurred to the AFW pumps because they would be following the emergency operating procedures (Ref. 2).

- ***Operators close the discharge valves for all of the AFW pumps resulting in deadheading of all AFW pumps.*** This assumptions is based on the following:

  – As discussed above, the preferred method for controlling AFW flow was by throttling or closing the AFW flow control valves (for the motor-driven AFW pumps) or discharge valves (for the turbine-driven AFW pumps) rather than securing the pumps.

  – As discussed above, EOP-0.1 did not provide guidance on how to reduce AFW flow.

  – Operating experience demonstrated that operators would drastically reduce AFW flow within several minutes of pump start due to RCS overcooling under some transient conditions.

  For example, on June 27, 2001, the Unit 2 reactor was manually tripped due to low and decreasing water level in the Unit 2 circulating water pump bay (reported in LER 05000301/2001-002-00). Due to subsequent low steam generator water levels, the Unit 2 turbine-driven AFW pump and both motor-driven AFW pumps initiated and began feeding the Unit 2 steam generators. Only one steam generator in a unit nominally requires 200 gpm feedwater flow for decay heat removal. However, with three AFW pumps running, approximately 800 gpm of feedwater flow (approximately four times the required flow) was provided to the Unit 2 steam generators. Consequently, the reactor coolant system was cooled down at an excessive rate. Approximately 3 minutes after the reactor was tripped, operators closed either the flow control valves or the discharge valves to stop flow from the motor-driven AFW pumps. Approximately 4 minutes after the reactor was tripped, operators closed the discharge valves from the Unit 2 turbine-driven AFW pump, stopping all AFW flow to the steam generators. The AFW pumps were not secured until approximately 8 minutes after the reactor was tripped when feed flow using main feedwater was partially restored. In this particular event, the AFW recirculation valves were functional because instrument air had not been lost. However, had instrument air not been available, as would happen in transients such as loss of instrument air, loss of offsite power, and loss of service water events, all AFW pumps could have been damaged (Ref. 2).

- ***No credit for operators detecting pump deadhead conditions (i.e., closed recirculation valves) and taking corrective actions to protect one or more AFW pumps.*** This assumption is based on the following:

  – As discussed above, EOP-0.1 does not caution operators about the potential to damage the AFW pumps during deadhead conditions.

  – Operators have no indication of flow in the AFW pumps' recirculation lines. Indication is provided for AFW flow to individual steam generators and flow from each AFW pump. However, the flow element for each AFW pump is located downstream of where the recirculation line branches off from the pump's discharge line. Therefore, indications of little or no flow for the AFW pumps would be expected with the pumps discharge or flow control valves throttled or closed (Ref. 2).

  – As discussed above, AOP-58 had steps that address the failed closed recirculation valves; however, operators would likely not get to these steps until the AFW pumps had operated in deadhead conditions and damage occurred.

- ***No credit for leakage past either the closed recirculation valves or the closed discharge or flow control valves providing adequate flow through the AFW pumps to prevent pump damage.***

  – In 1988, NMC installed modifications to increase the design minimum recirculation flow for the AFW pumps to 70 gallons per minute (gpm) for the motor-driven pumps and 100 gpm for the turbine-driven pumps. Previously, the minimum recirculation flow was 30 gpm, which the AFW pump vendor, Byron Jackson, indicated would be sufficient to prevent pump damage, based on pump heat up when on recirculation flow (Ref. 2). Leakage past the AFW pump's closed recirculation valve or the closed discharge or flow control valve could provide enough flow to prevent pump damage. However, inclusion of component failures as success logic in a risk model is typically not done. The failure probability for an air-operated valve failing to close on demand is 1.0E-3; the failure probability for a motor-operated valve failing to close on demand is 3.0E-3. The likelihood that the air- operated flow control valve or air-operated recirculation valve for one of the motor-driven AFW pumps fails to close on demand or the motor-operated discharge valve or air- operated recirculation valve for the turbine-driven pump fails to close is 8.0E-3. Unless leakage past these valves, sufficient to prevent pump damage, normally occurs, the likelihood that the valves close when demanded, resulting in pump deadhead conditions, is 0.992.

- ***No credit for recovery of AFW pumps given failure due to deadheading conditions.*** This assumption is based on the following:

– The AFW recirculation lines were installed as part of original construction to ensure the pumps would have a flow path to prevent deadheading the pump, which would damage the pump. As indicated in the inspection report (Ref. 2), discussions with licensee engineering staff indicated that a pump could be damaged within minutes under insufficient flow condition due to lack of cooling.

– Damage to the pump and pump seals would be catastrophic.

- ***No credit for the recovery of nitrogen air bottles to the pressurizer PORVs.*** This assumption is based on the following:

  – The pressurizer PORVs are air-operated valves with a backup nitrogen supply. However, since 1979, the backup nitrogen supply has been isolated, by procedure, during power operation. A containment entry is required to restore the backup nitrogen supply. Consequently, upon a loss of instrument air, the PORVs would not be available (Ref. 2).

  – Containment entry during any one of the LOIA initiators would not be a normal plant evolution (would be considered a heroic action). Further, EOP-0.1 and CFP-H.1 do not provide the steps and cautions for such action.

- ***No credit for recovery of secondary cooling without instrument air or service water.*** This assumptions is based on the following:

  – Critical Safety Procedure (CSP) - H.1, *Response to Loss of Secondary Heat Sink,* provides instructions for restoring secondary heat sink (Ref. 11). Because several valves in the secondary side system are air operated and because specific procedural guidance for restoring secondary side cooling when instrument air is not available is NOT provided in CSP-H.1, no credit for recovery of secondary cooling is taken when instrument air (or service water cooling to the instrument air or service air compressors) is not available.

Attachment 3 - LOOP Initiating Event Frequency Estimate

- ***Data sources.*** For this condition assessment, a frequency estimate for loss of instrument air due to LOOP events was developed that is based on events identified in NUREG/CR-5496, *Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980- 1996* (Ref. 12), and updated to include LER data through 2002. A search of the Sequence Coding and Search System database was conducted to select LERs involving failures in the instrument air system for the years 1997 through 2002. The total time period reviewed is 1987–2002.

- ***Review criteria.*** Because of the design for the instrument air and service air systems at Point Beach (redundant compressors and diverse power sources), loss of all four compressors due to a LOOP will only occur if power is lost to both units. The types of LOOP events that would involve both units include dual-unit, plant-centered LOOPs; grid-related LOOPs; and severe weather-related LOOPs. Other review criteria include the following:

  – Causes of weather-related and grid-related LOOP events are independent of plant mode; therefore, both operating and shutdown experience were included.

  – LOOP events that occurred when all units at the site were shut down were not included.

  – LOOP caused by outage maintenance activity on one shutdown unit (even though the activity is not performed while the plant is operating) were included. This type of LOOP will be used to calculate a dual-unit, plant-centered LOOP frequency for the fraction of time that one unit at Point Beach is shut down.

  – Hurricane-related LOOP events were not included.

- ***Results.*** The results of the review of LOOP events during the 1987–2002 period are given in Table 3.1 below.

Table 3.1. Events selected for dual-unit LOOP frequency assessment.

LOOP Type	No. Events	LER(s)
Grid-related	1	395/89-012
*Weather-related	5	333/88-011, 282/96-012, 346/98-006, 302/93-002, and 325/93-008
Dual-unit, plant-centered		
Both units operating	2	317/87-012 and 327/92-027
One unit shut down	1	334/93-013
*Exclude: Pilgrim (outlier from NUREG/CR-5496); 2 of 3 events at Crystal River (302/93-002) caused by the same storm; hurricane events when plant was shut down prior to the hurricane-induced LOOP.		

- ***Industry frequency calculation.*** The LOOP frequency is estimated by:

$$F_{LOOP} = F_{Grid} + F_{Severe\ Weather} + F_{Dual}$$

Where,

F_{Grid} =frequency of grid-related LOOPs
$F_{Severe\ Weather}$ =frequency of weather-related LOOPs
F_{Dual} = frequency of plant-centered, dual-unit LOOPs

The total operating and shutdown time for all sites (single and multiunit sites) during 1987–2002 is 1,080 site calendar years, as shown in Table 3.2. The operating and shutdown time for only multiunit sites during the same time is 570.9 site calendar years. Using the criticality factor calculated in Table 3.3 of 0.78, the multiunit critical time is 0.78 × 570.9 calendar years = 445 critical years. Therefore, the mean frequency is:

F_{Grid} = 1/1,018 yr = 9.8E-4/yr or 1.1E-7/hr

$F_{Severe\ Weather}$ = 5/1,018 yr = 4.9E-3/yr or 5.6E-7/hr

F_{Dual} = 3/445 yr = 6.7E-3/yr or 7.7E-7/hr

The industry LOOP frequency (per site calendar year) is:

F_{LOOP} = 1.1E-7/hr + 5.6E-7/hr +7.7E-7/hr = 1.4E-6/hr or 1.2E-2/yr

- ***Point Beach plant-specific frequency calculation.*** In order to obtain a rigorous probability distribution for F_{LOOP} a numeric analysis of each parameter would be required. Since the number of events controls the uncertainty bounds, a reasonable distribution can be created from an approximate analysis for the purpose of ASP uncertainty analysis. The number of LOOP events (nine) and the industry LOOP frequency (1.2E-2/yr) are used to estimate a pseudo-exposure (732 years) so that a probability distribution can be created to express the uncertainty in the estimate.

The constrained noninformative prior distribution (Ref. 13) was used. The distribution is given by:

$$Gamma(\alpha, \beta) \;= Gamma(\,0.5, \frac{1}{2F})$$

Grid reliability and severe weather frequency vary between plants, so the more diffuse prior distribution is appropriate. The Gamma distribution parameters (in years) of the prior are α=0.5 and β=41. Performing a Bayesian update on the above distribution with Point Beach's 16 operating years without a LOOP event, the mean LOOP frequency for Point Beach is 8.8E-3/yr or 1.0E-6/hr. The Gamma distribution parameters of the posterior are α=0.5 and β=57. The 5th percentile of this distribution is 3.5E-5/yr and the 95th percentile is 3.4E-2/yr.

Operating history at Point Beach for the time period from November 30, 2000, to November 29, 2001 (approximates the condition duration), shows that Unit 1 was critical for a total of 7,680 hours (for a criticality factor of 0.88) and shut down for a total of 1,080 hours;

and Unit 2 was critical for a total of 8,316 hours (for a criticality factor of 0.95) and shut down for a total of 444 hours (Refs. 14 and 15).

Table 3.2. Commercial site calendar years- calendar years 1987-2002.[1]

Plant Name	Multi-unit Sites[2] (site calendar years)		All Sites[3] (site calendar years)	
	1987–1995 (9 yrs)	1996–2002 7 yrs)	1987–1995 (9 yrs)	1996–2002 (7 yrs)
Arkansas 1			9	7
Arkansas 2	9	7	—	—
Beaver Valley 1			9	7
Beaver Valley 2	8.4	7	—	—
Big Rock Point			9	2
Braidwood 1			8.6	7
Braidwood2	7.8	7	—	—
Browns Ferry 1			—	—
Browns Ferry 2			9	7
Browns Ferry 3	9	7	—	—
Brunswick 1			9	7
Brunswick2	9	7	—	—
Byron 1			9	7
Byron 2	9	7	—	—
Callaway			9	7
Calvert Cliffs 1			9	7
Calvert Cliffs 2	9	7	—	—
Catawba 1			9	7
Catawba2	9	7	—	—
Clinton 1			8.8	7
Columbia			9	7
Comanche Peak 1			5.8	7
Comanche Peak 2	2.8	7	—	—
Cook 1			9	7
Cook 2	9	7		
Cooper Station			9	7
Crystal River 3			6	7
Davis-Besse			9	7
Diablo Canyon 1			9	7
Diablo Canyon 2	9	7	—	—
Dresden 2			9	7
Dresden 3	9	7	—	—
Duane Arnold			9	7
Farley 1			9	7
Farley2	9	7	—	—
Fermi2			9	7

Plant Name	Multi-unit Sites[2] (site calendar years)		All Sites[3] (site calendar years)	
	1987–1995 (9 yrs)	1996–2002 7 yrs)	1987–1995 (9 yrs)	1996–2002 (7 yrs)
Fitzpatrick			9	7
Fort Calhoun			9	7
Fort St. Vrain			2.7	0
Ginna			9	7
Grand Gulf			9	7
Haddam Neck			9	1.2
Harris			9	7
Hatch 1			9	7
Hatch 2	9	7	—	—
Hope Creek			9	7
Indian Point 2			9	7
Indian Point 3	9	7	—	—
Kewaunee			9	7
Lacrosse			0.4	0
LaSalle 1			9	7
LaSalle 2	9	7	—	
Limerick 1			9	7
Limerick2	6.4	7	—	—
Maine Yankee			9	1.7
McGuire 1			9	7
McGuire 2	9	7	—	—
Millstone 1			—	—
Millstone 2			9	7
Millstone 3	9	7	—	—
Monticello			9	7
Nine Mile Pt. 1			9	7
Nine Mile Pt. 2	8.6	7	—	—
North Anna 1			9	7
North Anna2	9	7	—	—
Oconee 1			9	7
Oconee 2			—	—
Oconee 3	9	7	—	—
Oyster Creek			9	7
Palisades			9	7
Palo Verde 1			9	7
Palo Verde 2	9	7	—	—
Palo Verde 3			—	—
Peach Bottom 2			9	7
Peach Bottom 3	9	7	—	—
Perry			9	7

Plant Name	Multi-unit Sites[2] (site calendar years)		All Sites[3] (site calendar years)	
	1987–1995 (9 yrs)	1996–2002 7 yrs)	1987–1995 (9 yrs)	1996–2002 (7 yrs)
Pilgrim			9	7
Point Beach 1			9	7
Point Beach 2	9	7	—	—
Prairie Island 1			9	7
Prairie Island 2	9	7	—	—
Quad Cities 1			9	7
Quad Cities 2	9	7	—	—
Rancho Seco			2.4	0
River Bend			9	7
Robinson 2			9	7
Salem 1			9	7
Salem 2	9	7	—	—
San Onofre 1			—	—
San Onofre 2			9	7
San Onofre 3	9	7	—	—
Seabrook			6.6	7
Sequoyah 1			9	7
Sequoyah 2	9	7	—	—
South Texas 1			7.8	7
South Texas 2	6.5	7	—	—
St. Lucie 1			9	7
St. Lucie 2	9	7	—	—
Summer			9	7
Surry 1			9	7
Surry2	9	7	—	—
Susquehanna 1			9	7
Susquehanna 2	9	7	—	—
Three Mile Island 1			9	7
Trojan			6	0
Turkey Point 3			9	7
Turkey Point 4	9	7	—	—
Vermont Yankee			9	7
Vogtle 1			8.8	7
Vogtle 2	6.8	7	—	—
Waterford 3			9	7
Watts Bar 1			0	6.6
Wolf Creek			9	7
Yankee-Rowe			5.2	0
Zion 1			9	2.4
Zion2	9	1.6	—	—

Plant Name	Multi-unit Sites[2] (site calendar years)		All Sites[3] (site calendar years)	
	1987–1995 (9 yrs)	1996–2002 7 yrs)	1987–1995 (9 yrs)	1996–2002 (7 yrs)
SUBTOTALS	317.3	253.6	618.1	461.9
TOTALS (site calendar years)				
Multiunit sites	570.9			
All sites			1080	

Notes:

1. Sources: CY 1987–1995 from NUREG/CR-5750; CY 1996-2002 from "Precursors to Potential Severe Core Damage Accidents-Fiscal Year 1999," Appendix C, ADAMS Accession No. ML0216801631. CY 1996–2002 see Table 3.4.
2. For site calendar years for multiunit sites, only sites having more than one operating unit were included (single-unit sites were excluded). Site calendar time was based on time when second unit began operations.
3. For all site calendar years, each site is counted once. For multiunit sites, site calendar time is based on time when first unit began operations.

Table 3.3. Industry average criticality factor- calendar years 1987–2001.

Year	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001
Critical Hours[1,2]															
PWR	417775.7	466182.3	461652.3	474942.9	504981.8	512763.6	491488.7	518225.2	518681.1	515809.3	463214.3	499729.5	529114.5	538829.8	546269.5
BWR	197489.5	199293.3	204484.8	231608.8	230335.2	221641.0	234735.5	233389.0	259566.2	249177.9	236965.5	239544.1	265672.3	277399.2	276843.9
TOTAL	615265.2	665475.6	666137.1	706551.7	735317.0	734404.6	726224.2	751614.2	778247.3	764987.2	700179.8	739273.6	794786.8	816229.0	823113.4
Critical Years[1,2]															
PWR	47.69	53.07	52.70	54.22	57.65	58.37	56.11	59.16	59.21	58.72	52.88	57.05	60.40	61.34	62.36
BWR	22.54	22.69	23.34	26.44	26.29	25.23	26.80	26.64	29.63	28.37	27.05	27.35	30.33	31.58	31.60
TOTAL	70.24	75.76	76.04	80.66	83.94	83.61	82.90	85.80	88.84	87.09	79.93	84.39	90.73	92.92	93.96
Calendar Years[2,3]															
TOTAL	105.34	108.34	110.40	111.89	112.00	111.37	109.91	110.00	110.00	108.50	106.50	104.20	104.00	104.00	104.00
Criticality factor															
TOTAL	0.67	0.70	0.69	0.72	0.75	0.75	0.75	0.78	0.81	0.80	0.75	0.81	0.87	0.89	0.90
TOTALS															
	Critical Years	**Calendar Years**	**Criticality Factor**												
TOTAL	1256.81	1620.45	0.78												

Notes:

1. Data from Idaho National Engineering and Environmental Laboratory's database (MORP1.DBF) which is based on licensee's monthly operating reports as of December 2002.
2. Data are included from critical date until permanent shutdown. Ft. St. Vrain critical hours are excluded.
3. Data from NUREG/CR-5750 for CY 1987-1995 (Ref.). Data calculated for CY 1996-2001; see Table 3-4.)

Table 3.4. Data used to calculate reactor calendar years (CY 1996–2002).

		Calendar Days						
		1996	**1997**	**1998**	**1999**	**2000**	**2001**	**2002**
Decommissioned PWRs	**Defuel Date[1]**							
San Onofre 1	11/30/92	0	0	0	0	0	0	0
Trojan	11/09/92	0	0	0	0	0	0	0
Haddam Neck	12/5/96	339	0	0	0	0	0	0
Maine Yankee	6/23/97	365	174	0	0	0	0	0
Zion 1	4/28/97	365	118	0	0	0	0	0
Zion 2	2/26/98	365	365	57	0	0	0	0
Initial Startup-PWRs	**Startup Date[1]**							
ComanchePeak2	8/3/93	365	365	365	365	365	365	365
Watts Bar 1	5/27/96	147	365	365	365	365	365	365
Operating-PWRs267 units x 365 days =		24455	24455	24455	24455	24455	24455	24455
Total PWR (reactor calendar years)		72.33	70.80	69.16	69.00	69.00	69.00	69.00
Decommissioned BWRs	**Defuel Date[1]**							
Big Rock	9/22/97	365	264	0	0	0	0	0
Millstone 1	11/19/95	50	0	0	0	0	0	0
Operating-BWRs35 units x 365 days =		12775	12775	12775	12775	12775	12775	12775
Total BWR (reactor calendar years)		36.14	35.72	35.00	35.00	35.00	35.00	35.00
TOTAL (PWR + BWR)		108.5	106.5	104.2	104.0	104.0	104.0	104.0

Notes:
1. Startup date from NUREG-1350, "Information Digest." Defuel date from the NRC Status Reports. Dates for San Onofre 1 and Trojan are shutdown dates from NUREG-1350. Defuel date for Millstone 1 from letter—Northeast Nuclear Energy to NRC dated 7/21/98.
2. Number of plants in operation (not shut down for decommissioning) during the end of FY-02 minus new plants that were started during the period.

Attachment 4 - Resolution of Comments

A letter from Nuclear Management Company, LLC (NMC) to the NRC dated May 19, 2003 (Ref. 16), describes NMC's review of and comments on the Preliminary Precursor Analysis of the condition reported in LER 266/01-005. The NRC has reviewed these comments and has the following responses:

Licensee's Comment 1: Page 1 - AFW pumps are listed as P39A and P39B. The correct designations are P38A and P38B.

Response: Text corrected to show correct pump designations.

Licensee's Comment 2: Page 2; 2"d paragraph - The "Importance" section states that, "the pumps' discharge valves fail closed..." following loss of instrument air (IA). The phrase should state that "the pumps' recirculation valves fail closed..."

Response: Text corrected to indicate that AFW pumps' recirculation valves fail closed on loss of instrument air.

Licensee's Comment 3: Page 3- "Seismic event": States that IA was assumed failed due to soldered joint failure. The analysis was not this detailed. The assumption was actually based upon the vast piping network that went through non-seismic structures that include block walls. This comment also applies to page 5 under "Sequence of interest".

Response: Text modified to delete reference to "soldered joint failure."

Licensee's Comment 4: Page 6; 6th bulleted item - Operators fail to recover instrument air pressure in time before initiation of feed and bleed should include "seismic" as an initiator, but should not include "loss of offsite power" (LOOP).

Response: Text for this bulleted item modified to clarify conditions. Text added to indicate that for seismic events, the resulting damage to the instrument air system is assumed to not be recoverable. Text also added to clarify that for LOOP, non-SBO sequences, operators must manually restore electric power to the instrument air or station air compressors and instrument air pressure recovered before feed and bleed cooling can be initiated. In the LOOP event tree, this action is assumed to be part of the initiation of the "bleed" portion of feed and bleed cooling.

Licensee's Comment 5: Page 6; 7th bulleted item - Should read, "Main Feed Water not available with a loss of Service Water (SW) due to its dependency on Service Water in addition to the subsequent loss of IA with loss of SW."

Response: Text for this bulleted item modified to clarify conditions. Text added to indicate that main feedwater is unavailable and instrument air pressure and/or service water flow must be recovered to restore secondary cooling.

Licensee's Comment 6: Page 7; 4th bulleted item - Should read, "No credit for recovery of secondary cooling without instrument air or service water."

Response: Text for this bulleted item and text on page 27 in Attachment 2 were corrected to include need for service water.

Licensee's General Comment: In both the PBNP and NRC ASP analyses, no credit was taken for operators discovering the closed recirculation valve. This remains a bounding assumption in that some uncertainty remains in the operators' ability to diagnose the pump failure and take action to prevent additional pump failure. Factors affecting this are the short duration time between pump failures and the high stress following the first and second pump failures.

Response: Agreed. Additional text added to "Important assumptions" section (page 7) and in Attachment 2, "Details of Important Assumptions," describing the assumption that operators fail to detect pump deadhead conditions in time to prevent damage to one or more AFW pumps. This assumption is necessary because an accurate estimate of the likelihood of the operator failing to diagnose pump deadhead conditions in time to save one or more AFW pumps cannot be made. Although this assumption is conservative, it may not be overly conservative

A memo from Cynthia D. Pederson, Director, Division of Reactor Safety (DRS), Region III, to Patrick Baranowsky, Chief, Operating Experience Risk Analysis Branch (OERAB), Division of Risk Analysis and Application (DRAA), Office of Nuclear Regulatory Research (RES) (dated May 22, 2003), provided review comments on the Preliminary Precursor Analysis of the condition reported in LER 266/01-005. The comments have been reviewed and following responses provided:

DRS Region III General Comment 1: The analysis report needs to consistently state that the Point Beach preferred method for controlling AFW flow was by throttling or closing the AFW flow control valves (for the motor-driven AFW pumps) or discharge valves (for the turbine-driven pump) rather than securing the pumps. Many sections of the report merely state that operators would close the AFW pumps' discharge or flow control valves. Throttling the valves would yield the same insufficient recirculation flow to the pumps.

Response: Agreed. Text modified to indicate that operators could throttle AFW pumps' flow control valves as well as close the valves.

DRS Region III General Comment 2: For the loss of service water initiating event, only the TDAFW pump is assumed available. Both MDAFW and TDAFW pumps require service water for bearing cooling, but fire water is automatically supplied to the TDAFW pump. It wasn't clear in the report whether the ASP analysis credited only the TDAFW pump.

Response: Section 10.2.2. of the Point Beach FSAR indicates that the motor-driven AFW pumps' bearing oil is cooled by service water. For the turbine-driven AFW pumps, both the turbine and pump are normally cooled by service water with an alternate source of cooling water from the firewater system. During discussions between John Schroeder of Idaho National Engineering and Environmental Lab and Paul Knoespel of NMC, Mr. Knoespel indicated that the motor-driven pumps will operate satisfactorily without service water cooling. Therefore, the SPAR model correctly credits the motor-driven pumps during a loss of service water event.

DRS Region III Specific Comment 1: Page 1, second paragraph, last sentence should read, *"Prior to November 30, 2001,* there were no backup air or nitrogen accumulators associated with these recirculation valves." Since the identification of this design deficiency, the licensee has installed a backup air source for the recirculation valves. The ASP report makes a similar statement in Attachment 2 when discussing the EOPs that were subsequently changed.

Response: Agreed. Text added to indicate that prior to the discovery of this design deficiency, no backup air or nitrogen accumulators were associated with the AFW pumps' recirculation valves.

DRS Region III Specific Comment 2: Page 2, "Importance" section: First sentence after the table should read ".pumps' *minimum recirculation valve*..." instead of "...pumps' discharge valve..."

Response: Agreed. Text corrected.

DRS Region III Specific Comment 3: Pages 3 and 4, "Dominant sequences" section, loss of service water and loss of instrument air: The failure to feed and bleed isn't really in sequence #28 or sequence #20; although, the failure to restore service water will lead to the failure to feed and bleed.

Response: Text modified to emphasize that failure to recover instrument air pressure or service water flow results in the inability to initiate feed and bleed cooling or secondary cooling.

DRS Region III Specific Comment 4: Page 5, "Sequence of interest," Initiating events: Due to a second preliminary RED finding in the AFW recirculation line (identified on October 29, 2002), the licensee has re-performed its seismic analysis and determined that the instrument air system will be able to withstand a safe shutdown earthquake. Would this ASP analysis have to consider this new licensee analysis?

Response: A seismically induced loss of instrument air is the largest contributor to the CCDP for the condition analyzed in this precursor analysis. If new seismic analyses show that the instrument air system would survive a seismic event, then this precursor analysis will be updated to remove the seismic initiating event. We are only aware of the utility's new seismic study on the condensate storage tanks.

A memo from Michael Tschiltz, Chief, Probabilistic Safety Assessment Branch, Division of Safety System Analysis (OSSA), Office of Nuclear Reactor Regulation (NRR), to Patrick Baranowsky, Chief, OERAB, DRAA, RES (dated May 19, 2003), provided peer review comments on the Preliminary Precursor Analysis of the condition reported in LER 266/01-005. The comments have been reviewed and the following responses provided:

DSSA/NRR Comment 1: *Assumptions of AFW pump failure:* The current analysis assumes a pump failure probability of 1.0 given a loss of instrument air. In review of industry operating experience, similar issues at other Westinghouse plants have not been as significant when considering actual system performance. In particular, the attached LER (excerpts highlighted) documents an actual loss of AFW pump recirculation event at McGuire, unit 1 and indicates that leakage past closed flow control valves and/or AFW flow recirculation valves may be sufficient to prevent imminent AFW pump failure. Subsequent inspection of the AFW pumps revealed no damage even though the pumps operated from 20 to 60 minutes in the so called "deadhead" condition. The AFW pumps were multistage, horizontal centrifugal pumps (eight-stage motor- driven pumps and a nine-stage turbine-driven pump). Note that the current McGuire AFW system uses automatic recirculation control (ARC) valves and is not dependent on the instrument air system (IAS). The ARC valves were installed after the event. Note also for Point Beach, the licensee's AFW pump vendor has indicated that 10 to 20 gallons per minute flow is sufficient to prevent imminent pump failure (similar to that of the McGuire experience).

An evaluation of the type of flow control valves and/or flow recirculation valves and their susceptibility to leakage under high AFW pump discharge pressure could provide higher confidence in the upper bound pump failure probabilities used in the ASP analysis.

Response: In 1988, NMC installed modifications to increase the design minimum recirculation flow for the AFW pumps to 70 gallons per minute (gpm) for the motor-driven pumps and 100 gpm for the turbine-driven pumps. Previously, the minimum recirculation flow was 30 gpm, which the AFW pump vendor, Byron Jackson, indicated would be sufficient to prevent pump damage, based on pump heat up when on recirculation flow (Ref. 2). For the concern analyzed, leakage past the AFW pump's closed recirculation valve or the closed discharge or flow control valve could provide enough flow to prevent pump damage. However, inclusion of component failures as success logic in a risk model is typically not done. The failure probability for an air- operated valve failing to close on demand is 1.0E-3; the failure probability for a motor-operated valve failing to close on demand is 3.0E-3. The likelihood that the air-operated flow control valve or air-operated recirculation valve for one of the motor-driven AFW pumps fails to close on demand or the motor-operated discharge valve or air-operated recirculation valve for the turbine- driven pump fails to close, preventing damage to at least one pump is 8.0E-3. Unless leakage past these valves, sufficient to prevent pump damage, normally occurs, the likelihood that the valves will meet their design intent and close when demanded, resulting in pump deadhead conditions is 0.992.

The following events demonstrate that damage may or may not occur when an AFW pump is run in deadhead conditions. An event occurred at McGuire Unit 1(LER 369/97-009) in which the AFW pumps were run for 20 to 60 minutes without adequate recirculation flow while the pumps' flow to the steam generators was throttled back. Leakage past valves (10 to 12 gpm) provided adequate flow to prevent pump damage. At Zion Unit 1 during pump performance testing (LER 295/90-002), the turbine-driven AFW pump was run in full deadhead conditions (with all discharge valves and the recirculation valve fully closed) for about 8 minutes, resulting in damage to the pump's impeller.

Several sensitivity cases were run varying the likelihood that all of the AFW pumps would fail. One case assumed that the likelihood that at least one AFW pump would be saved (not fail) was 90%. The increase in core damage probability (CDP) was reduced by 60%; however, the CDP was greater than 1.0E-04.

DSSA/NRR Comment 2: *Clarification of the seismic analysis section:* The current discussion notes that the "design basis earthquake" is 0.06g. Our review of the licensee's IPEEE indicates that the "safe shutdown earthquake" is 0.12 g peak ground accelerations (PGA). Also, the relationship of the plant's existing IAS piping design to the cited ANSI standard should be explained. The context may be intuitive to those individuals who perform seismic evaluations; however, it is not obvious to the non-informed reader what the relationship to the standard means. It should be noted that the IPEEE indicated that the piping was determined to be "seismically weak" due to the long pipe runs. Should you choose to state this in the ASP analysis, it may be beneficial to note that no credit for instrument air is a conservative assumption and suffices to meet the IPEEE intent of identification of potential severe accident vulnerabilities. Such an assumption in ASP analysis may be overly conservative if attempting to quantify a best-estimate risk value.

The seismic event tree and assumptions indicate that earthquakes exceeding event the lowest range reported in NUREG-1488 (50 cm/sec^2) would result in core damage appears to be quite conservative. Review of the LLNL curve distribution for annual probability of exceedance versus peak ground accelerations reveals that for the Point Beach site, the probability distribution is skewed in favor of smaller magnitude earthquakes. The current assumption that exceeding even very small magnitude earthquakes would render IAS unavailable appears unjustified based on not meeting an ANSI pipe design-specification alone. Review of actual earthquake performance of nonnuclear power stations near the Lorna Prieta, California, 1989 earthquake epicenter (considered a strong earthquake) only sustained "minor" damage (see EQE Engineering report, *The October 17, 1989 Lorna Prieta Earthquake).*

Footnote 7 on Table 4 (ASP model basic event probabilities that were modified) states that the base case value for the seismic initiating event (IE) frequency was 1.5E-05/year and was taken from "the Point Beach Units 1 and 21ndividual Plant Examination (IPE)" update of 1997. This reported number appears to represent the base, nominal annualized seismic risk and not the seismic initiating event frequency.

Response: A seismically induced loss of instrument air was considered as a contributor to core damage. A seismic event is assumed to result in non-recoverable damage to the non-seismically qualified instrument air system, leading directly to core damage. A simplified event tree was created for this purpose (see Figure 2). The safe shutdown earthquake for Point Beach is 0.12 g and the operating basis earthquake is 0.06 g. Because the instrument air system piping design is less robust than ANSI B31.1 piping design, the instrument air system cannot be assumed to withstand any seismic event greater than 0.06 g, without either performing a seismic analysis of the piping design or conducting visual inspections of the piping to determine seismic tolerance. Therefore, the return frequency for seismic events that would result in a loss of instrument air is conservatively estimated at 3.1E-4/yr (3.5E-8/hour) based on the lowest estimated ground acceleration value (50 cm/sec^2 or 0.05 g) at Point Beach from NUREG/CR-1488, *Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains* (Ref. 6). (The mean frequency for this acceleration is 3.1E-4/yr.) As the lowest site-specific value found in NUREG/CR-1488, this frequency is more appropriate than arbitrarily selecting the design basis earthquake, because no design value exists for instrument air system piping.

The frequency of seismically induced core damage events for the base case is taken from the IPEEE (Ref. 7). The base case frequency is 1.5E-05/yr. The operating basis earthquake leads directly to core damage (i.e., initiating event frequency= CDF), because the loss of instrument air results in loss of both the AFW system and PORVs. The IPE event tree for loss of instrument air shows that this sequence goes straight to core damage.

Comments were provided by Ian Jung in the Division of Regulatory Improvement Programs (DRIP), NRR to Don Marksberry, OERAB, DRAA, RES, via e-mail (dated March 11, 2003). Responses to these comments are as follows:

DRIP/NRR Comment 1: The degraded condition identified at Point Beach 1&2 had a preliminary CCDP of 7E-4 in ASP. The difference between 1E-3 and 7E-4 is statistically insignificant. One could argue that one of the agency's strategic goals was not essentially met.

Response: The analysis results (for Unit 2) gave a mean CCDP of 6.6E-4, with an upper bound (95%) of 1.4E-3 and a lower bound (5%) of 1.8E-5. Although this ASP analysis made every effort to determine "best estimate" risk results, conservative assumptions were made whenever more realistic conditions or assumptions could not be reliably predicted. Conservative assumptions include the following: (1) operators fail to detect (or respond to) the failed closed recirculation valves on loss of instrument air, (2) operators fail to detect and respond to pump deadhead conditions, preventing damage to at least one AFW pump, and (3) the instrument air system catastrophically fails following a seismic event. These assumptions and the bases for making them are discussed in Attachment 2. Because conservative assumptions were made in several key areas, we are confident that the true risks values are bounded by the values presented.

DRIP/NRR Comment 2: If the CCDP of 1E-3 in the agency's strategic goal is meant to include all risks, e.g., external initiating events and LP/SD (low power/shut down) events, RES should have included specific discussion on these contributors.

Response: The analysis considered all initiating events included in the SPAR model, plus external events. As stated in "Sequences of interest, Initiating events" (page 5), only those initiating events in which loss of instrument air is the direct result of the initiating event were considered when quantifying the CCDP. (Although the condition evaluated in this assessment [deadhead of the AFW pumps following loss of instrument air pressure] could result due to causes that are independent of the initiating event or due to causes that share some dependency with the initiating event, it is the latter case that has the greater risk significance.) The events considered were: loss of instrument air (due to equipment failure in the instrument air system), loss of service water (which results in loss of cooling water to the instrument air and service air compressors), loss of offsite power (which results in loss of electric power to the instrument air and service air compressors), and seismic events (which result in seismically induced failure of instrument air piping or components). When calculating the CCDP and CDP, the condition assessment used the number of hours that the unit 1 and unit 2 reactors were critical (critical hours were taken from Point Beach's monthly operating reports). This includes low power conditions. Risks during shutdown were not estimated because the fraction of time during shutdown because the period of time AFW would be needed was small.

DRIP/NRR Comment3: In terms of the use of the mean value with the uncertainty bounds, the two events with a similar mean value but with significantly different uncertainty distribution should be distinguished. Since the users of the ASP output would focus on the mean value, the use of uncertainty information should be done carefully.

Response: Agreed. No response required.

Standardized Plant Analysis Risk Model for Point Beach Units 1 and 2 *Addendum*

1. INTRODUCTION

This report provides documentation of modifications to the Point Beach Units 1 and 2 Standardized Plant Analysis Risk (SPAR) model. The modifications discussed below (shown in italics) are independent of the condition being analyzed; therefore, applicable to the baseline risk.

The format and numbering of preceding sections is consistent with main SPAR manual for Point Beach.

2. EVENT TREE MODELS

This section presents details concerning the two event trees that were modified – loss of instrument air and loss of service water. The event trees are shown in Figures 2-7 and 2-8. Following the event tree figures, Table 2-1 provides event tree specific system success criteria. Table 2-2 lists changes to the event tree linking rules that were used in generating the core damage sequences. Table 2-3 list changes to the flag sets included in the model.

2.1 Event Tree Assumptions

The following *new* assumptions apply to the event tree models:

7. ***The recovery of secondary cooling via main feedwater and steam generator atmospheric dump valve is a proceduralized action. Secondary cooling is assumed to be available for events where service water, instrument air and offsite power are available. The main feedwater pumps at Point Beach are motor-driven pumps. Service water provides cooling to the condensate and main feedwater pumps, and to the instrument air and service air compressors. Instrument air is required to operate the air-operated feedwater regulating bypass valves and the air-operated steam generator atmospheric dump valves.***

2.8 Loss of Instrument Air Event Tree

The loss of instrument air ***even tree*** is a ~~much simplified~~ derivation of the transient event tree. The following provides a description of the SPAR model loss of instrument air event tree. Event tree specific success criteria are provided, followed by a description of the event tree headings and the event tree structure. Figure 2-7 shows the loss of instrument air event tree.

2.8.1 Success Criteria

The general success criteria for loss of instrument air are provided in Table 2-1. ~~Given a loss of instrument air event, AFW is required for decay heat removal providing a successful reactor trip.~~ ***Following a loss of instrument air,*** MFW is unavailable due to the initiating event isolating the MFW bypass and regulating valves. ***Feed and bleed cooling is also unavailable since the PORVs require air to open. (Nitrogen backup source for the PORVs is procedurally isolated during power operations and containment entry, which is assumed to not be practical, would be required to restore the nitrogen backup.) Additionally, cooldown and/or RCS depressurization using the atmospheric dump valves is also made unavailable due to the loss of instrument air.***

Following a successful reactor trip, successful operation of the AFW, along with no PORVs opening, puts the reactor in a stable condition. However, if a PORV opens and then fails to reclose, HPI is required to provide makeup flow to replenish the RCS. If HPI succeeds, then long-term cooling is required. Long-term cooling is provided by either RHR or HPR.

~~Feed and bleed cooling is unavailable since the PORVs require air to open. Additionally, cooldown and/or RCS depressurization using the atmospheric dump valves is also made unavailable due to the loss of instrument air.~~

Following loss of instrument air and failure of AFW, the operators could recover air pressure and initiate feed and bleed cooling or secondary cooling (feedwater injection), avoiding core damage, if the recovery action occurs before the steam generators boil dry. The frequency for the initiating event, IE-LOIA does not include recovery of instrument air, even though instrument air could be recovered. The event tree model was updated to include the possibility of recovery of instrument air. If AFW fails, recovery of instrument air within certain time periods (see below) allows the use of feed and bleed cooling or secondary cooling as alternate means of decay heat removal.

Three recovery times are modeled:

(1) Recovery in less than 10 minutes (AIR-REC-ST) where there is little risk of failure of the AFW system (i.e., failure of the AFW pumps due to the dependency of their recirculation valves on instrument air) or of feed and

bleed functions. Any contribution to core damage probability due to this situation is not considered in the model.

(2) Recovery in more than 10 minutes and less than 20 minutes (AIR-REC-MT) where AFW system is modeled as failed and feed and bleed function is modeled with nominal failure probabilities. Feed and bleed cooling must be initiated within 30 minutes following total loss of AFW flow. Because the PORVs require air to open, instrument air pressure must be recovered before feed and bleed cooling can be initiated. If instrument air is recovered within 20 minutes, then sufficient time is available (about 10 minutes) to initiate feed and bleed cooling. Feed and bleed cooling requires a single PORV to be opened to remove the decay heat and a single HPI pump to supply makeup water to the RCS. Long-term cooling can be provided by either secondary cooling (feedwater injection) or HPR.

(3) Recovery in more than 20 minutes and less than 60 minutes (AIR-REC-LT) where neither AFW system or feed and bleed function are available. If instrument air is not recovered within 20 minutes but is recovered within 60 minutes, then secondary cooling (feedwater injection) can be initiated. Feedwater injection requires main feedwater flow to the steam generators be restored and a steam generator atmospheric dump valve be opened. Because the feedwater main and bypass regulating valves and steam generator atmospheric dump valves require air to operate, instrument air pressure must be recovered before feedwater injection can be initiated.

Steam generators dry out in about 56 minutes following the loss of feedwater. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 2-10). (Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop.) The latest time to start the procedure for initiating main feedwater injection is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing Critical Safety Procedure CSP-H.1 (Ref. 2-11) before RCS voiding becomes too excessive and impacts operator performance to recover secondary cooling before the reactor core is uncovered.

The following fault trees were developed for the new top events added to the event tree:

a. Fault trees AIR-REC-ST, AIR-REC-MT, and AIR-REC-LT were developed to represent failure of the operator to recover instrument air in the short term, medium term, and long term respectively. (See section 3.2 for descriptions

of the trees.) To estimate the nonrecovery probabilities, the LERs used to estimate the initiating event frequency were reviewed to determine the times required to recover instrument air pressure. The nonrecovery probabilities for AIR-REC-MT and AIR-REC-LT are conditional failure probabilities. See section 4.3, technical basis, for the LER data reviewed and details of the nonrecovery probability estimates.

b. Fault tree FWI was developed to represent failure to initiate feedwater injection as a mechanism for cooling the secondary plant in the event that instrument air is recovered and feed and bleed are not available. Feedwater injection is described in Critical Safety Procedure CSP-H.1 (Ref. 2-11). To achieve feedwater injection, multiple pumps and flow paths, many of which would have been operating at the time of the trip are available for an injection path. Therefore, the failure to achieve feedwater injection is dominated by the operator failing to accomplish the action. Appendix H provides the human error worksheets for event COND-XHE-XM-IA, and the dependent events COND-XHE-XM-IA1 and COND-XHE-XM-IA2.

Figure 2-7 presents the updated event tree.

2.8.2 General Description/Philosophy

2.8.2.1 Top Event Descriptions

The loss of instrument air event tree has the following events arranged in the approximate order in which they would be expected to occur following the event.

IE-LOIA Initiating event loss of instrument air ***to both units. (Instrument air and station air systems are shared systems, providing air to both units.)***

AIR-REC-ST Success or failure to recover instrument air pressure in the short term. Success implies instrument air will be recovered to the AFW minimum flow recirculating valves prior to AFW pump dead heading. Failure implies that instrument air is not recovered within 10 minutes (4 minutes after reactor scram).

RT Success or failure of the reactor protection system (RPS) to insert enough negative reactivity by the control rods to shutdown the reactor.

AFW Success or failure of the auxiliary feedwater system to remove decay heat via the steam generators. The main feedwater pumps will be unavailable due to the initiating event. This will require the use of the AFW system to provide flow to the steam generators. Success implies automatic actuation and operation of the AFW system. The AFW system supplies sufficient cooling water to the steam generators. The success criteria are the TDP AFW train ~~from Unit 1~~ or ~~operator aligns the~~ ***one of two*** AFW motor-driven ***trains*** ~~pump from Unit 2~~. The mini-flow valves are air-operated and fail closed due to the initiator. Operator action to control these valves is required.

CHG-CNTRL Success or failure of operator action to control charging and letdown. Letdown isolates on a loss of instrument air. If the operator fails to control charging injection and/or fails to reestablish letdown, pressurizer level is assumed to increase to the point that water is released through the ***pressurizer power operated relief valves (***PORVs***)*** and/or ***safety*** relief valves ***(SRVs)***. The likelihood of the valves reclosing after passing water is expected to be lower than after passing steam.

PORV Success or failure of the ***pressurizer*** ~~power operated relief valves (~~PORVs~~)~~ ***and SRVs***. Success requires that no PORVs/***SRVs*** opened given the transient. Failure implies that the RCS pressure increased to a point that caused at least one PORV ***or SRV*** to lift and relieve RCS pressure.

PORV-RES Success or failure of the PORVs/***SRVs*** to reclose. Success implies that all opened PORVs/***SRVs*** reclose once RCS pressure is lower than the relief pressure setpoints for the PORVs/***SRVs*** or the operator closes the PORV block valve(s).

AIR-REC-MT Success or failure to recover instrument air pressure within 20 minutes. Success implies instrument air is available to the PORVs in time for effective feed and bleed cooling. Failure implies that instrument air pressure is not recovered in time to initiate feed and bleed cooling. This event is a conditional event; probability of non-recovery given failure to recover instrument air within 4 minutes (a previous recovery event that appears earlier in the LOIA event tree.)

HPI Success or failure of the high pressure injection (HPI) system to provide makeup water to the RCS. Success implies automatic actuation and operation of the high pressure injection system (safety injection [SI] pumps). The pumps take suction from the RWST and provide flow to the RCS cold legs. The HPI system provides sufficient water to keep the core covered. The success criteria are one-of-two SI trains.

AIR-REC-LT Success or failure to recover instrument air pressure within 60 minutes. Success implies instrument air supply is recovered to one feedwater regulating bypass valve and one steam generator atmospheric dump valve allowing secondary cooling to be initiated. Failure implies that instrument air pressure was not recovered within 60 minutes.

FWI ***Success or failure to initiate secondary cooling via main feedwater system and steam generator atmospheric dump valve. Success implies that operators are successful in recovering main feedwater flow to a least one steam generator. Failure implies that main feedwater flow was not recovered.***

HPR Success or failure of high pressure recirculation. Success requires the HPI pumps to take suction from the discharge of the RHR pumps and deliver the water to the RCS. HPR will provide long-term cooling for the reactor given the HPI system was successful in supplying early makeup water to the reactor. HPR is required if residual heat removal cannot be established. The decay heat will be removed from the containment sump by the RHR pump train heat exchangers. An operator action is required to align the RHR pump discharge to the HPI pump suction and verify that the containment sump valves are open and the RWST suction valves are closed. The success criteria are one-of-two SI trains and one-of-two RHR trains.

2.8.2.2 Event Tree Structure Philosophy

The event tree structure is used to represent the interactions among three functional event groupings. The first grouping is reactor shutdown as represented by the RT top event. The next grouping is early decay heat removal/inventory control as represented by the AFW and HPI events ***or BLEED and HPI events***. The final event grouping is late decay heat removal/long-term cooling as represented by the AFW***, FWI,*** and HPR events.

Short-term inventory control is only necessary if the loss of instrument air caused the RCS pressure to lift a PORV and the PORV failed to reclose on decreasing pressure. The HPI system will meet the need for short-term inventory control. Low pressure injection cannot be used since the pressure will remain above the shutoff head for the LPI pumps due to initiator induced failure of the PORVs and atmospheric dump valves.

Late decay heat removal can be established and the plant placed in a stable condition by the AFW system provided there was no stuck-open PORV. However, if AFW is unavailable, then long-term cooling can be established by ***FWI or*** HPR.

A stuck-open PORV will require the use of HPR for long-term cooling. With a single stuck-open PORV RCS pressure will not bleed off quickly enough to allow LPI/LPR in time to prevent core damage. The decay heat is removed via the RHR pump trains dedicated heat exchangers. Since the RCS cannot be depressurized HPR is required to stabilize the reactor. For HPR success, the RHR pumps need to take suction from the containment sump and cool the water by passing it through their dedicated heat exchangers. Once the sump water is cooled, it is discharged to the suction of the HPI system pump trains. Long-term cooling success requires both the RHR pumps and HPI systems to operate in a piggy-back mode. It is assumed that the RHR pump train heat exchangers will remove not only the decay heat from the RCS, but also the decay heat rejected to the containment by the opened PORVs. Therefore, containment spray or containment fan coolers are not questioned for the Level 1 analysis.

2.9 Loss of Service Water Event Tree

The following provides a description of the SPAR model loss of service water event tree. Event tree specific success criteria are provided, followed by a description of the event tree headings and the event tree structure. Figure 2-8 shows the loss of service water event tree.

2.9.1 Success Criteria

The general success criteria for loss of service water induced transients are provided in Table 2-1. Given a loss of service water, the state of the RCP seals needs to be assessed provided a successful reactor trip has occurred. RCP seal cooling requires either component cooling water to the thermal barriers or seal injection from the charging system, however, both of these systems are dependent on cooling water. Therefore, loss of service water ~~can be assumed to~~ *may* cause an RCP seal LOCA. The event tree model assumes that the RCP seals can remain intact, therefore, if there is no RCP seal LOCA, then successful operation of the secondary cooling (AFW ~~or MFW~~), along with no power operated relief valves (PORVs) opening, puts the reactor in a stable condition. However, if a PORV opens and fails to reclose or there is an RCP seal leak, then HPI is required to provide makeup to replenish the RCS. Recovery of cooling water is required prior to use of ~~HPI~~ *HPR*. If cooling water is recovered, then HPI can provide sufficient makeup. Given success of HPI, long-term cooling using HPR is required.

Feed and bleed cooling can provide successful decay heat removal given secondary cooling is unavailable. For feed and bleed success, a single PORV is required to open and remove the decay heat while HPI provides makeup flow to replenish the lost RCS inventory. With the exception of the PORV success criteria, these success criteria are consistent within the PWR Class B plants.

Following loss of service water and failure of AFW, the operators could recover service water (and the instrument air compressors) and initiate feed and bleed cooling or secondary cooling (feedwater injection), avoiding core damage, if the recovery action occurs before the steam generators boil dry. The frequency for the initiating event, IE-LOSWS does not include recovery of service water, even though service water could be recovered in the short term (i.e., in time for the operators to initiate feed and bleed cooling before core damage would occur). The event tree model was updated to include the possibility of recovery of service water. If AFW fails, recovery of service water (and instrument air pressure) within certain time periods (see below) allows the use of feed and bleed cooling or secondary cooling as alternate means of decay heat removal.

Three recovery times are modeled:

(1) Recovery in less than 10 minutes (AIR-REC-SW-ST) where there is little risk of failure of the AFW system (i.e., failure of the AFW pumps due to the dependency of their recirculation valves on instrument air) or of feed and bleed functions. Any contribution to core damage probability due to this situation is not considered in the model.

(2) Recovery in more than 10 minutes and less than 20 minutes (AIR-REC-SW-MT) where AFW system is modeled as failed and feed and bleed function is modeled with nominal failure probabilities. Feed and bleed cooling must be initiated within 30 minutes following total loss of AFW flow. Because the PORVs require air to open, instrument air pressure must be recovered before feed and bleed cooling can be initiated. If service water flow and instrument air is recovered within 20 minutes, then sufficient time is available (about 10 minutes) to initiate feed and bleed cooling. Feed and bleed cooling requires a single PORV to be opened to remove the decay heat and a single HPI pump to supply makeup water to the RCS. Long-term cooling can be provided by either secondary cooling (feedwater injection) or HPR.

(3) Recovery in more than 20 minutes and less than 60 minutes (AIR-REC-SW-LT) where neither AFW system or feed and bleed function are available. If service water flow and instrument air is not recovered within 20 minutes but is recovered within 60 minutes, then secondary cooling (feedwater injection) can be initiated. Feedwater injection requires main feedwater flow to the steam generators be restored and a steam generator atmospheric dump valve be opened. Because the feedwater main and bypass regulating valves and steam generator atmospheric dump valves require air to operate, instrument air pressure must be recovered before feedwater injection can be initiated.

Steam generators dry out in about 56 minutes following the loss of feedwater. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 2-10). (Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop.) The latest time to start the procedure for initiating main feedwater injection is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing Critical Safety Procedure CSP-H.1 (Ref. 2-11) before RCS voiding becomes too excessive and impacts operator performance to recover secondary cooling before the reactor core is uncovered.

The following fault trees were developed for the new top events added to the event tree:

a. ***Fault trees AIR-REC-SW-ST, AIR-REC-SW-MT, and AIR-REC-SW-LT were developed to represent failure of the operator to recover service water flow and air pressure in the short term, medium term, and long term respectively. (See section 3.2 for descriptions of the trees.) To estimate the failure probabilities, LERs involving failures in service water systems were reviewed to determine the times required to recover service water flow. The nonrecovery probabilities for AIR-REC-SW-MT and AIR-REC-SW-LT are conditional failure probabilities. See section 4.3, technical basis, for the LER data reviewed and details of the nonrecovery probability estimate.***

b. ***Fault tree FWI was developed to represent failure to initiate feedwater injection as a mechanism for cooling the secondary plant in the event that instrument air pressure is recovered and feed and bleed are not available. Feedwater injection is described in Critical Safety Procedure CSP-H.1 (Ref. 2-11). To achieve feedwater injection, multiple pumps and flow paths, many of which would have been operating at the time of the trip are available for an injection path. Therefore, the failure to achieve feedwater injection is dominated by the operator failing to accomplish the action. Appendix H provides the human error worksheets for event COND-XHE-XM-IA, and the dependent events COND-XHE-XM-IA1 and COND-XHE-XM-IA2.***

Figure 2-8 presents the updated event tree.

2.9.2 General Description/Philosophy

2.9.2.1 Top Event Descriptions

The loss of service water event tree has the following events arranged in the approximate order in which they would be expected to occur following the event.

IE-LOSWS — Initiating event loss of service water *to both units (the service water system consists of a total six pumps, providing service water flow to both units)*

AIR-REC-SW-ST — ***Success or failure to recover service water flow (and instrument air pressure) in the short term. Success implies instrument air will be recovered to the AFW minimum flow recirculating valves prior to AFW pump dead heading. Failure implies that service water (and instrument air pressure) is not recovered within 10 minutes (4 minutes after reactor scram).***

RT — Success or failure of the reactor protection system (RPS) to insert enough negative reactivity by the control rods to shutdown the reactor.

RCPSL-SWS Success or failure of the RCP seals. Success requires that the charging system or component cooling water system provides RCP seal cooling Success implies that there is no seal LOCA.

AFW Success or failure of the auxiliary feedwater system to remove decay heat via the steam generators. The main feedwater pumps will isolate given low T_{ave} signal. This will require the use of the AFW system to provide flow to the steam generators. Success implies automatic actuation and operation of the AFW system. The AFW system supplies sufficient cooling water to the steam generators. The success criteria are ~~one-of-two~~ ***the TDP*** AFW train ~~from Unit 1~~ or ~~operator aligns the AFW~~ ***one of two*** motor-driven pump ***AFW trains*** ~~from Unit 2~~.

~~MFW-NT~~ ~~Success or failure of the main feedwater system given AFW is unavailable. Main feedwater is automatically tripped and isolated when an SI signal is actuated. However, the bypass valves can be opened and the MFW system manually placed in operation.~~ ***This branch was deleted and replaced with feedwater recovery event.***

CHG-CNTRL ***Success or failure of operator action to control charging and letdown. Letdown isolates on a loss of instrument air (following loss of service water flow to the air compressors). If the operator fails to control charging injection and/or fails to reestablish letdown, pressurizer level is assumed to increase to the point that water is released through the pressurizer PORVs or SRVs. The likelihood of the valves reclosing after passing water is expected to be lower than after passing steam.***

PORV Success or failure of the ***pressurizer*** ~~power operated relief valves (~~PORVs***/SRVs***~~)~~. Success requires that no PORVs***/SRVs*** opened given the transient. Failure implies that the RCS pressure increased to a point that caused at least one PORV ***or SRV*** to lift and relieve RCS pressure.

PORV-RES Success or failure of the PORVs***/SRVs*** to reclose. Success implies that all opened PORVs***/SRVs*** reclose once RCS pressure is lower than the relief pressure setpoints for the PORVs***/SRVs*** or the operator closes the PORV block valve(s).

AIR-REC-SW-MT ***Success or failure to recover service water flow (and instrument air pressure) within 20 minutes. Success implies instrument air is available to the PORVs in time for effective feed and bleed cooling. Failure implies that service water flow is not recovered in time to initiate feed and bleed cooling.***

BLEED Success or failure of the bleed portion of feed and bleed cooling given secondary cooling is unavailable. Success requires a single PORV to open and remove decay heat from the RCS. An operator is required to open the PORVs and the PORV block valves if they are closed. The success criteria are one-of-two PORVs open to create the bleed path. ***Because the PORVs are air-operated, recovery of service water flow to the instrument air or service air compressors (and recovery of instrument air pressure) is required for BLEED.***

~~REC-SWS Success or failure of recovering cooling water. Success implies that an operator was able to restore cooling water flow in a timely manner.~~ ***This branch was deleted and replace with other recovery events.***

HPI Success or failure of the high pressure injection (HPI) system to provide makeup water to the RCS. Success implies automatic actuation and operation of the high pressure injection system (safety injection [SI] pumps). The pumps take suction from the RWST and provide flow to the RCS cold legs. The HPI system provides sufficient water to keep the core covered. The success criteria are one-of-two SI trains. ***HPI does not require service water or component cooling water during the injection phase.***

AIR-REC-SW-LT Success or failure to recover service water flow (and instrument air pressure) within 60 minutes. Success implies instrument air supply is recovered to one feedwater regulating bypass valve and one steam generator atmospheric dump valve allowing secondary cooling to be initiated. Failure implies that service water flow or instrument air pressure was not recovered within 60 minutes.

FWI Success or failure to initiate secondary cooling via main feedwater system and steam generator atmospheric dump valve. Success implies that operators are successful in recovering main feedwater flow to a least one steam generator. Failure implies that main feedwater flow was not recovered. Recovery of service water flow to the instrument air or service air compressors (and recovery of instrument air pressure) is required for FWI.

HPR-S Success or failure of high pressure recirculation. Success requires the HPI pumps to take suction from the discharge of the RHR pumps and deliver the water to the RCS. HPR will provide long-term cooling for the reactor given the HPI system was successful in supplying early makeup water to the reactor. HPR is required if residual heat removal cannot be established. The decay heat will be removed from the containment sump by the RHR pump train heat exchangers. An operator action is required to align the RHR pump discharge to the HPI pump suction and verify that the containment sump valves are open and the RWST suction valves are closed. The success criteria are one-of-two SI trains and one-of-two RHR trains. ***Recovery of service water flow to component cooling water is required for RHR and HPR.***

2.9.2.2 Event Tree Structure Philosophy

The event tree structure is used to represent the interactions among three functional event groupings. The first grouping is reactor shutdown as represented by the RT top event. The next grouping is early decay heat removal/inventory control as represented by the AFW, ~~MFW-T~~, BLEED, and HPI events. The final event grouping is late decay heat removal/long-term cooling as represented by the AFW, ~~MFW-T~~ ***FWI***, and HPR events.

Short-term inventory control is only necessary if the loss of service water event caused an RCP seal leak or the RCS pressure to lift a PORV and the PORV failed to reclose on decreasing pressure. The HPI system will meet the need for short-term inventory control. For early decay heat removal, ~~either the~~ AFW ~~or MFW systems~~ can be used. If ~~both~~ AFW ~~and MFW are~~ ***is*** unavailable, then feed and bleed cooling is required to remove the decay heat. Feed and bleed cooling requires a single PORV to be opened to remove the decay heat and the HPI pumps to supply makeup water to the RCS. ***However, service water flow to the instrument air or service air compressors and recovery of instrument air pressure is required to initiate feed and bleed cooling.***

Late decay heat removal can be established and the plant placed in a stable condition by the AFW ~~or MFW~~ systems provided there was no stuck-open PORV or RCP seal LOCA. However, if AFW ~~and MFW were~~ ***was*** unavailable and feed and bleed was utilized for early decay heat removal, then long-term cooling can be established by ***either FWI or*** HPR. HPR can be used for long-term cooling provided ***service*** ~~cooling~~ water is recovered. ***FWI can be used for cooling if service water flow to the instrument air and service air compressors (and recovery of instrument air pressure) has occurred.*** Without the successful recovery of the cooling water system, long-term cooling using ***FWI or*** HPR is rendered unavailable.

A stuck-open PORV or RCP seal leak will require the use of HPR for long-term cooling. For HPR success, cooling water must be recovered in order to provide cooling to the component cooling water system which provides cooling to the RHR pump train dedicated heat exchangers and pump oil coolers. HPR is the primary method for long-term core cooling. For HPR success, the RHR pumps need to take suction from the containment sump and cool the water by passing it through their dedicated heat exchangers. Once the sump water is cooled, it is discharged to the suction of the HPI system pump trains. Long-term cooling success requires both the RHR pumps and HPI systems to operate in a piggy-back mode. It is assumed that the RHR pump train heat exchangers will remove not only the decay heat from the RCS, but also the decay heat rejected to the containment by the opened PORVs or the RCP seal LOCA. Therefore, containment spray or containment fan coolers are not questioned for the Level 1 analysis.

2.15 References

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2-2. Memo from Ralph Caruso to Mark Cunningham, Thermal-hydraulic Calculation of SGTR with Feed and Bleed at a B&W Plant, TAC-M98487.

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2-5. U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, Volume 2, NUREG-1150, June 1989.

2-6. D. L. Kelly, et al., *Assessment of ISLOCA Risk-Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant*, NUREG/CR-5744, April 1992.

2-7. D. L. Kelly, et al., *Assessment of ISLOCA Risk-Methodology and Application to a Combustion Engineering Plant*, NUREG/CR-5745, April 1992.

2-8. W. J. Galyean, D. I. Gertman, *Assessment of ISLOCA Risk-Methodology and Application to a Babcock and Wilcox Nuclear Power Plant*, NUREG/CR-5604, April 1992.

2-9. W. J. Galyean, et al., *ISLOCA Research Program Final Report*, NUREG/CR-5928, July 1993.

2-10 Wisconsin Electric Power Company, Point Beach Nuclear Power Plant - Units 1 and 2, Individual Plant Examination, Revised December 1997.

2-11. Wisconsin Electric Power Company, Point Beach Nuclear Plant Critical Safety Procedure, CSP-H.1, "Response to Loss of Secondary Heat Sink," Rev. 21 (Unit 1), Rev. 22 (Unit 2), 4/26/2001.

Table 2-1. Point Beach Unit 1 and 2 success criteria **(changes in bold)**.

Initiator	Inventory control		Decay heat removal	
	Injection Phase	Recirculation Phase	Early	Late
Loss of Instrument Air (IA)	1 SI train	1 SI train and 1 RHR train[1]	1 AFW train <u>or</u> ***MFW***[7] <u>or</u> ***(1 PORV***[7] ***and 1 SI train)***	1 AFW train <u>or</u> ***MFW***[7] <u>or</u> (1 SI train and 1 RHR train[1])
Loss of Service Water (SW)	1 SI train	1 SI train and 1 RHR train[1]	1 AFW train <u>or</u> ***MFW***[7] <u>or</u> ***(1 PORV***[7] ***and 1 SI train)***	1 AFW train <u>or</u> ***MFW***[7] <u>or</u> (1 SI train and 1 RHR train[1])

Notes:

1. The RHR pumps are used in high pressure recirculation to provide suction to the HPI pumps in a piggy-back mode **(known as high-pressure recirculation or HPR)**
2. The RHR pumps are used for shutdown cooling.
7. **Main feedwater (MFW) and pressurizer power-operated relief valve (PORV) available on the recovery of service water and instrument air compressors. Service water required for cooling of condensate and main feed pumps and air compressors at Point Beach. Instrument air needed for PORVs, steam generator atmospheric dump valves, and feed regulating bypass valves.**

Table 2-2. Point Beach Unit 1 and 2 event tree linking rules.

Event Tree	Rule	Comments
LOIA	1. If always then apply flag set FLAG-LOIA	This rule is a dynamic flag set rule which adds the flag set FLAG-LOIA to those accident sequences meeting the criteria. Information about this flag set is discussed in Table 2-3.
	2. If always then PORV is PORV-L, and /PORV is /PORV-L.	During a LOIA event the condenser will be unavailable. With the condenser unavailable, the likelihood of PORVs lifting is increased. It is assumed that the likelihood of lift is similar to that experienced during a LOOP event.
	3. If CHG-CNTRL fails, then PORV-RES is PORV-A, and /PORV-RES is /PORV-A.	During a LOIA event where operators fail to control charging or reestablish letdown, the PORVs are expected to pass liquid as pressurizer level increases. This rule increases the likelihood of failure to reclose in this situation.
LOSWS	1. If always then apply flag set FLAG-LOSWS	This rule is a dynamic flag set rule which adds the flag set FLAG-LOSWS to those accident sequences meeting the criteria. Information about this flag set is discussed in Table 2-3.

Table 2-3. Point Beach Unit 1 and 2 sequence flag sets.

Flag set	Action	Comments
FLAG-LOIA	Set HE-AFW-LOIA, PPR-SRV-OO-SRV1 and PPR-SRV-OO-SRV2 to TRUE	This flag set activates the dependency of AFW recirc valves on instrument air. It also sets the pressurizer safety failures precipitated by passing liquid to ignore. The safeties are not expected to pass water in this scenario.
FLAG-LOSWS	Set HE-LOSWS to TRUE	This flag set sets the SWS system to TRUE to evaluate the initiating event.

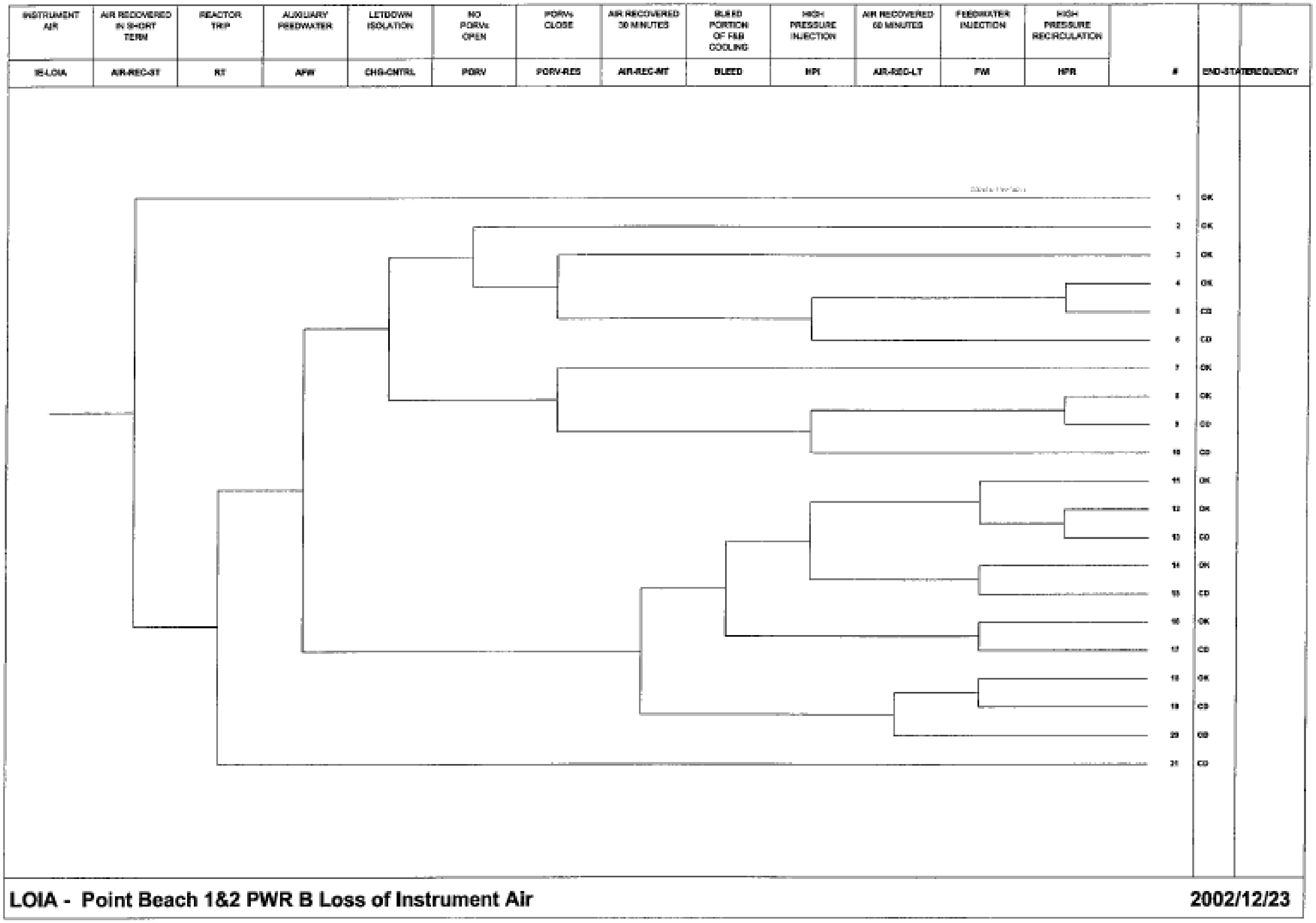
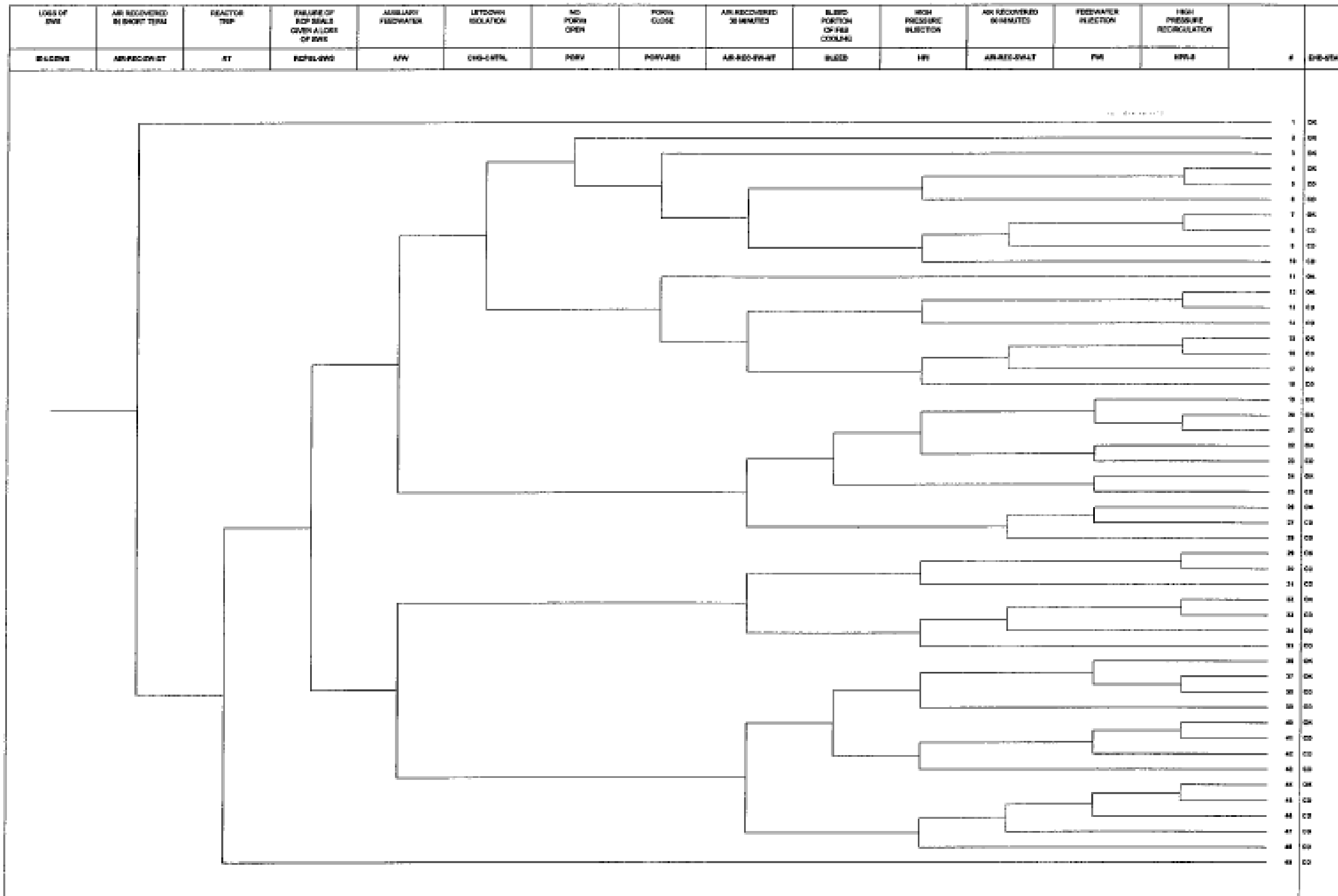


Figure 2-7 Loss of Instrument Air Event Tree



LOSWS - Point Beach 1&2 PWR B Loss of Service Water **2003/01/10**

Figure 2-8 Loss of Service Water Event Tree

3. FAULT TREE MODELS

Six fault trees were added to represent events for recovering instrument air or service water; AIR-REC-LT, AIR-REC-MT, AIR-REC-ST, AIR-REC-SW-LT, AIR-REC-SW-MT, and AIR-REC-SW-ST. One fault tree was added to represent feed water injection; FWI. The fault tree for the AFW system was modified to remove development for the AFW pumps' recirculation valves. The modified support system dependency matrix is shown in Figure 3-1. Table 3-2 lists the fault tree flag set and house events. The graphic representations of each new or modified fault tree are included in Appendix A.

3.1 Fault Tree Modeling Guidelines

The following guidelines have been used in developing the SPAR system models:

1. Only failure of major components (pumps, valves, heat exchangers, etc.) identified in the system P&IDs will be considered (i.e., no pipe rupture). Also, only hardware, human error, and common mode failures will be considered.

2. Spurious actuation or trip events are not modeled. This includes the spurious opening or closing of boundary valves and full-flow test line isolation valves during the course of the system demand.

3. Failure of the automatic actuation circuitry is modeled as a single basic event, if deemed necessary.

4. Pre-accident human errors, such as miscalibration, misalignment, etc. are included as an element of demand failure probabilities (e.g., a pump fails to start event).

5. Small diversion paths such as sample lines, small relief valves, vent lines, etc. that are smaller than one-third the main flowpath or have a flow-limiting orifice are not modeled, except HPI minflow valves are modeled in the HPR fault tree for administrative requirements. Full-flow test line isolation valve failures are included in the pump supercomponent as a mechanism for failing the pump (when system configuration allows and when not precluded by items 2 and 5 above).

6. By-pass lines around heat exchangers are modeled for the cooling function but are not considered for injection purposes.

7. In general, manual valves that are not expected to be operated are not modeled. However, those manual valves modeled in Revision 2 will be left in their appropriate supercomponent for completeness. The failure probability for manual valves not required to change positions

will be set to n/a. Manual valves that must change position are modeled, however, failure is dominated by human error.

8. Common cause failures are modeled for active components only, except for plugging of heat exchangers. Common cause is only modeled for like components within a system, never across system boundaries. Common cause is modeled for the following: motor-operated valves, air-operated valves, explosive valves, BWR safety relief valves, PWR PORVs, check valves, pumps, heat exchangers, and diesel generators.

9. Tanks are assumed to be filled to the minimum allowed by technical specifications. This amount is assumed to be sufficient for the required mission time, unless other information is available. For the Refueling Water Storage Tank (RWST) or equivalent, the assumption is that there is sufficient water for the injection phase.

10. Valve failure basic events include the valve body, the driver, local I&C circuitry (mounted on or near the valve), and limit switches. Valve rupture is not modeled. Plugging of valves is not generally modeled. Exceptions include valves in dirty systems, valves known to have an interval between flow tests of several years or more, and, based on engineering judgement, valves that may be in a key location that can fail multiple trains or systems.

11. Heat exchanger (usually decay heat removal heat exchangers) plugging events are included.

12. Pump failure basic events include the pump body, the driver, the controller and local I&C circuitry, and any local self-contained cooling or lubricating systems.

13. Diesel generator basic events include all contributions to failures to start or run. These include all support systems unique to the diesel generator.

3.2 Fault Tree Model Notes and Comments

3.2.4 Auxiliary Feedwater System (AFW)

1. This fault tree is used in the Transient, LOOP, LDC01, ***LOIA,*** LOCCW, LOSWS, and Small LOCA, event trees whenever reactor trip (top event RT) has been successful.

2. Success is defined as one-out-of-two AFW pump trains (i.e., one motor-driven pump and one turbine-driven pump) from Unit 1 or operator aligns the AFW motor-driven pump from Unit 2 delivering water to at least one-out-of-two steam generators. Mission time is 24 hours.

3. Not included in the model:
   1. Failures of the testing/recirc line as a divergence.
   2. Actuation signal failures.

4. Assumptions:
   - The motor-driven pumps are self-contained and require no room cooling nor bearing cooling.
   - The motor-driven and turbine-driven pumps are dependent upon ac and dc power for starting and control.
   - Room cooling for the turbine-driven pump is also assumed not to be required.
   - The auxiliary feedwater fault tree models are based on Reference 3-1.

3.2.7 Bleed Portion of Feed and Bleed (BLEED)

1. This fault tree is used in the Transient, LDC01, LOCCW, *LOIA*, LOSWS, SGTR, and Small LOCA event trees whenever secondary cooling (i.e., AFW and MFW) is not available.

2. Success requires an operator to open one-of-two power-operated relief valves (PORVs) to create a bleed path to remove the decay heat. Success also requires a PORV to open along with its respective block valve if it is closed during full power operation.

3.2.9 Reestablish Letdown Cooling (CHG-CNTRL)

1. This fault tree is used during LOIA ***and LOSWS*** events to reestablish letdown cooling after it isolates on a loss of instrument air. ***(Loss of service water results in trip of the instrument air and service air compressors and therefore a loss of instrument air pressure.)***

2. Success requires the operator to limit charging or to reestablish letdown cooling in time to preclude opening of the PORVs due to overfilling the RCS.

3. Not included in the model:
   - Hardware failures associated with valve manipulations.

4. Assumptions:
   - Hardware failures are negligible compared to failure of the operator action..
   - PORV opening due to overfill results in passing of liquid and a subsequently higher PORV failure rate.

3.2.18 High Pressure Injection (HPI)

1. This fault tree is used in all of the event trees except LOOP whenever reactor coolant inventory is being lost.

2. Success requires delivery of water from the RWST to the reactor vessel by one-out-of-two SI pumps. Mission time is 24 hours.

3. Not included in the model:
   - Failure of HPI recirc line components (miniflow lines back to the RWST) as pump failure or divergence.
   - Actuation signal failures.

4. Assumptions:
   - HPI pump seal water cooling by the CCW is not required for injection.
   - Room cooling is not required during injection.
   - The system is automatically actuated and support system dependencies include division A and B ac and dc power.
   - This fault tree model contains the fault tree flag set to switch ac power dependency from onsite to offsite ac power.

3.2.20 High Pressure Recirculation (HPR)

1. This fault tree is used in all event trees except LOOP whenever HPI has been successful and RHR is unavailable.

2. Success requires delivery of water from the containment recirculation sump to the reactor vessel using at least one-out-of-two RHR pumps and one-out-of-two SI pumps. Mission time is 24 hours.

3. Not included in the model:
   - Actuation signal failures.
   - Automatic switchover to recirculation signal failures.

4. Assumptions:
   - Failure to close the RWST supply isolation valves will cause the HPI pumps to lose suction in the recirculation mode.
   - An operator action is required to align the discharge of the RHR pumps to the suction of the SI and charging pumps.
   - The pumps do not require room cooling for success.
   - HPI pump seal water cooling by the CCW is required for recirculation.
   - RHR pump seal water cooling and RHR heat exchanger cooling by the CCW is required for recirculation.
   - The support system dependencies include division A and B ac and dc power.
   - This fault tree model contains the fault tree flag set to switch ac power dependency from onsite to offsite ac power.

3.2.22 High Pressure Recirculation during LOCCW and LOSWS (HPR-S)

1. This fault tree is used in the LOCCW and LOSWS event trees whenever HPI has been successful and RHR is unavailable.

2. Success requires delivery of water from the containment recirculation sump to the reactor vessel using at least one-out-of-two RHR pumps and one-out-of-two SI pumps. Mission time is 24 hours.

3. Not included in the model:
   - Actuation signal failures.
   - Automatic switchover to recirculation signal failures.

4. Assumptions:
   - Failure to close the RWST supply isolation valves will cause the HPI pumps to lose suction in the recirculation mode.
   - An operator action is required to align the discharge of the RHR pumps to the suction of the SI and charging pumps.
   - The pumps do not require room cooling for success.
   - HPI pump lube oil cooling and bearing jacket cooling by the CCW is required for recirculation.
   - This fault tree model contains the fault tree flag set to turn off the house event which fails CCW and SWS.

3.2.27 Main Feedwater during Non-Transients (MFW-NT)

1. This fault tree is used in the Small LOCA, LOCCW, ~~LOSWS~~ and SGTR event trees.

2. Success requires that the system operates for 24 hours.

3. Assumption: The MFW system is normally operating. Upon a safety injection signal, the MFW system will isolate. The dominant factor in the system operability can be modeled by an operator action to restore the MFW flow to the steam generators. Therefore, MFW and condensate system hardware failures are not explicitly modeled.

3.2.24 PORVs/SRVs Challenged (PORV)

1. This fault tree is used in the Transient, LDC01, LOCCW, *LOIA*, LOSWS event trees whenever secondary cooling (top event AFW) has been successful.

2. Success means that no PORVs or SRVs opened during the transient. Failure means that at least one PORV or SRV opened and reclosure must be considered in the event tree.

3. Note: This system contains the "I" process flag which allows for success logic to be quantified in the accident sequences.

3.2.38 PORV/SRV Reclosure (PORV-RES)

1. This fault tree is used in the Transient, LDC01, LOCCW, ***LOIA,*** LOSWS, LOOP, and SBO, and event trees.

2. Success requires closure of all open PORVs or PORV block valves and SRVs on decreasing plant pressure.

3. Operator action to close the block valve for any PORV that fails to close is modeled. The PORV block valve failures are modeled.

~~3.2.47 Recover Cooling Water (REC-SWS)~~

~~1. This fault tree is used in the LOSWS event tree.~~

~~2. Success means that the operator was able to recover cooling water in order for long term cooling to be successful given there was a seal LOCA or a stuck open PORV.~~

3.2.48 Residual Heat Removal (RHR)

1. This fault tree is used in all event trees except Medium and Large LOCAs once RCS pressure and temperature have been lowered to a point below RHR pump conditions.

2. Success requires one-out-of-two trains of RHR to operate and remove decay heat for 24 hours. Failure requires use of high pressure recirculation (top event HPR).

3. Interlock failures are not included in the fault tree model.

4. Assumptions:
   - The pumps do not require room cooling for success.
   - RHR pump seal water cooling by the CCW is required for residual heat removal.
   - CCW is also required for removing decay heat from the heat exchangers.
   - The system is dependent upon division A and B ac and dc power.
   - This fault tree model contains the fault tree flag set to switch ac power dependency from onsite to offsite ac power.

3.2.56 Reactor Trip (RT)

1. This fault tree is used in all event trees except the LOOP event tree.

2. Success requires a sufficient number of control rods to be inserted into the reactor core to stop the nuclear chain reaction. Operator action to manually scram the reactor is modeled. The operator action includes failure to manually scram the reactor for those failures for which

manual action can effect scram (i.e., electrical failures). Reactor Trip System failures are dominated by trip breakers and control rod drives.

3. Failure of top event RT in the Transient event tree results in an Anticipated Transient Without Scram (ATWS). The plant response is modeled in the ATWS event tree. Failure of top event RT in the other event trees is assumed to result in core damage.

3.2.60 Recovery of Secondary Cooling (SGCOOL)

1. This fault tree is used in the Transient, Small LOCA, and LOOP event trees whenever offsite power is available. This top event is asked when feed and bleed cooling has been successful.

2. Success requires recovery of secondary cooling (either AFW or MFW) to supply water to the steam generators prior to initiating sump recirculation. Success also requires the PORVs to close after successful feed and bleed. Failure requires high pressure recirculation.

3.2.70 Service Water System (SWS)

1. These fault trees are support fault trees to the component cooling water, emergency diesel generators and auxiliary feedwater systems.

2. Success requires three-of-six service water pump trains if turbine building loads are isolated and four-of-six pumps if turbine building loads are not isolated.

3. Assumptions:
   - There are a total of six pump trains that supply service water to the service water header. These pumps are shared between the two units.
   - The service water system supports the DG01 and DG02 emergency diesel generators.
   - Three-of-the-six pump trains are running and providing cooling water to both Units.

3.2.71 Recovery of Instrument Air Pressure (AIR-REC-ST, AIR-REC-MT, AIR-REC-LT)

1. These fault trees are used in the LOIA event tree and represent recovery of the instrument air system within the specified time periods. The recovery events in fault trees AIR-REC-MT and AIR-REC-LT are conditional events: AIR-XHE-RECOVERY-MT is operator fails to recover instrument air pressure within 20 minutes, given that air was not recovered within the short term (10 minutes); AIR-XHE-RECOVERY-LT is operator fails to recover instrument air pressure within 60 minutes, given that air pressure was not recovered within 20 minutes.

2. Success requires the operator to recover instrument air pressure within 10 minutes (short term), within 20 minutes (medium term), or within 60 minutes (long term) so that feed and bleed cooling or secondary cooling (feedwater injection) can be initiated.

3. *Assumptions:*
   - *AIR-REC-MT – Feed and bleed cooling must be initiated within 30 minutes following total loss of AFW flow. Because the PORVs require air to open, instrument air pressure must be recovered before feed and bleed cooling can be initiated. If instrument air is recovered within 20 minutes, then it is assumed that sufficient time is available (about 10 minutes) to initiate feed and bleed cooling.*
   - *AIR-REC-LT – Feedwater injection requires main feedwater flow to the steam generators be restored and a steam generator atmospheric dump valve be opened. Because the feedwater main and bypass regulating valves and steam generator atmospheric dump valves require air to operate, instrument air pressure must be recovered before feedwater injection can be initiated. Steam generators dry out in about 56 minutes following the loss of feed. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 3-5). (Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop.) The latest time to start the procedure for initiating main feedwater injection is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing Critical Safety Procedure CSP-H.1 (REF. 3-6) before RCS voiding becomes too excessive and impacts operator performance to recover secondary cooling before the reactor core is uncovered.*

3.2.72 Recovery of Service Water Flow (AIR-REC-SW-ST, AIR-REC-SW-MT, AIR-REC-SW-LT)

1. *These fault trees are used in the LOSWS event tree and represent recovery of service water flow (and instrument air pressure) within the specified time periods. The recovery events in fault trees AIR-REC-SW-MT and AIR-REC-SW-LT are conditional events: AIR-XHE-RECOVERY-SW-MT is operator fails to recover service water and instrument air pressure within 20 minutes, given that service water and air was not recovered within the short term (10 minutes); AIR-XHE-RECOVERY-SW-LT is operator fails to recover service water and instrument air pressure within 60 minutes, given that service water and air pressure was not recovered within 20 minutes.*

2. *Success requires the operator to recover service water flow and instrument air pressure within 10 minutes (short term), within 20 minutes (medium term), or within 60 minutes (long term) so that feed and bleed cooling or secondary cooling (feedwater injection) can be initiated.*

3. *Assumptions:*
   - *AIR-REC-SW-MT – Feed and bleed cooling must be initiated within 30 minutes following total loss of AFW flow. Because the PORVs require air to open, service water flow and instrument air pressure must be recovered before feed and bleed cooling can be initiated. If instrument air is recovered within 20 minutes, then it is*

assumed that sufficient time is available (about 10 minutes) to initiate feed and bleed cooling.

- *AIR-REC-SW-LT – Feedwater injection requires main feedwater flow to the steam generators be restored and a steam generator atmospheric dump valve be opened. Because the feedwater main and bypass regulating valves and steam generator atmospheric dump valves require air to operate, service water flow and instrument air pressure must be recovered before feedwater injection can be initiated. Steam generators dry out in about 56 minutes following the loss of feed. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 3-5). (Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop.) The latest time to start the procedure for initiating main feedwater injection is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing Critical Safety Procedure CSP-H.1 (Ref. 3-6) before RCS voiding becomes too excessive and impacts operator performance to recover secondary cooling before the reactor core is uncovered.*

3.2.73 Feedwater Injection (FWI)

1. *This fault tree is used in the LOIA and LOSWS event trees and represents recovery of main feedwater cooling in the event that instrument air is recovered and feed and bleed cooling is not available.*

2. *Success requires the operator to initiate feedwater injection as a mechanism for cooling the secondary plant.*

3. *Assumptions:*
   - *The main feedwater system has motor-driven condensate and main feed pumps. The only air-operated valves in the condensate and feed systems associated with feeding water from the condenser hot well to the steam generator are the feedwater regulating valves. Both the main and bypass regulating valve are air-operated and fail closed on the loss of instrument air pressure. To achieve feedwater injection, multiple pumps and flow paths, many of which would have been operating at the time of the trip are available for an injection path. Therefore, the failure to achieve feedwater injection is dominated by the operator failing to accomplish the action of opening the air-operated valves that failed closed. (Ref. 3-4)*
   - *Critical Safety Procedure CSP-H.1, Response to Loss of Secondary Heat Sink (Ref. 3-6), provides instructions on restoring main feedwater flow to the steam generators.*

3.3 References

3-1. J. P. Poloski, et al., Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995, NUREG/CR-5500, Vol. 1, August 1998.

3-2. G. M. Grant, et al., Emergency Diesel Generator Power System Reliability, 1987-1993, INEL-95/0035, February 1996.

3-3. D. L. Kelly, et al., *Assessment if ISLOCA Risk Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant*, NUREG/CR-5745, April 1992.

3-4 Wisconsin Electric Power Company, Point Beach Nuclear Power Plant - Units 1 and 2, Final Safety Analysis Report, Revised June 1997.

3-5 Wisconsin Electric Power Company, Point Beach Nuclear Power Plant - Units 1 and 2, Individual Plant Examination, Revised December 1997.

3-6. Wisconsin Electric Power Company, Point Beach Nuclear Plant Critical Safety Procedure, CSP-H.1, "Response to Loss of Secondary Heat Sink," Rev. 21 (Unit 1), Rev. 22 (Unit 2), 4/26/2001.

Table 3-1. Point Beach Units 1 and 2 PWR B system dependency matrix.

Train		4160 V AC Buses				125V DC Buses				Component Cooling Water Trains		Service Water	*Instr. Air*[c]
		1A	1B	2A	2B	1A	1B	2A	2B	A	B		
Auxiliary Feedwater[a]	MDP-38A	X				X						L	
	TDP											L	
	MDP-38B				X				X			L	
RHR Heat Exchangers	1A									X			
	1B										X		
High Pressure Injection	SI-A	X				X				X			
	SI-B		X				X				X		
High Pressure Recirculation	SI-A	X				X				X			
	SI-B		X				X				X		
Low Pressure Injection	1A	X				X				X			
	1B		X				X				X		
Low Pressure Recirculation	1A	X				X				X			
	1B		X				X				X		
Residual Heat Removal	1A	X				X				X			
	1B		X				X				X		
PORVs[c]	RC-431C	X[b]				X							*X*
	RC-430				X[b]				X				*X*

Train		4160 V AC Buses				125V DC Buses				Component Cooling Water Trains		Service Water	*Instr. Air*[c]
		1A	1B	2A	2B	1A	1B	2A	2B	A	B		
Emergency Power	DG01	X		[a]								X	
	DG02	[a]		X								X	
	DG03		X		[a]								
	DG04		[a]		X								
	GT05	[a]	[a]	[a]	[a]								
4160/480 V AC Div. Buses	1A	X											
	1B		X										
125 V DC Div. Buses	1A	X											
	1B		X										
Component Cooling Water	1A	X				X						X	
	1B		X				X					X	
Service Water System	P32A	X											
	P32B	X											
	P32C		X										
	P32D				X								
	P32E				X								
	P32F			X									

Train		4160 V AC Buses				125V DC Buses				Component Cooling Water Trains		Service Water	***Instr. Air***[c]
		1A	1B	2A	2B	1A	1B	2A	2B	A	B		
Instrument and Service Air Compressors[c]	***K2A***	***X***										***X***	
	K2B				***X***							***X***	
	K3A		***X***									***X***	
	K3B				***X***							***X***	

a. Alternate supply

b. The PORV block valves require ac power to open.

c. The instrument air system is not modeled, however, the PORVs require instrument air to open and AFW min-flow valves fail closed on a loss of instrument air.

L. Required after 4 hours for TDP bearing cooling and AFW pump suction water supply. Fire water is alternate supply.

X. Complete dependency.

Table 3.2. Point Beach Units 1 and 2 PWR B fault tree flag sets.

Flag Set	Action	Comments
FLAG-SYS-LOSP	Set LOSP-A and LOSP-B to FALSE	This flag set changes the power dependencies of the HPI, HPR, and RHR fault trees from onsite emergency power to strictly offsite ac power. By using system flag sets, only one fault tree logic model is required to handle ac power dependency in the LOOP and SBO fault tree.
FLAG-SYS-CCW-SWS	Set CCW-HE and SWS-HE to FALSE	This flag set sets the component cooling water and nuclear service water systems house events to FALSE (i.e., allows for CCW and SWS to provide cooling water to the HPI and RHR pump trains for injection and long-term cooling).

4. BASIC EVENT DATA

Changes to basic events are listed in Table 4-1. This section deals primarily with the independent hardware failure probabilities, although human error probabilities are included. Section 7 provides additional explanation of the human error probability calculations that uses the ASP human error reliability worksheets. Technical basis for each change are provided in Section 4.3. References are provided in Section 4.4.

Table 4-1. Basic event data report.

Event Name	Event Description	Failure Rate [/hr]	Mission Time [hr]	Uncertainty Distribution Type	Dist Param	Mean Probability
ACP-XHE-NOREC-BD	Operator fails to recover offsite power before battery depletion					***0.36***[a]
AIR-XHE-RECOVERY-LT	***Operator fails to recover instrument air in long term***			***Lognormal***	***3.43***	***0.14***[b]
AIR-XHE-RECOVERY-MT	***Operator fails to recover instrument air in medium term***			***Lognormal***	***3.43***	***0.17***[b]
AIR-XHE-RECOVERY-ST	***Operator fails to recover instrument air in short term***			***Lognormal***	***3.43***	***0.58***[b]
AIR-XHE-RECOVERY-SW-LT	***Operator fails to recover SW to instrument air in long term***			***Lognormal***	***2.00***	***0.50***[b]
AIR-XHE-RECOVERY-SW-MT	***Operator fails to recover SW to instrument air in medium term***			***Lognormal***	***2.00***	***0.61***[b]
AIR-XHE-RECOVERY-SW-ST	***Operator fails to recover SW to instrument air in short term***			***Lognormal***	***2.00***	***0.88***[b]
IE-LOIA	***Initiating event frequency for loss of instrument air***	***9.0E-7***[b]		***Gamma***	***0.5***	
IE-LOSWS	***Initiating event frequency for loss of service water***	***4.5E-6***[b]		***Gamma***	***0.5***	
RCS-MDP-SEALS	RCP seals fail w/o cooling and injection	**2.2E-1**[a]				

Table 4-1. Basic event data report (continued).

Event Name	Event Description	Failure Rate [/hr]	Mission Time [hr]	Uncertainty Distribution Type	Dist Param	Mean Probability
RCS-MDP-SEALS2	RCP seals fail w/o cooling and injection given LOSWS or LOCCW	*2.2E-1*[c]				

Notes:

a. ***Model update to incorporate 1-hour battery depletion time for Point Beach 1 and 2. For SBO sequences, electric power must be recovered before battery depletion. The actual time for recovering offsite power is assumed to be about 0.5 hour, which allows 30 minutes for the operator to perform the necessary system recovery actions. The probability of not recovering offsite power, for the weighted average of all types of LOOPs, within 0.5 hour is 0.36 (see Table 6-1). Therefore, ACP-XHE-NOREC-BD was set to 0.36.***

b. ***Values calculated. See section 4.3.***

c. ***To incorporate the Rhodes model, modifications to events were made in accordance with guidance provided in Refs. 4-1 and 4-2. High temperature seals were assumed to be installed on all RCPs. Therefore, RCS-MDP-SEALS and RCS-MDP-SEALS2 were set to 0.22. Based on the Rhodes model, the time available to prevent core damage by high pressure injection if RCP seals fail is 4 hours. However, for all SBO sequences, electric power must be recovered before battery depletion, which is assumed to occur in 1 hour. Because Point Beach has 1-hour batteries, the modifications to top events in the SBO tree were not needed.***

4.3 Technical basis

4.3.1 Total loss of instrument air frequency (IE-LOIA)

a. ***Data sources.*** In NUREG/CR-5750, *Rates of Initiating Events at U.S. Nuclear Power Plants* (Ref. 4-3), the analysis of loss of instrument air concluded there was a significant difference between reactor types (both statistical and engineering). Thus the NUREG/CR-5750 estimate applicable to Point Beach is based on LERs involving loss of instrument air in pressurized water reactors (PWRs) only. The PWR UFSARs were reviewed to determine which multi-unit sites have shared instrument air systems, meaning that one event can cause a dual unit loss of instrument air. These sites were counted as a single unit for the purpose of determining the loss of instrument air frequency. For this condition assessment, a frequency estimate for loss of instrument air was developed that is based on events identified in NUREG/CR-5750 and updated to include LER data through 2001. A search of the Sequence Coding and Search System (SCSS) database was conducted to select LERs involving failures in the instrument air system for the years 1996 through 2001.

b. ***Review criteria.*** Events were rejected using the following criteria:

- Loss of instrument air was due to loss of electric power to air compressors following a LOOP (to be accounted for in the dual LOOP frequency calculation)

- Loss of instrument air only involved a local loss of air pressure to selected components (i.e., system-wide instrument air system pressure was not affected)

- Recovery of instrument air was instantaneous due to automatic isolation of the fault (e.g., automatic isolation of instrument air system from faulted service air)

- Loss of instrument air pressure did not result in a manual or automatic reactor trip

c. ***Results.*** Table 4-2 lists the 20 events reviewed and presents brief descriptions of the causes of loss of instrument air, affected components, a disposition of each event, and whether the loss of air was recovered and the length of time required. One LOIA event was calculated for each plant trip caused by a LOIA that met the review criteria above. Some plants, such as Point Beach and Braidwood (see LER 356/88-025 in Table 4-2), have a single instrument air systems shared between two plants. The review yielded the following results:

- Six events contribute to the overall frequency for loss of instrument air (loss of service water events are handled separately and are addressed later in this section).

- Four of the six events were recovered within 4 minutes of a reactor trip: 400/87-041, 280/90-006, 285/90-026, and 306/96-002.

- The remaining two events were recovered within 30 minutes of a reactor trip: 317/87-003, and 456/88-025.

d. ***Frequency calculation.*** LOIA events can effect one unit or several units at a site with a shared Instrument air system. Therefore, the operating experience was tabulated on a per Instrument air system basis. In other words, experience at sites with a shared Instrument air system was tabulated on a per site basis, and experience at single units and on sites with generally independent Instrument air system was tabulated on a per unit basis. This was done to ensure that the numerator of the frequency calculation (events) is consistent with the denominator (critical years). The operating experience for PWRs during 1987–2001, as tabulated in Table 4-3, is 624 critical years.[1] Therefore, subtracting Point Beach's 12.4 critical years from the denominator, the estimated industry mean frequency for LOIA is:

[1]Total reactor critical years for the years 1987 through 2001 is 624 yr for all PWRs and 12 yrs for Point Beach. Data was provided by Idaho National Engineering and Environmental Laboratory. Multi-unit sites with shared instrument air systems are counted as single units as shown in Table 4-3. Reactor critical years was calculated from reactor critical hour data for the years 1987 through 2001, compiled from the monthly operating reports filed by each plant. Data are included from critical date until permanent shutdown. Data for Ft. St. Vrain was excluded.

F_{LOIA} = 6/(624-12.4) yr = 9.8E-3/yr or 1.1E-6/hr

The constrained non-informative prior distribution (Ref. 4-3) was used to a more diffuse prior to account for differences among Instrument air systems. The distribution is given by:

$$Gamma(\alpha,\beta) = Gamma\left(0.5, \frac{1}{2F}\right)$$

The gamma distribution parameters of the prior are α=0.5 and β=52.0. Performing a Bayesian update on the above distribution with Point Beach's 12.4 operating years without a LOIA event, the mean LOIA frequency for Point Beach is 7.9E-3/yr or 9.0E-7/hr. The gamma distribution parameters of the posterior are α=0.5 and β=63.4. The 5^{th} percentile of this distribution is 3.1E-5/yr and the 95^{th} percentile is 3.0E-2/yr. The calculations, which were done with Microsoft Excel, are shown in Table 4-4.

4.3.2 Total loss of service water frequency (IE-LOSWS)

a. ***Data sources.*** Due to the rare occurrence of total loss of service water events resulting in a reactor trip, the 1969 – 2001 U.S. operating experience was reviewed for such events. Sources reviewed included:

- ASP database (1969 – 2001)
- SCSS database (1996 – 2001)
- AEOD/S98-01 (1986 – 1995) (Ref. 4-4)
- NUREG-1275, Vol.3, (1980 – 1985), (Ref. 4-5).

b. ***Review criteria.*** Events were rejected using the following criteria:

- Total loss of service did not result in a manual or automatic reactor trip
- Total loss of service occurring several hours following a reactor trip
- Failure mode can not result in a total loss of service water at Point Beach.

c. ***Results.*** No total loss of service water events were discovered in the 1987 – 2001 operating experience. Information about events that occurred before 1987 is not always adequate to apply the review criteria. Based on this and the operational changes throughout the years, this older data was not used. There is no indication in the older data that loss of service water frequency is significantly higher or lower than the estimates provided below.

d. ***Frequency calculation.*** The total operating experience for PWRs and BWRs during 1987 – 2001, as tabulated in Table 4-5, is 1257 site critical years. Therefore, using the Jeffries prior distribution parameters of the gamma distribution are α=0.5 and β=0, and the parameters of the posterior are α=0.5 and β=1257. The estimated of the mean frequency for loss of service water is:

$$F_{LOSW} = 0.5/1257/yr = 4.0E\text{-}4/yr \text{ or } 4.5E\text{-}8/hr$$

The 5th percentile of this distribution is 1.6E-6/yr. and the 95th percentile is 1.5E-3/yr. The calculations, which were done with Microsoft Excel, are shown in Table 4-6.

4.3.3 Probabilities for recovery from loss of instrument air

Following loss of instrument air pressure and failure of the AFW pumps (due to the dependency of their recirculation valves on air), the operators could recover air pressure and initiate feed and bleed cooling, or secondary cooling, thus avoiding core damage. The frequency estimates for the initiating event LOIA (see Section 4.3.1) does not include recovery. The LOIA model includes recovery of instrument air pressure in the short term (i.e., before the AFW pumps fail, in the mid-term (i.e., in time for the operators to initiate feed and bleed cooling before core damage would occur) and in the long term (i.e., in time for the operators to initiate secondary cooling using the MFW system).

Failure to recover instrument air before AFW failure (AIR-XHE-RECOVERY-ST). Considering a post trip overcooling transient as most limiting, an operator may secure AFW flow within first few minutes of a reactor trip. Had the minimum recirculating valve been in the closed position due to low instrument air pressure with the AFW pumps running, the pumps would have failed within 1 to 2 minutes (Ref. 4-6). An actual reactor trip at Point Peach resulted in RCS overcooling and the closure of all AFW pump injection valves within 4 minutes of a reactor trip.

Failure to recover instrument air before last chance for effective feed and bleed cooling (AIR-XHE-RECOVERY-MT). One pressurizer PORV is needed for feed and bleed cooling. The recovery of instrument air pressure and initiation of feed and bleed must be completed within 30 minutes following the loss of AFW. Recovery of instrument air pressure and initiation of feed and bleed must occur within 30 minutes following loss of AFW flow to prevent core damage. Instrument air pressure and, thus, feed and bleed capability could be restored, even if the AFW pumps are failed. The recovery event in fault tree AIR-REC-MT is a conditional event: AIR-XHE-RECOVERY-MT is operator fails to recover instrument air pressure within 20 minutes, *given that air was not recovered within the short term (10 minutes)*.

Failure to recover instrument air before last chance to initiate main feedwater (AIR-XHE-RECOVERY-LT). Steam generators dry out in about 56 minutes following the loss of feed. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 4-7). Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop. The latest time to start the procedure for initiating main feedwater is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing the EOP (CSP-H.1) before RCS voiding becomes too excessive and impacts operator performance to recover before core uncovery. The recovery event in fault tree AIR-REC-LT is a conditional event: AIR-XHE-RECOVERY-LT is operator fails to recover instrument air pressure within 60 minutes, *given that air pressure was not recovered within 20 minutes.*

a. **Data sources.** Recovery from the LOIA event is based on an analysis of U.S. operating experience between 1987 and 2001. Events used to develop the LOIA frequency were also reviewed to determine whether instrument air was recovered and the time require to recovery air pressure. LERs used are described in Table 4-2.

b. **Results.** The data used to estimate the loss of instrument air frequency (due to equipment failures) ultimately included six LERs. Each of these LERs was reviewed to determine the recovery time for instrument air. Table 4-7 lists the recovery time for each event. Results for the recovery events are as follows.

- **Failure to recover instrument air before AFW failure (AIR-XHE-RECOVERY-ST).** The LOIA event tree has the top event AIR-REC-ST, "Recovery of instrument air within the short term." A review of the 1987-2001 U.S. operating experience shows that 4 out of 6 initiating events involving the total loss of instrument air were recovered within 10 minutes after the start of air depressurization. In addition, recovery actions were started before the reactor trip and instrument air pressure restored within 4 minutes following the reactor trip.

  These events were caused by ongoing maintenance activities. With control room annunciation of low air pressure and the awareness of ongoing maintenance on the instrument air system, the majority of failures seen in the operating experience can be recovered quickly. Although, the operators may not be aware of the limitation of the AFW pump recirculation valves, prompt action to restore air pressure would be taken to prevent a plant trip.

- **Failure to recover instrument air before last chance for effective feed and bleed cooling (AIR-XHE-RECOVERY-MT).** The LOIA event tree has the top event AIR-REC-MT, "Recovery of instrument air within 20 min," which is a conditional probability of non-recovery given failure to recover instrument air within the short term (a previous recovery event that appears earlier in the LOIA event tree.) A review of the 1987-2001 U.S.

operating experience shows that all six initiating events involving the total loss of instrument air were recoverable within 30 minutes.

- ***Failure to recover instrument air before last chance to initiate main feedwater (AIR-XHE-RECOVERY-LT).*** The LOIA event tree has the top event AIR-REC-LT, "Recovery of instrument air within 60 min," which is a conditional probability of non-recovery given failure to recover instrument air within 20 minutes (a previous recovery event that appears earlier in the LOIA event tree). A review of the 1987-2001 U.S. operating experience shows that all six initiating events involving the total loss of instrument air were recoverable within 20 minutes.

c. **Calculations.** The recovery times shown in Table 4-7 were fit to a lognormal distribution with the parameters μ=15.3 and σ=13.3. From this distribution, the non-recovery probabilities were calculated as shown in Table 4-8. The P-value for the fit to a lognormal distribution is 0.8678. Note that no events took longer than 20 minutes to recover the Instrument air system, so any analysis of recovery beyond this time requires an extrapolation. The first recovery event in the LOIA event tree, recovery of instrument air within the short term (AIR-XHE-RECOVERY-ST) is 0.58. The second recovery event in the LOIA event tree, recovery of instrument air pressure in 20 minutes (AIR-XHE-RECOVERY-MT) is 0.17. The third and final recovery event in the LOIA event tree, recovery of instrument air in 60 minutes (AIR-XHE-RECOVERY-LT) is 0.14.

4.3.4 Probabilities for recovery from loss of service water

Following loss of instrument air pressure and failure of the AFW pumps (due to the dependency of their recirculation valves on air), the operators could recover air pressure and initiate feed and bleed cooling, or secondary cooling, thus avoiding core damage. The frequency estimates for the initiating event LOSWS (see Section 4.3.2) does not include recovery. The LOSWS model includes recovery of service water flow (and instrument air pressure) in the short term (i.e., before the AFW pumps fail, in the mid-term (i.e., in time for the operators to initiate feed and bleed cooling before core damage would occur) and in the long term (i.e., in time for the operators to initiate secondary cooling using the MFW system). The LOSWS analysis assumed that the operator quickly recognizes the need for and starts an instrument air compressor once service water cooling is restored.

Failure to recover service water before AFW failure (AIR-XHE-RECOVERY-SW-ST). Considering a post trip overcooling transient as most limiting, an operator may secure AFW flow within first few minutes of a reactor trip. Had the air to the minimum recirculating valve been in the closed position due to low instrument air pressure (following trip of the compressors on loss of service water cooling) with the AFW pumps running, the pumps would have failed within 1 to 2 minutes (Ref. 4-6). An actual reactor trip at Point Peach resulted in RCS overcooling and the closure of all AFW pump injection valves within 4 minutes of a reactor trip.

Failure to recover service water before last chance for effective feed and bleed cooling (AIR-XHE-RECOVERY-SW-MT). One pressurizer PORV is needed for feed and bleed cooling. The recovery of service water flow (and instrument air pressure) and initiation of feed and bleed must be completed within 30 minutes following the loss of AFW. Recovery of instrument air pressure and initiation of feed and bleed must occur within 30 minutes following loss of AFW flow to prevent core damage. Instrument air pressure and, thus, feed and bleed capability could be restored, even if the AFW pumps are failed. The recovery event in fault tree AIR-REC-SW-MT is a conditional events: AIR-XHE-RECOVERY-SW-MT is operator fails to recover service water and instrument air pressure within 30 minutes, *given that service water and air were not recovered within the short term (10 minutes).*

Failure to recover service water before last chance to initiate main feedwater (AIR-XHE-RECOVERY-SW-LT). Steam generators dry out in about 56 minutes following the loss of feed. Core uncovery starts about 40 to 50 minutes (approximately) later. Thus, core uncovery is expected to begin at about 90 minutes if the AFW is initially not available (Ref. 4-7). Core uncovery time is longer for conditions where the AFW fails to run after filling the steam generator following reactor trip and the initial decay heat drop. The latest time to start the procedure for initiating main feedwater is assumed at 60 minutes. This assumes a reasonable time of 30 minutes for implementing the EOP (CSP-H.1) before RCS voiding becomes too excessive and impacts operator performance to recover before core uncovery. The recovery event in fault tree AIR-REC-SW-LT is a conditional event: AIR-XHE-RECOVERY-SW-LT is operator fails to recover service water and instrument air pressure within 60 minutes, *given that service water and air were not recovered within 30 minutes.*

a. **Data sources.** Recovery from the LOSWS event is data from AEOD/S98-01 (Reference 4-4).

b. **Results.** AEOD/S98-01 broke SWS events into 6 categories. The three most severe categories are of interest: Category 1 - actual total failure of SWS, Category 2 - actual total failure of SWS - conditional, and Category 3 - potential total failure or degradation of SWS. No Category 1 events occurred, 9 Category 2 events occurred, and 39 Category 3 events occurred. Category 2 events are defined as a complete failure of all redundant SWS trains, where recovery occurred, or is judged could have occurred within the time necessary to avoid degradation or failure of safety systems. Category 3 are potential failures or degradation of all redundant SWS trains, but no actual or conditional total failure. This category includes events in which equipment is declared inoperable, but still functioning.

   According to AEOD/S98-01, Category 2 events included 5 personnel/procedural errors, 2 design/seismic deficiencies, and 1 each biofouling, corrosion/erosion and foreign material/debris. The report states that recovery time for these events that occurred during operations was usually less than ½ hour. The events that occurred during shutdown took much longer to recover, but that is mainly due to the amount of equipment that can be out of service during shutdown and the comparative lack of urgency.

According to AEOD/S98-01, Category 3 events included 16 design/seismic deficiencies, 9 personnel/procedural errors, 6 biofouling, 3 silt and sedimentation, 3 corrosion/erosion and 2 foreign material/debris. The 16 design/seismic deficiencies are too hypothetical to make a quantitative estimate or qualitative judgement about recovery. The breakdown of the remaining events is statistically very similar to the Category 2 breakdown, so they are included in the recovery analysis.

Hypothetically, a Category 1 event or an actual LOSWS event could occur. The category 2 and 3 events provide a surrogate data set of recovery actions and time estimates that are considered to be applicable to this hypothetical Category 1 event. The recovery estimates focus on the nature of the failed equipment, and ignore the operational equipment that made these events conditional (Category 2) or potential (Category 3). Focusing on failed equipment rather than available back-up equipment makes the analysis more conservative. However, the use of surrogate events rather than real loss of service water events could make the results conservative or non-conservative. Overall, the analysis should be regarded as the best estimate, but there is a high level of uncertainty.

- **Failure to recover service water before AFW failure (AIR-XHE-RECOVERY-SW-ST).** The LOSWS event tree has the top event AIR-REC-SW-ST, "Recovery of service water within the short term." A review of the 1986-1995 U.S. operating experience shows that 3 out of 26 initiating events involving conditional and potential losses of service water were recovered within 10 minutes after the loss of service water.

  These events were generally caused by system misalignment, often accompanied by ongoing maintenance activities. With control room annunciation of loss of service water and the awareness of ongoing maintenance on the system, these failures were recovered quickly.

- **Failure to recover service water before last chance for effective feed and bleed cooling (AIR-XHE-RECOVERY-SW-MT).** The LOSWS event tree has the top event AIR-REC-SW-MT, "Recovery of service water within 30 min," which is a conditional probability of non-recovery given failure to recover service water (and instrument air pressure) within the short term (a recovery event that appears earlier in the LOSWS event tree). A review of the 1986-1995 U.S. operating experience shows that four events involving the conditional and potential losses of service water were recoverable between 10 and 30 minutes after the event. These events generally involved simple activities outside the control room, such as shifting strainers or operating valves.

- **Failure to recover instrument air before last chance to initiate main feedwater (AIR-XHE-RECOVERY-SW-LT).** The LOSWS event tree has the top event AIR-REC-SW-LT, "Recovery of service water within 60 min," which is a conditional probability of non-recovery given failure to recover service water (and instrument air pressure) within 30 minutes (a recovery event that appear earlier in the LOSWS event tree). A review of the 1986-1995 U.S. operating experience shows that six events involving the potential or conditional loss of service water were recoverable between 30 minutes and 60 minutes. These events involved simple and moderately complicated activities outside the control room, and could include some diagnosis.

c. **Calculations.** Table 4-9 gives a breakdown of the estimated times to solving the actual Service Water system problem. Each event was individually analyzed for recovery; the notes in Table 4-9 are the general rules applied to estimate recovery times. The uncertainties are quite high because the data used is not actually total LOSWS events. Table 4-10 gives a summary of the results of the analysis.

The first non-recovery event in the LOSWS event tree, recovery of service water and instrument air pressure within the short term (AIR-XHE-RECOVERY-SW-ST) is 0.88. The second non-recovery event in the LOSWS event tree, recovery of service water and instrument air pressure in 30 minutes (AIR-XHE-RECOVERY-SW-MT) is 0.83. The third and final recovery event in the LOSWS event tree, recovery of service water and instrument air pressure in 60 minutes (AIR-XHE-RECOVERY-SW-LT) is 0.68. The uncertainty of these events is best modeled using a lognormal distribution with an error factor of 2, based on the rules explained in Section 8.3.3.4 of NUREG/CR-6144, the Surry Shutdown PRA.

4.4 References

4-1. R. G. Neve, et al., Cost/Benefit Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure, NUREG/CR-5167, U.S. Nuclear Regulatory Commission, Washington, DC, April 1991.

4-2. Memorandum from Ashok C. Thadani to William D. Travers, "Closeout of Generic Safety Issue 23: Reactor Coolant Pump Seal Failure," U.S. Nuclear Regulatory Commission, November 8, 1999.

4-3. C. L. Atwood, "Constrained Noninformative Priors in Risk Assessment," Journal of Reliability Engineering and System Safety, Vol. 53, Issue 1, pp. 37-46, 1996.

4-4. AEOD/S98-01, Operating Experience Feedback From Service Water Systems Failures and Degradations (1986–1995), February 1998.

4-5. NUREG-1275, Vol.3, Operating Experience Feedback Report—Service Water System Failures and Degradations (1980-1985), November 1988.

4-6. *NRC Inspection Reports No. 50-266/01-17 and No. 50-301/01-17, February 21, 2002 (ADAMS Accession No. ML020950889).*

4-7. *Wisconsin Electric Power Company, Point Beach Nuclear Plant – Units 1 and 2, Individual Plant Examination, revised December 1997.*

4-8. *J. P. Poloski, et al., Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995, NUREG/CR-5750, U.S. Nuclear Regulatory Commission, Washington, DC, February 1999.*

Table 4-2. Loss of instrument air event summaries.

LER No.	Description	Affected Component	Disposition	Recovery
317/87-003 (Calvert Cliffs 1)	Operator inadvertently isolated Instrument air header from compressors resulting in loss of air pressure. Reactor manually tripped 9 min later on decreasing steam generator levels	Piping	Keep for frequency assessment	Valve manually opened 25 min after being closed, 16 min after reactor trip
346/87-015 (Davis-Besse)	Solenoid valve failed resulting in flow path of air to atmosphere on Instrument air dryer causing low Instrument air pressure. Reactor automatically tripped 13 min later. Dryer isolated 31 min after failure, 18 min after reactor trip. Lowest instrument air pressure was 74 psig	Dryer	Reject - local loss to selected components	NA
369/87-021 (McGuire 1)	Loss of offsite power (LOOP) resulting in loss of power to compressors. Power restored and compressors returned to normal service about 6 min after loss of power. Loss of air is result of LOOP which is analyzed as LOOP event	Compressors	Reject - LOOP	NA
400/87-041 (Shearon Harris)	Improper alignment of valves following return of dryer to service. Low Instrument air pressure led to FW transient and reactor scram. Valve lineup corrected and Instrument air pressure recovered 4 min after error, same time reactor trip occurred	Dryer, valves	Keep for frequency assessment	Instrument air recovered by time of trip (<1 min).
424/88-043 (Vogtle 1)	Valve on SA dryer failed to close allowing open path to atmosphere causing low SA and Instrument air pressure. Instrument air automatically isolated from SA and turbine building loads resulting in FW problems. Reactor manually tripped on decreasing steam generator levels. Total loss of Instrument air pressure did not occur	Dryer, valves	Reject - auto-isolation protected safety-related equipment	NA
456/88-025 (Braidwood 1&2)	Coupling failure on Instrument air header piping. Loss of air pressure leading to FW transient and manual reactor scram	Piping	Keep for frequency assessment	No time given when line isolated. Assume air recovery between 10 and [redacted] minutes.
457/88-019 (Braidwood 2)	Manual closure of Instrument air valve supplying Instrument air to various FW heater drain valves. FW transient and manual reactor scram. Local loss of Instrument air to components. Not a total loss of Instrument air system pressure	Valves	Reject - local loss to selected components	NA

LER No.	Description	Affected Component	Disposition	Recovery
247/89-002 (Indian Point 2)	Failed 0.5-in solder joint on air connection to valve air operator. Level control valve on FW heater drain tank failed open resulting in low level in drain tank and trip of drain tank pumps. Manual load rejection coupled with FW transient resulted in reactor trip on overpower/delta T. Local loss of instrument air only	Piping	Reject - local loss to selected components	NA
301/89-002 (Point Beach 2)	Loss of power event (bus undervoltage) resulted in trip of running compressor. Standby compressor was out of service for maintenance resulting in loss of Instrument air. Loss of air is result of LOOP which is analyzed as LOOP event	Compressors	Reject - LOOP	NA
280/90-006 (Surry 1)	Following loss of power to a 4160 V bus, Instrument air dryer failed allowing air to vent to atmosphere causing loss of Instrument air pressure. Reactor was manually tripped. Dryer manually bypassed about 8 min after failure, 2 min after reactor trip	Dryer	Keep for frequency assessment	Instrument air system recovered 8 minutes after failure, about 2 minutes after reactor trip
285/90-026 (Fort Calhoun 1)	Failed pipe joint on turbine building Instrument air header resulted in low Instrument air pressure and FW transient. Reactor manually tripped 6 min later. Leak manually isolated 10 min after failure, 4 min after reactor trip. Lowest instrument air pressure was 74 psig	Piping	Keep for frequency assessment - cannot determine if pressure was adequate for safety system operation.	Instrument air system was recovered 4 minutes after the trip.
270/92-004 (Oconee 2)	LOOP event resulting in loss of power to compressors. Loss of Instrument air occurred after scram. Loss of air is result of LOOP which is analyzed as LOOP event	Compressors	Reject - LOOP	NA
327/92-018 (Sequoyah 1)	Water intrusion in non-essential air lines lead to low system pressure and main FW reg. valve failure. Resulting FW transient caused turbine and reactor trip on high steam generator level. Essential air system automatically isolated from non-essential air system. Essential air system not affected	Receivers, piping	Reject - auto-isolation protected safety-related equipment	NA
530/92-001 (Palo Verde 3)	Solder joint failure on Instrument air valve drain line resulting in loss of Instrument air pressure to FW pumps' recirculation valves causing them to fail open. Manual trip of a FW pump caused power cutback and reactor scram. Low instrument air pressure was localized	Piping, valves	Reject - local loss to selected components	NA

LER No.	Description	Affected Component	Disposition	Recovery
244/95-002 (Ginna)	The Instrument air system isolated to containment due to air leak inside containment. SG levels fluctuated significantly due to loss of air, but were controlled manually to prevent a trip. Air operated equipment in containment (several valves and dampers) went to fail-safe positions.	Piping	Reject - did not cause a trip	NA
306/96-002 (Prairie Island 2)	Instrument air dryer purge exhaust valve failed to close during a drying cycle while one instrument air compressor was out of service for maintenance resulting in low Instrument air pressure and closure of FW reg. valves. Reactor tripped on low steam generator level. Operators quickly isolated the valve, but not before a reactor trip	Valves	Keep for frequency assessment	Instrument air was recovered shortly after the trip(<1 minute), but was not available at the time of the trip.
327/97-012 (Sequoyah 1)	Debris in isolation gate valve prevented isolation when modifications personnel cut into 6-inch control air header resulting in loss of control air and power runback. Reactor manually tripped	Piping, valves	Reject - auto-isolation protected safety-related equipment	Operations personnel isolated the breach within 7 minutes of reactor trip. Essential air system automatically isolated from control air system
334/01-001 (Beaver Valley)	Instrument air dryer blowdown valve transferred open (aging/cyclic fatigue of the spring). Resulting low Instrument air pressure caused the cooling water isolation valves for two reactor coolant pumps to fail closed. Reactor manually tripped. Dryer was isolated 15 minutes after first alarm for low Instrument air pressure, 8 min after reactor trip	Valves	Reject - auto-isolation protected safety-related equipment	NA
334/01-004 (Beaver Valley)	Maintenance on one air compressor caused a short and a blown fuse in a control circuit common to the redundant compressor. A backup diesel air compressor was started, but had insufficient capacity to maintain Instrument air pressure for normal reactor operation	Compressors	Reject - auto-isolation protected safety-related equipment	NA

Table 4-3. PWR commercial critical hours (1987-2001).[1, 2]

Plant Name	CY87-95	CY1996	CY1997	CY1998	CY1999	CY2000	CY2001	Totals (cr yr)
ARKANSAS 1	65632	7683.00	8760.00	7425.00	7962.80	7786.30	8171.80	12.9
ARKANSAS 2	65439	8083.50	8043.70	8009.70	7259.20	6176.50	8610.10	12.7
BEAVER VALLEY 1	61439	7196.10	5014.30	3617.80	7841.90	7498.00	7462.50	11.4
BEAVER VALLEY 2	62829	6239.20	7623.50	2251.80	7207.60	7838.50	8712.10	11.7
BRAIDWOOD 1	56640	7047.50	7377.70	7018.80	8688.60	8346.30	8287.70	11.8
BRAIDWOOD 2								
BYRON 1	66117	6666.20	6794.80	7174.10	7974.63	8362.37	8760.00	12.8
BYRON 2								
CALLAWAY	69048	7951.40	8760.00	7997.10	7768.70	8768.00	7592.00	13.5
CALVERT CLIFFS 1	51569	5869.40	8460.60	7235.50	8332.30	7674.00	8734.20	11.2
CALVERT CLIFFS 2	49357	8599.60	7148.10	8402.60	7452.60	8619.80	7334.30	11.1
CATAWBA 1	63257	5940.10	7968.70	8010.20	8006.10	7875.60	8741.70	12.5
CATAWBA 2								
COMANCHE PEAK 1	41128	7342.00	8667.00	7873.00	7966.00	8784.00	7848.70	10.2
COMANCHE PEAK 2								
COOK 1	62165	8595.00	4698.00	0.00	0.00	331.00	7905.00	9.6
COOK 2								
CRYSTAL RIVER 3	60116	3307.30	0.00	7908.50	7691.00	8574.50	7817.20	10.9
DAVIS-BESSE	62650	7490.10	8236.20	7291.80	8374.30	7664.60	8760.00	12.6
DIABLO CANYON 1	67266	8346.80	7746.80	8575.20	7837.50	7598.60	8760.00	13.3
DIABLO CANYON 2								
FARLEY 1	69610	8755.30	6836.90	6621.60	8722.20	6812.30	7808.20	13.2
FARLEY 2	67547	7213.60	8760.00	7551.80	7290.40	8750.70	6976.10	13.0
FORT CALHOUN	63477	6983.60	8200.90	7228.30	7812.40	8213.10	7735.00	12.5
GINNA	67663	6247.30	8044.80	8760.00	7494.80	8048.30	8760.00	13.1
HADDAM NECK	54112	4896.30	0.00					6.7
HARRIS	64651	8378.90	7043.60	7915.87	8574.95	8127.66	6342.68	12.7
INDIAN POINT 2	59984	8325.60	3815.70	2758.45	7723.48	1143.67	8513.98	10.5
INDIAN POINT 3	39915	6569.10	4813.60	8345.78	7736.31	8722.98	8156.38	9.6
KEWAUNEE	68864	6305.40	4906.70	7642.70	8760.00	7803.30	7080.80	12.7
MAINE YANKEE	56909	6864.50	0.00					7.3
MCGUIRE 1	58412	7952.60	6481.20	7931.90	7613.00	8756.20	7756.90	12.0

Plant Name	CY87-95	CY1996	CY1997	CY1998	CY1999	CY2000	CY2001	Totals (cr yr)
MCGUIRE 2								
MILLSTONE 2	51550	1223.60	0.00	0.00	5445.90	7453.30	8656.20	8.5
MILLSTONE 3	57644	2158.80	0.00	3665.20	7403.70	8784.00	7466.10	9.9
NORTH ANNA 1	63572	6897.60	8760.00	8073.00	8080.00	8751.10	6830.70	12.7
NORTH ANNA 2								
OCONEE 1	68596	6745.40	4569.40	7303.80	7521.00	7468.90	8416.40	12.6
OCONEE 2								
OCONEE 3								
PALISADES	51161	7145.90	7872.40	7314.10	7026.90	7732.90	3184.10	10.4
PALO VERDE 1	50463	7362.20	8671.10	7872.90	7836.90	8784.00	7739.00	11.3
PALO VERDE 2	53637	7598.60	7692.00	8760.00	7893.60	7794.90	8027.30	11.6
PALO VERDE 3	52680	8763.80	7868.90	7869.90	8760.00	7941.60	7526.55	11.6
POINT BEACH 1	69290	8220	1918	5555	7171	8412	7680	12.4
POINT BEACH 2								
PRAIRIE ISLAND 1								
PRAIRIE ISLAND 2	71334	8684.20	7233.00	6595.90	8725.80	7858.20	8092.00	13.5
ROBINSON 2	55657	7783.30	8670.00	7786.25	8040.05	8759.96	7935.38	11.9
SALEM 1	53098	0.00	0.00	6440.00	7724.00	8464.00	7219.00	9.5
SALEM 2								
SAN ONOFRE 2	64027	8016.70	6265.10	7812.00	7506.00	7846.00	8556.00	12.6
SAN ONOFRE 3								
SEABROOK	40930	8744.90	6988.60	7399.40	7665.90	6965.70	7809.40	9.9
SEQUOYAH 1	44449	8428.20	7533.00	8021.40	8760.00	7056.40	8012.12	10.5
SEQUOYAH 2								
SOUTH TEXAS 1	44302	8242.20	8156.60	8760.00	7900.20	6983.20	8270.40	10.6
SOUTH TEXAS 2	39638	8453.70	8237.10	8210.10	8147.80	8784.00	7880.70	10.2
ST. LUCIE 1	65467	6627.70	6858.70	8443.20	7848.50	8784.00	7998.50	12.8
ST. LUCIE 2								
SUMMER	64696	7928.90	7870.40	8656.70	7830.30	6724.90	7295.40	12.7
SURRY 1	59506	8784.00	7191.10	7290.70	8760.00	8227.60	7457.50	12.2
SURRY 2	57392	7572.70	8074.70	8760.00	7618.80	8059.80	8229.10	12.1

Plant Name	CY87-95	CY1996	CY1997	CY1998	CY1999	CY2000	CY2001	Totals (cr yr)
THREE MILE IS 1	69458	8784.00	7384.60	8760.00	7857.70	8784.00	7372.90	13.5
TROJAN	28096							3.2
TURKEY POINT 3	50841	8545.50	7666.30	7792.20	8707.40	8142.40	8020.00	11.4
TURKEY POINT 4								
VOGTLE 1	67799	7248.80	7257.20	8760.00	8131.70	8011.80	8734.10	13.2
VOGTLE 2								
WATERFORD 3	67152	8271.50	6238.80	8046.80	6969.50	7797.60	8727.00	12.9
WATTS BAR 1	0	4825.40	7341.60	8683.50	7642.90	8155.30	8450.30	5.1
WOLF CREEK	65303	7135.20	7343.30	8760.00	7890.10	7859.40	8760.00	12.9
YANKEE-ROWE	34595							3.9
ZION 1	50855	7464.10	1238.30	0.00	0.00	0.00	0.00	6.8
ZION 2								
TOTALS	**3189194**	**382464.6**	**339100.4**	**364939.55**	**392957.12**	**394663.43**	**402973.89**	**624.0**

Notes:

1. Data for BWRs was excluded due to plant differences in instrument air systems. Multi-unit sites with shared instrument air systems (from UFSARs) counted once—unit with longest critical time.
2. Data for calendar years 1987–1995 was taken from NUREG/CR-5750 (Ref. 4-8). Data for calendar years 1995–2001 were calculated from Idaho National Engineering and Environmental Laboratory's database (MORP1.DBF). Data is based on licensee's monthly operating reports as of December 2002.

Table 4-4. Frequency for loss of instrument air.

Constrained Non-informative Prior		Point Beach Data		Posterior	
Failures =	6	Failures =	0		
Exposure[1] =	611.6 yr	Exposure =	12.4 yr		
a =	0.5			a' =	0.5
1/b =	51.0			1/b' =	63.4
95^{th}	3.8E-02/yr			95^{th}	3.0E-02/yr
Mean	9.8E-03/yr			**Mean**	**7.9E-03/yr**
Median	4.5E-03/yr			Median	3.6E-03/yr
5^{th}	3.9E-05/yr			5^{th}	3.1E-05/yr

Note:

1. National (generic) total exposure is 624 yr (see Table 4-3).
2. Constrained, non-informative prior used because of expected variation in instrument air systems (relatively dissimilar systems between plants).

Table 4-5. Industry commercial critical hours (1987-2001).[1, 2]

	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001
Critical Hours															
PWR	417775.7	466182.3	461652.3	474942.9	504981.8	512763.6	491488.7	518225.2	518681.1	515809.3	463214.3	499729.5	529114.5	538829.8	546269.5
BWR	197489.5	199293.3	204484.8	231608.8	230335.2	221641.0	234735.5	233389.0	259566.2	249177.9	236965.5	239544.1	265672.3	277399.2	276843.9
TOTAL	615265.2	665475.6	666137.1	706551.7	735317.0	734404.6	726224.2	751614.2	778247.3	764987.2	700179.8	739273.6	794786.8	816229.0	823113.4
Critical Years															
PWR	47.69	53.07	52.70	54.22	57.65	58.37	56.11	59.16	59.21	58.72	52.88	57.05	60.40	61.34	62.36
BWR	22.54	22.69	23.34	26.44	26.29	25.23	26.80	26.64	29.63	28.37	27.05	27.35	30.33	31.58	31.60
TOTAL	70.24	75.76	76.04	80.66	83.94	83.61	82.90	85.80	88.84	87.09	79.93	84.39	90.73	92.92	93.96
Totals															
	Critical Years														
PWR	850.93														
BWR	405.88														
TOTAL	**1256.81**														

Notes:

1. Data were calculated from Idaho National Engineering and Environmental Laboratory's database (MORP1.DBF). Data is based on licensee's monthly operating reports as of December 2002..
2. Data are included from critical date until permanent shutdown. Ft. St. Vrain critical hours are excluded.

Table 4-6. Frequency for loss of service water.

Jeffries Non-informative Prior[1]		National Experience Data		Posterior	
Failures =	0.5	Failures =	0		
Exposure =	0 yr	Exposure[2] =	1257 yr		
a =	0.5			a' =	0.5
1/b =	0			1/b' =	1257
95th	–[3]			95th	1.5E-03/yr
Mean	–[3]			**Mean**	**4.0E-04/yr**
Median	–[3]			Median	1.8E-04/yr
5th	–[3]			5th	1.6E-06/yr

Note:
1. Jeffries non-informative prior used because no failures experienced.
2. National (generic) total exposure is 1257 yr (see Table 4-5).
3. Not calculated because Jeffries non-informative prior is an improper gamma distribution.

Table 4-7. Recovered LOIA events.[1]

Plant/LER	Recovery time Total time (min)	Reactor trip
Harris (400/87-041)	4	automatic
Calvert Cliffs (317/87-003)	25	manual
Braidwood 1&2 (456/88-025)	[redacted][2]	manual
Surry (280/90-006)	8	manual
Ft Calhoun (285/90-026)	10	manual
Prairie Island 2 (306/96-002)	10	automatic

Notes:
1. There are 6 total events for recovery of instrument air. For Braidwood, it is consider to be only one event and not two events, since only one recovery action is needed.
2. No specific recovery time provided in LER or event notification (50.72). Engineering evaluation determined the air line break could have been isolated within [redacted] minutes, given the need.

Table 4-8. Non-recovery probabilities for recovery of loss of instrument air.

Time	Recovery Probability[1]	Non-recovery Probability	Conditional Non-recovery Probability
10	0.42	1 - 0.42 = 0.58	0.58
30	0.90	1 - 0.90 = 0.10	0.10/0.58 = 0.17
60	0.99	1 - 0.99 = 0.014	0.014/0.10 = 0.14

Notes:
1. From a log normal distribution with the mean of 15.3224 and standard deviation of 13.3008 or an error factor of 3.43.

Table 4-9. Service water events between 1986 and 1995.

Time to fix problem in the service water system (minutes)[1]	AEOD Category		
	Category 2 - Operating	Category 2 - Shutdown	Category 3
Recovery[2] < 10		Oconee 1 (86-11)	Arkansas 2 (91-12) Catawba 1 (93-002)
10 < Recovery < 30[3]	Shearon Harris (88-12)	River Bend 1 (89-20)	Callaway 1 (87-18) Haddam Neck (92-015)
30 < Recovery < 60[4]			Waterford 3 (93-006) Haddam Neck (93-017) Callaway 1 (87-18) Crystal River 3 (90-18) Haddam Neck 1 (91-17) McGuire 1 (91-14)
Recovery > 60[5]	Dresden 2 (89-28) Surry 1 (90-12) Indian Point (93-009)	Salem 2 (89-01) Turkey Pt. 3 (89-01)	Oyster Creek (94-010) Oyster Creek (94-015) Crystal River 3 (94-013) Vermont Yankee (94-013) McGuire (87-17) Millstone 1 (88-07) Indian Point 2 (89-11) Millstone 1 (90-16)

Notes
1. In all cases, a cross connection or other source of service water cooling was available.
2. Problem was fixable from the control room by shifting / starting pumps.
3. Problem was fixable with direct action in the plant. Shifting strainers falls into this category.
4. Problem was fixable with troubleshooting, followed by direct action in the plant. Cleaning strainers falls into this category.
5. Problem required significant diagnosis and/or corrective action in the plant.

Table 4-10. Non-recovery probabilities for recovery of loss of service water.

Time	Recovery Probability[1]	Non-recovery Probability	Conditional Non-recovery Probability
10	3/26 = 0.12	1 - 0.12 = 0.88	0.88
30	(3+4)/26 = 0.27	1 - 0.27 = 0.73	0.73/0.88 = 0.83
60	(3+4+6)/26 = 0.5	1 - 0.5 = 0.5	0.5/0.73 = 0.68

Notes:

1. Actual data used for recovery probabilities.

5. COMMON CAUSE FAILURE MODEL

No changes made to this section.

6. GEM AND GEMDATA INTERACTIONS

6.3.8 RCP Seal Model

The RCP seal model used for calculating the RCP seal leak for Point Beach Unit 1 and 2 was the Westinghouse ~~old~~ ***new*** o-ring seal design. ***The failure probability values for RCP-MDP-SEALS and RCP-MDP-SEALS2 were set to 0.22 (Refs. 6-9 and 6-10).***

6.4 References

6-1. K.D. Russell, et al., *Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 5.0, Volume 6 - Graphical Evaluation Module (GEM) Reference Manual*, NUREG/CR-6116, February 28, 1995.

6-2 K.J. Kvarfordt, et al., *GEMDATA Reference Manual*, INEL-95/0040, March 16, 1995.

6-3. P.W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

6-4. J.W. Minarick, *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989.

6-5. D. M. Ericson, et al, *Analysis of Core Damage Frequency: Internal Events Methodology*, NUREG/CR-4550, Vol. 1, Rev. 1, January 1990.

6-6. Belles, R. J., et al, *Precursors to Potential Severe Core Damage Accidents: 1994 A Status Report*, NUREG/CR-4674, Vol. 21.

6-7. M. B. Sattison, e-mail 4/14/99 to Ed Rodrick, NRC, Subject: Westinghouse RCP Seal Models.

6-8. G. M. Grant, et al., *Emergency Diesel Generator Power System Reliability, 1987-1993*, INEL-95/0035, February 1996.

6-9. *R. G. Neve, et al., Cost/Benefit Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure, NUREG/CR-5167, U.S. Nuclear Regulatory Commission, Washington, DC, April 1991.*

6-10. *Memorandum from Ashok C. Thadani to William D. Travers, "Closeout of Generic Safety Issue 23: Reactor Coolant Pump Seal Failure," U.S. Nuclear Regulatory Commission, November 8, 1999.*

7. HUMAN RELIABILITY MODEL

Changes to basic events involving human error are listed in Table 7-1. The HEPs calculated using the SPAR Human Reliability Analysis (HRA) method are documented using HRA Worksheets that are provided in the tables that follow this section. These worksheets revise or supplement the worksheets in Appendix H in the original SPAR model manual. Changes to operator error events that pose dependency considerations are listed in Table 7-2. SAPHIRE recovery rules are used to make the basic event substitutions required to implement the formal dependency calculations. Changes to SAPHIRE rule file are provided in Listing 1 at the end of this section.

Table 7-1. Point Beach Unit 1 and 2 PWR B human action summary.

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
ACP-XHE-NOREC-ST	Operator fails to recover offsite power in short term	3.60E-01[a]	(data)
ACP-XHE-NOREC-BD	Operator fails to recover ac power before battery depletion	1.70E-02[a]	(data)
AFW-XHE-XA-PSWA	Operator fails to align backup water source (ATWS)	4.00E-03[c]	*Lognormal(3)*
AFW-XHE-XA-PSW	Operator fails to align backup water source	1.00E-03[c]	*Lognormal(3)*
AFW-XHE-XA-PSWS	Operator fails to align backup water source (SGTR)	4.00E-03[c]	*Lognormal(3)*
AFW-XHE-XL-MDPFR	Operator fails to recover AFW MDP (fails to run)	7.50E-01[b]	Beta(0.5)
AFW-XHE-XL-MDPFS	Operator fails to recover AFW MDP (fails to start)	2.10E-01[b]	Beta(5.5)
AFW-XHE-XL-TDPFR	Operator fails to recover AFW TDP (fails to run)	8.80E-01[b]	Beta(0.5)
AFW-XHE-XL-TDPFS	Operator fails to recover AFW TDP (fails to start)	4.70E-01[b]	Beta(9.5)
AFW-XHE-XL-TDPFR2	Operator fails to recover AFW TDP 2P29 (fails to run)	8.80E-01[b]	Beta(0.5)
AFW-XHE-XL-TDPFS2	Operator fails to recover AFW TDP 2P29 (fails to start)	4.70E-01[b]	Beta(9.5)
AFW-XHE-XM-RECIRC	Operator fails to re-establish AFW recirc	1.00E-03[c]	*Lognormal(3)*
AFW-XHE-XM-MDP38B	Operator fails to cross-tie unit 2 AFW MDP-38B	2.00E-02[c]	*Lognormal(3)*
AFW-XHE-XO-TDP	Operator fails to control AFW TDP during SBO	2.50E-02[c]	*Lognormal(3)*
CCW-XHE-XA-HX	Operator fails to align standby heat exchanger	1.00E-03[c]	*Lognormal(3)*
CCW-XHE-XA-UNIT2	Operator fails to align unit 2 CCW to unit 1	True	na
CCW-XHE-XL-CCWSYS	Operator fails to recover from loss of component cooling water	True	na
CCW-XHE-XR-HTX12B	Operator fails to restore CCW HTX-P12B after T&M	1.00E-03[d]	*Lognormal(3) XHXR*

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
CCW-XHE-XR-MDP11B	Operator fails to restore CCW MDP-P11B after T&M	1.00E-03[d]	*Lognormal(3) XHXR*
COND-XHE-XM-IA	***Operator fails to initiate condensate injection***	***4.0E-02[a]***	***Lognormal (3)***
COND-XHE-XM-IA	***Operator fails to initiate condensate injection (dependent)***	***8.8E-02[a]***	***Lognormal (3)***
COND-XHE-XM-IA	***Operator fails to initiate condensate injection (dependent)***	***1.8E-01[a]***	***Lognormal (2)***
CVC-XHE-XM-BOR	Operator fails to initiate emergency boration	2.00E-02[c]	*Lognormal (3)*
CVC-XHE-XM-LETDOWN	Operator fails to initiate/establish letdown cooling	1.00E-03[c]	*Lognormal (3)*
EPS-XHE-XM-G05-1B	Operator fails to align Gas Turbine (G05) to 1A06	1.00E-01[c]	*Lognormal (2)*
EPS-XHE-XM-G02-1A	Operator fails to align alternate EDG (G02) to 1A05	2.00E-02[c]	*Lognormal (3)*
EPS-XHE-XM-G04-1B	Operator fails to align alternate EDG (G04) to 1A06	2.00E-02[c]	*Lognormal (3)*
EPS-XHE-XM-G05-1A	Operator fails to align Gas Turbine (G05) to 1A05	1.00E-01[c]	*Lognormal (2)*
EPS-XHE-XM-G05-2B	Operator fails to align Gas Turbine (G05) to 2A06	1.00E-01[c]	*Lognormal (2)*

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
EPS-XHE-XM-G05-2A	Operator fails to align Gas Turbine (G05) to 2A05	1.00E-01[c]	***Lognormal (2)***
EPS-XHE-XM-G05	Operator fails to align Gas Turbine (G05) to essential bus	1.300E-01[i]	(IPE)
EPS-XHE-XM-G01-2A	Operator fails to align alternate EDG (G01) to 2A05	2.00E-02[c]	***Lognormal (3)***
EPS-XHE-XM-G03-2B	Operator fails to align alternate EDG (G03) to 2A06	2.00E-02[c]	***Lognormal (3)***
FWS-XHE-XM-FWSCST	Operator fails to align fire water to refill CST	1.20E-01[c]	***Lognormal (2)***
FWS-XHE-XR-35A	Op fails to restore fire water motor-driven pump 35A	1.00E-03[c]	***Lognormal (3)***
FWS-XHE-XR-35B	Op fails to restore fire water engine-driven pump 35B	1.00E-03[c]	***Lognormal (3)***
HPI-XHE-XM-FB	Operator fails to initiate feed and bleed cooling	2.00E-02[c]	***Lognormal (3)***
HPI-XHE-XM-FB1	Operator fails to initiate feed and bleed cooling (dependent)	6.90E-02[c]	***Lognormal (3)***
HPI-XHE-XM-FB2	Operator fails to initiate feed and bleed cooling (dependent)	1.60E-01[c]	***Lognormal (2)***
HPI-XHE-XM-THRTL	Operator fails to throttle HPI to reduce pressure	1.00E-03[c]	***Lognormal (3)***

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
HPI-XHE-XR-MDP1A	Operator fails to restore HPI MDP-1A after T & M	1.00E-03[d]	***Lognormal*** ***(3)*** ***XHXR***
HPI-XHE-XR-MDP1B	Operator fails to restore HPI MDP-1B after T & M	1.00E-03[d]	***Lognormal*** ***(3)*** ***XHXR***
HPI-XHE-XM-DIS	Operator fails to open Rx vessel MOVs for injection	1.00E-03[e]	***Lognormal*** ***(3)***
HPR-XHE-XM	Operator fails to initiate HPR	2.00E-03[e]	***Lognormal*** ***(3)***
HPR-XHE-XM1	Operator fails to initiate HPR system (dependent)	5.20E-02[e]	***Lognormal*** ***(3)***
ISL-XHE-DETECT	Operator fails to diagnose and detect system ISLOCA	1.00E-02[e]	(NUREG/CR-5745)
LPI-XHE-XM-RCOOL	Operator fails to rapidly depressurize RCS to LPI injection	4.00E-02[e]	***Lognormal*** ***(3)***
LPI-XHE-LOCA	Operator fails to isolate LOCA and realign train	True	na
LPR-XHE-XM	Operator fails to initiate LPR	1.00E-02c	***Lognormal*** ***(3)***
MFW-XHE-NOREC	Operator fails to recover (restore) MFW	2.00E-01[f]	(???)
MFW-XHE-ERROR	Operator fails to restore MFW flow	4.00E-02[e]	***Lognormal*** ***(3)***

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
MFW-XHE-ERROR1	Operator fails to restore MFW flow (dependent)	8.80E-02[c]	***Lognormal (3)***
MSS-XHE-XM-BLK	Operator fails to close ADV block valve	2.00E-02[c]	***Lognormal (3)***
MSS-XHE-XM-BLK1	Operator fails to close ADV block valve (dependent)	6.90E-02[c]	***Lognormal (3)***
MSS-XHE-XM-ERROR	Operator fails to isolate faulted steam generator	2.00E-03[c]	***Lognormal (3)***
MSS-XHE-XM-ERROR1	Operator fails to isolate faulted steam generator (dependent)	5.20E-02[c]	***Lognormal (3)***
OEP-XHE-NOREC-SL	Operator fails to recover offsite power (seal LOCA)	8.40E-01[a]	(data)
OEP-XHE-NOREC-2H	Operator fails to recover offsite power within 2 hrs	7.50E-02[a]	(data)
OEP-XHE-NOREC-6H	Operator fails to recover offsite power within 6 hrs	3.60E-02[a]	(data)
PCS-XHE-XM-RCOOL	Operator fails to initiate RCS cooldown below RHR	1.00E-03[c]	***Lognormal (3)***
PCS-XHE-XM-RCOOL1	Operator fails to initiate RCS cooldown below RHR (dependent)	5.10E-02[c]	***Lognormal (3)***
PCS-XHE-XM-RCOOL2	Operator fails to initiate RCS cooldown below RHR (dependent)	1.44E-01[c]	***Lognormal (2)***
PCS-XHE-XO-SEC	Operator fails to establish secondary cooling	2.00E-01[g]	(???)
PCS-XHE-XO-SECL	Operator fails to establish secondary cooling during LOOP	3.40E-01[g]	(???)

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
PCS-XHE-XM-CDOWN	Operator fails to initiate COOLDOWN	1.00E-03[c]	***Lognormal (3)***
PPR-XHE-XM-BLK	Operator fails to close block valve	2.00E-03[c]	***Lognormal (3)***
RCS-XHE-DIAG	Operator fails to diagnose SGTR to start procedures	8.00E-03[c]	***Lognormal (3)***
RCS-XHE-RECOVER	Operator fails to depressurize RCS below SG SRV given ADV lifts	2.00E-02[c]	***Lognormal (3)***
RCS-XHE-XM-SG	Operator fails to initiate RCS depressurization	2.00E-03[c]	***Lognormal (3)***
RHR-XHE-XM2	Operator fails to initiate the RHR system (dependent)	1.45E-01[c]	***Lognormal (2)***
RHR-XHE-XM	Operator fails to initiate the RHR system	2.00E-03[c]	***Lognormal (3)***
RHR-XHE-XM1	Operator fails to initiate the RHR system (dependent)	5.20E-02[c]	***Lognormal (3)***
RHR-XHE-XE-ISOL	Operator fails to isolate RHR suction ISLOCA	1.00E+00[e]	(NUREG/CR-5745)
RHR-XHE-XE-ISOLDIS	Operator fails to isolate RHR discharge ISLOCA	1.00E+00[e]	(NUREG/CR-5745)
RHR-XHE-XM-DETECT	Operator fails to detect RHR discharge ISLOCA	6.07E-02[e]	(NUREG/CR-5745)

Human Error Event	Event Description	Mean Probability	Uncertainty Parameter
RHR-XHE-XM-DIAG	Operator fails to diagnose RHR discharge ISLOCA	6.70E-02[e]	(NUREG/CR-5745)
RHR-XHE-XR-HTXA	Operator fails to restore RHR HTX-11A after T&M	1.00E-03[d]	***Lognormal (3) XHXR***
RHR-XHE-XR-HTXB	Operator fails to restore RHR HTX-11B after T&M	1.00E-03[d]	***Lognormal (3) XHXR***
RHR-XHE-XR-MDP1A	Operator fails to restore RHR MDP-1A after T&M	1.00E-03[d]	***Lognormal (3) XHXR***
RHR-XHE-XR-MDP1B	Operator fails to restore RHR MDP-1B after T&M	1.00E-03[d]	***Lognormal (3) XHXR***
RPS-XHE-XM-SCRAM	***Operator fails to manually trip the reactor (Modification under INEEL review)***	***5.00E-03[c]*** ~~***1.00E-02[b]***~~	***Lognormal (3)***
RPS-XHE-XL-REC	***Operator fails to de-energize MG sets (Modification under INEEL review)***	***5.00E-03[c]*** ~~***4.40E-01[c]***~~	***Lognormal (3)***
SI-XHE-XE-ISOLCL	Operator fails to isolate SI cold leg discharge ISLOCA	2.50E-02[e]	(NUREG/CR-5745)
SI-XHE-XE-ISOLHL	Operator fails to isolate SI hot leg discharge ISLOCA	1.00E+00[e]	(NUREG/CR-5745)
SLOCA-XHE-NOREC	Operator fails to recover from a SLOCA in short term	4.30E-01[g]	(???)
SWS-XHE-XL-SWSSYS	Operator fails to recover from a loss of service water	True	na

Notes:

a. Station Blackout Model calculations described in more detail in Section 6.
b. Value obtained from Reference 7-5.
c. HEP calculated using a SPAR Model Human Error Worksheet. The calculation worksheets are provided in Appendix H.
d. The operator fails to restore component after T&M used the screening value of 1.0E-3 from the SPAR Model Human Error Worksheet.
e. Value obtained from Reference 7-7.
f. Recovery of MFW value obtained from Catawba IPE.
g. Value obtained from the draft Daily Events Evaluation Manual.
h. Value from ~~S.A. Eide, et al., *Reliability Study: Westinghouse Reactor Protection System*, 1984-1995,~~ NUREG/CR-5500 Vol. 2 ~~April 1999~~ ***(Reference 7-8)***.
i. Value from Point Beach IPE ***(Reference 7-9).***

Table 7-2. Changes to dependent human actions.

Original Events	Dependent Events	Dependent Value
AFW-XHE-XA-PSW * HPI-XHE-XM-FB	HPI-XHE-XM-FB1	6.9E-02
AFW-XHE-XM-MDP38B * HPI-XHE-XM-FB	HPI-XHE-XM-FB1	6.9E-02
RHR-XHE-XM * HPR-XHE-XM	HPR-XHE-XM1	5.2E-02
AFW-XHE-XM-PSW * HPI-XHE-XM-FB* COND-XHE-XM-IA	***HPI-XHE-XM-FB1 COND-XHE-XM-IA2***	***6.9E-02 1.8E-01***
HPI-XHE-XM-FB * COND-XHE-XM-IA	***COND-XHE-XM-IA1***	***8.8E-02***
COND-XHE-XM-IA* HPR-XHE-XM	***HPR-XHE-XM1***	***5.2E-02***

Listing 1. SAPHIRE Project recovery rules.

```
| The following list is dependency correction factors that are
|required in order to take into account the dependency of the
|operator failures in a sequence cut set.  By using value events,
|the original operator action names are not removed or renamed to
|take account for their dependencies.

elsif AFW-XHE-XM-MDP38B * HPI-XHE-XM-FB then
      DeleteEvent = HPI-XHE-XM-FB;
      AddEvent = HPI-XHE-XM-FB1;

elsif RHR-XHE-XM * HPR-XHE-XM then
      DeleteEvent = HPR-XHE-XM;
      AddEvent = HPR-XHE-XM1;

elsif AFW-XHE-XA-PSW * HPI-XHE-XM-FB then
      DeleteEvent = HPI-XHE-XM-FB;
      AddEvent = HPI-XHE-XM-FB1;

elsif AFW-XHE-XA-PSW * COND-XHE-XM-IA * HPI-XHE-XM-FB then
      DeleteEvent = HPI-XHE-XM-FB;
      AddEvent = HPI-XHE-XM-FB1;
      DeleteEvent = COND-XHE-XM-IA;
      AddEvent = COND-XHE-XM-IA2;

elsif COND-XHE-XM-IA * HPI-XHE-XM-FB then
      DeleteEvent = COND-XHE-XM-IA;
      AddEvent = COND-XHE-XM-IA1;
```

elsif COND-XHE-XM-IA * HPR-XHE-XM then
DeleteEvent = HPR-XHE-XM;
AddEvent = HPR-XHE-XM1;

7.3 References

7-1. H.S. Blackman and J.C. Byers, *ASP Human Reliability Methodology Development*, INEL-95/0139, April 1995.

7-2. A.D. Swain, and H. Gutman, *Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*, NUREG/CR-1278, August 1983. (THERP)

7-3. R.J. Belles, et al., *Precursors to Potential Severe Core Damage Accidents: 1994 A Status Report*, NUREG/CR-4674, Vol. 21, December 1995, pg. A.1-5.

7-4. K.D. Russell, et al., Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) Version 5.0, NUREG/CR-6116, Vol. 2, December 1993.

7-5. J.P. Poloski, et al, *Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995*, NUREG/CR-5500 Vol. 1, August 1998.

7-6. SPAR HRA Reference.

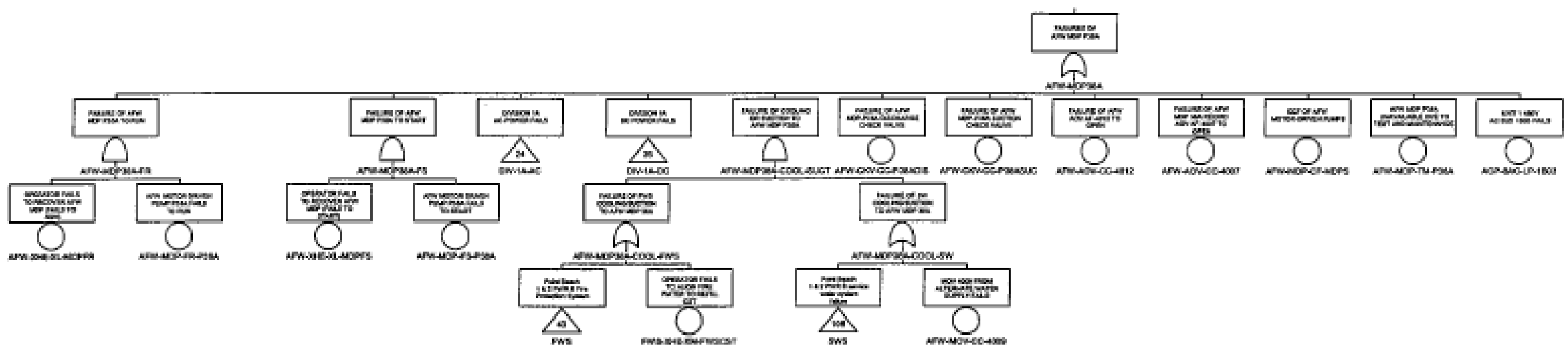
7-7. D.L. Kelly, et. al., *Assessment of ISLOCA Risk-Methodology and Application to a Combustion Engineering Plant*, NUREG/CR-5745, April 1992.

7-8. S.A. Eide, et al., Reliability Study: Westinghouse Reactor Protection System, 1984-1995, NUREG/CR-5500 Vol. 2 April 1999.

7-9. Wisconsin Electric Power Company, Point Beach Nuclear Plant – Units 1 and 2, Individual Plant Examination, revised December 1997.

Appendix A
Fault Trees

Fault Tree Index



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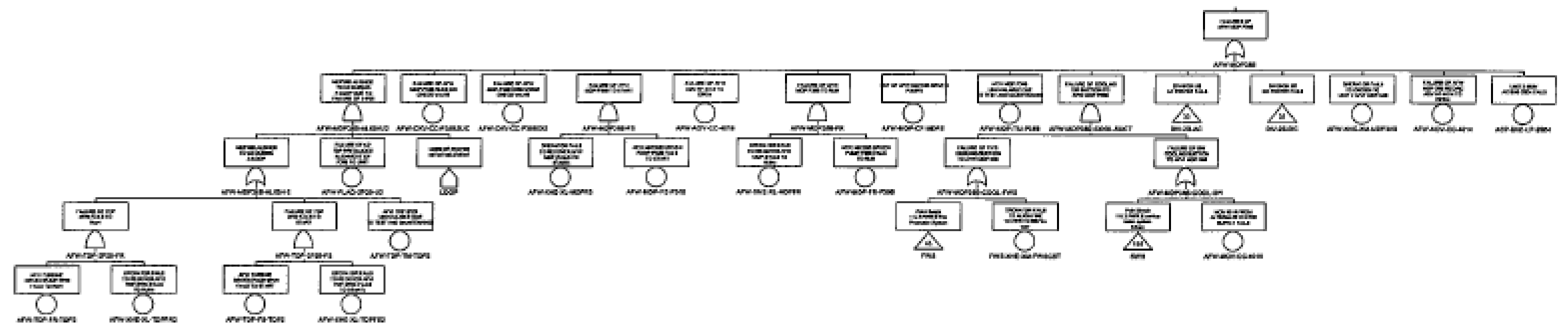
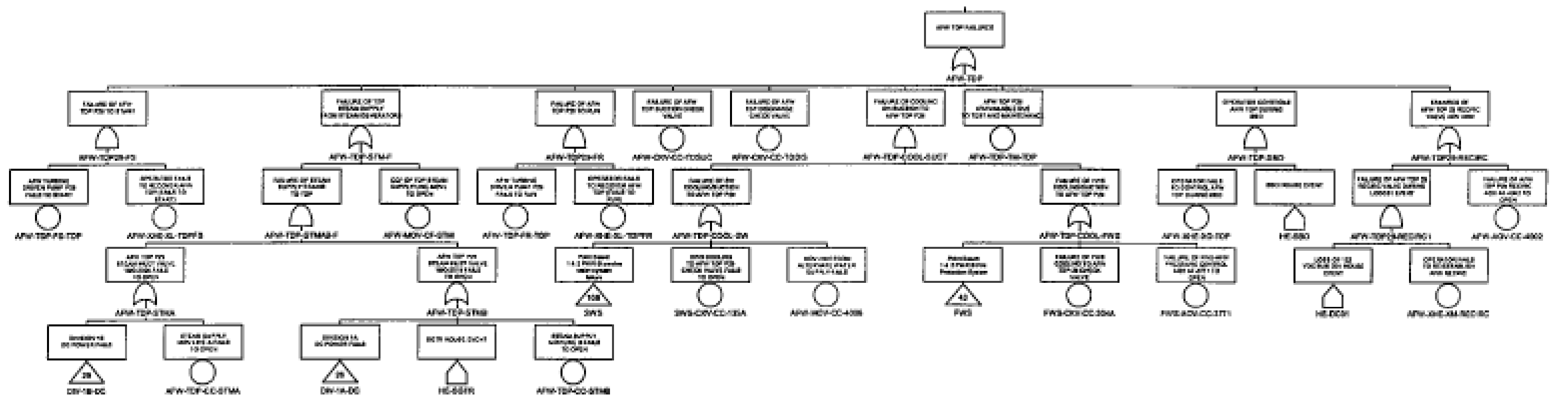
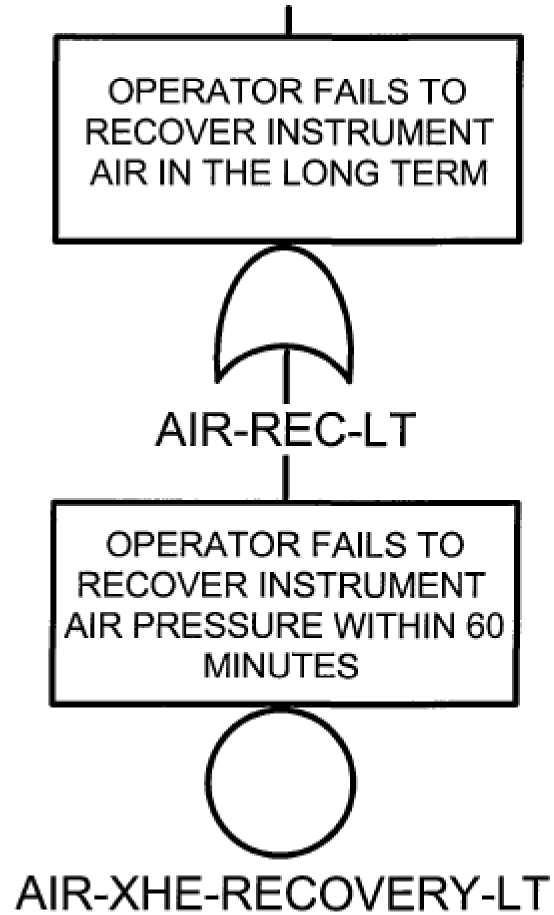


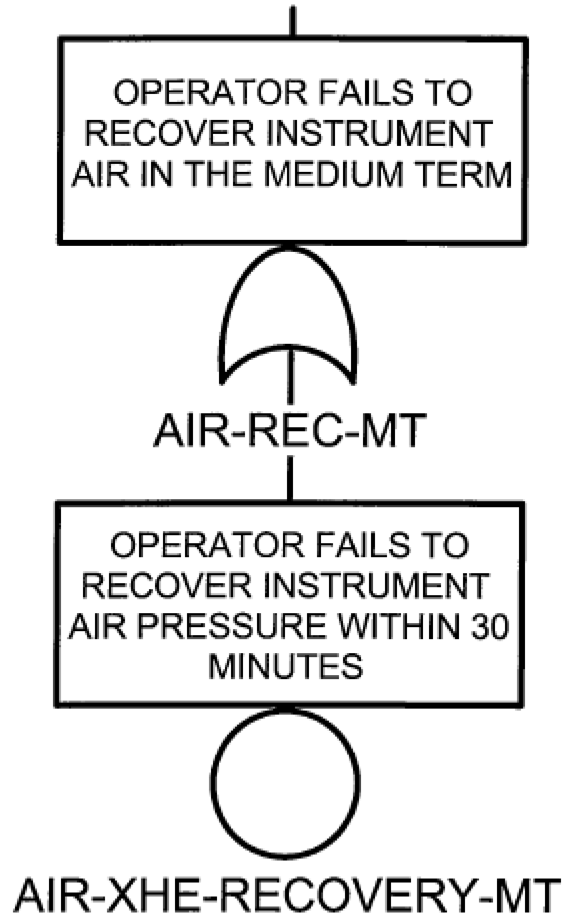
AFW-MDP38B - Point Beach 1 & 2 PWR B AFW MDP P38B failures 2003/01/14 Page 7



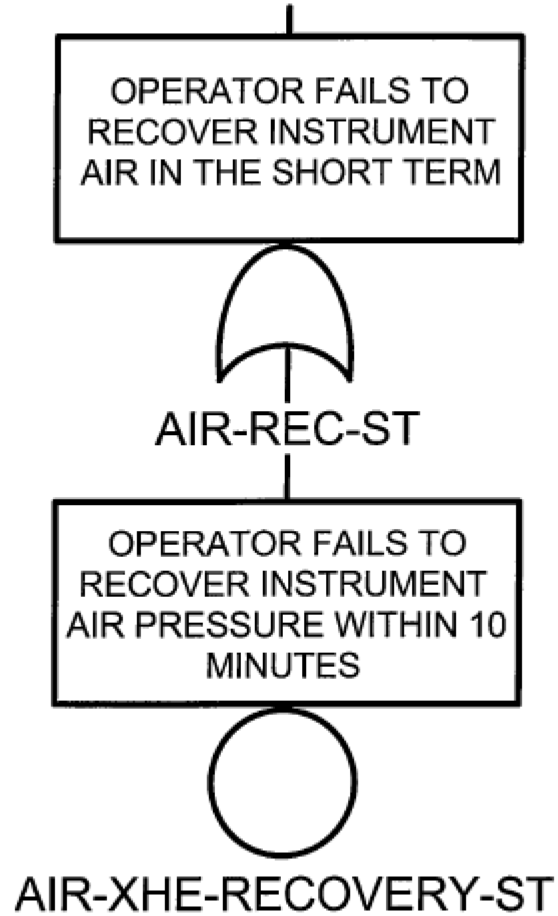
AFW-TDP - Point Beach 1 & 2 PWR B AFW TDP failures 2003/01/14 Page 9



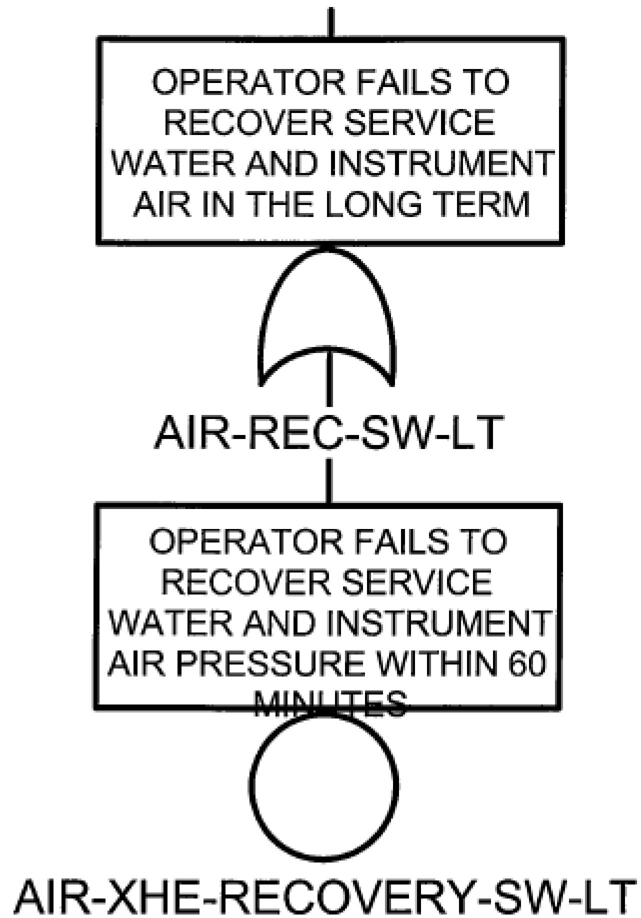
AIR-REC-LT - Point Beach 1 & 2 PWR Inst air not recovered - long term 2002/12/11 Page 10



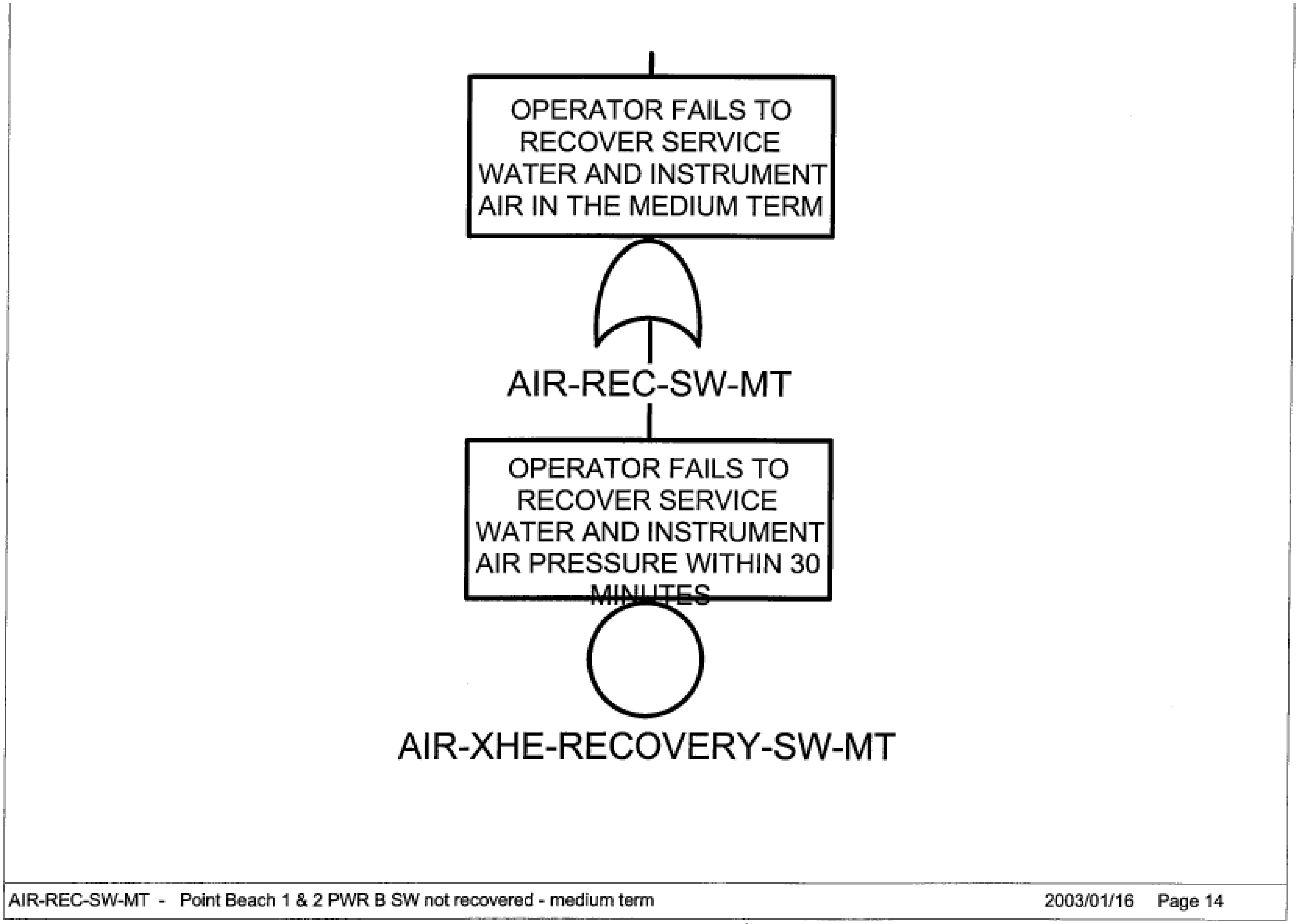
AIR-REC-MT - Point Beach 1 & 2 PWR Inst air not recovered - medium term 2002/12/11 Page 11

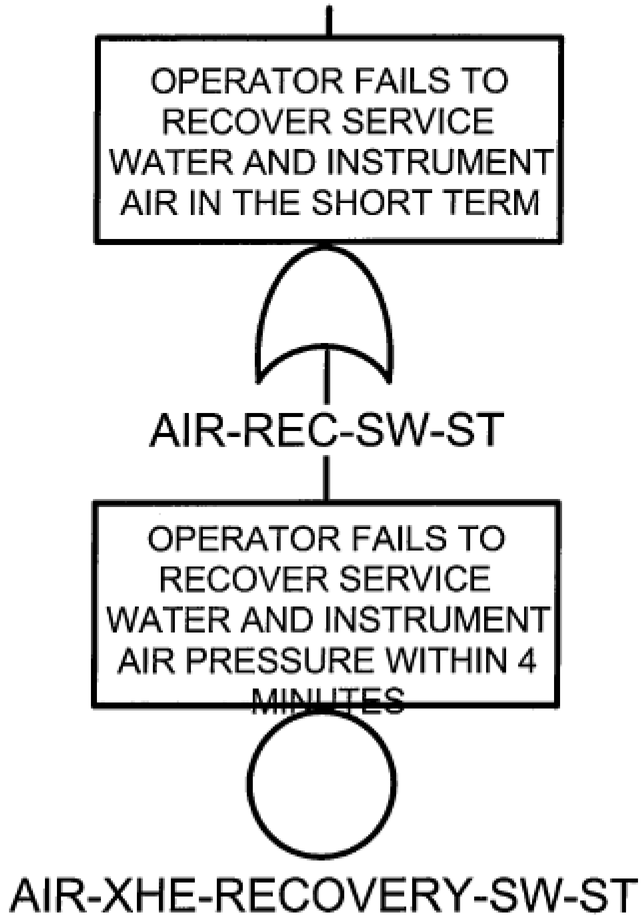


AIR-REC-ST - Point Beach 1 & 2 PWR Inst air not recovered - short term 2002/12/11 Page 12

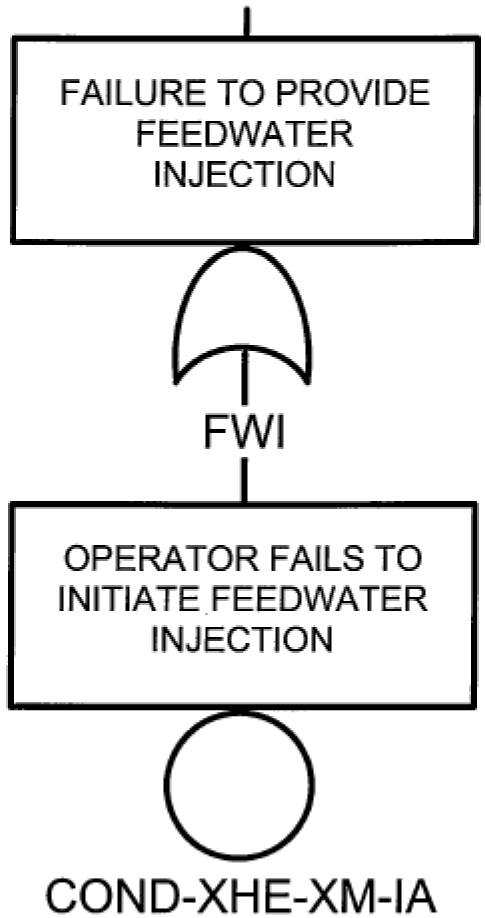


AIR-REC-SW-LT - Point Beach 1 & 2 PWR B fSW not recovered - long term 2003/01/16 Page 13


OPERATOR FAILS TO RECOVER SERVICE WATER AND INSTRUMENT AIR IN THE MEDIUM TERM
AIR-REC-SW-MT
OPERATOR FAILS TO RECOVER SERVICE WATER AND INSTRUMENT AIR PRESSURE WITHIN 30 MINUTES
AIR-XHE-RECOVERY-SW-MT
AIR-REC-SW-MT - Point Beach 1 & 2 PWR B SW not recovered - medium term
2003/01/16 Page 14



AIR-REC-SW-ST - Point Beach 1 & 2 PWR B SW not recovered - short term 2003/01/16 Page 15



FWI - Point Beach 1&2 PWR B feedwater injection 2002/12/23 Page 42

Appendix B
Basic Event Data Report

Refer to Section 4, Table 4-1 for list of modified or new basic events.

Appendices C-D

No changes made.

Appendix E
Alpha Factor Data Reports

Changes to alpha factors for common cause events are listed in Table E-1. Table E-1 shows the alpha factor names (Name), description (Name Description), uncertainty distribution parameter (Uncertainty Value), probability, and application source (Application).

Table E-1. Alpha Factor Values

Name	Name Description	Uncertainty Value	Probability	Application
SWS-MDP-FR-06A01	MOTOR DRIVEN PUMP ALPHA FACTOR 1 FOR 6 TRAINS	*4.12E+01*	*9.72E-1*	ESW MDP FR
SWS-MDP-FR-06A02	MOTOR DRIVEN PUMP ALPHA FACTOR 2 FOR 6 TRAINS	*1.51E+03*	*1.01E-02*	ESW MDP FR
SWS-MDP-FR-06A03	MOTOR DRIVEN PUMP ALPHA FACTOR 3 FOR 6 TRAINS	*1.51E+03*	*5.40E-03*	ESW MDP FR
SWS-MDP-FR-06A04	MOTOR DRIVEN PUMP ALPHA FACTOR 4 FOR 6 TRAINS	*1.52E+03*	*3.27E-03*	ESW MDP FR
SWS-MDP-FR-06A05	MOTOR DRIVEN PUMP ALPHA FACTOR 5 FOR 6 TRAINS	*1.52E+03*	*3.63E-03*	ESW MDP FR
SWS-MDP-FR-06A06	MOTOR DRIVEN PUMP ALPHA FACTOR 6 FOR 6 TRAINS	*1.52E+03*	*4.67E-03*	ESW MDP FR

Alpha Factor Data Reports

Application: ESW MDP FR

Alpha Factor Distribution Summary

	5th%	Mean	Median	95th%	MLE	a	b
1	0.9509522	0.9654111	0.9660325	0.9777435	0.9656281	4.79E+02	1.72E+01
2	2.23E-02	3.46E-02	3.40E-02	4.91E-02	3.44E-02	1.72E+01	4.79E+02
1	0.9562847	0.9676124	0.9680333	0.9775149	0.9680451	7.20E+02	2.41E+01
2	7.55E-03	1.38E-02	1.33E-02	2.14E-02	1.30E-02	1.02E+01	7.34E+02
3	1.13E-02	1.86E-02	1.82E-02	2.75E-02	1.89E-02	1.39E+01	7.31E+02
1	0.9596967	0.9692023	0.9695178	0.977623	0.9697911	9.66E+02	3.07E+01
2	8.06E-03	1.35E-02	1.31E-02	2.00E-02	1.26E-02	1.34E+01	9.83E+02
3	1.53E-03	4.28E-03	3.96E-03	8.15E-03	4.09E-03	4.27E+00	9.92E+02
4	7.75E-03	1.31E-02	1.27E-02	1.95E-02	1.35E-02	1.30E+01	9.83E+02
1	0.9628872	0.9710674	0.9713152	0.9783899	0.9717856	1.22E+03	3.64E+01
2	7.31E-03	1.19E-02	1.16E-02	1.73E-02	1.10E-02	1.49E+01	1.24E+03
3	2.30E-03	5.10E-03	4.84E-03	8.79E-03	4.86E-03	6.41E+00	1.25E+03
4	4.56E-04	1.99E-03	1.73E-03	4.40E-03	1.83E-03	2.50E+00	1.25E+03
5	5.86E-03	9.99E-03	9.73E-03	1.50E-02	1.06E-02	1.26E+01	1.24E+03
1	0.9657663	0.9729412	0.9731460	0.9794093	0.9737937	1.48E+03	4.12E+01
2	6.29E-03	1.01E-02	9.89E-03	1.47E-02	9.15E-03	1.54E+01	1.51E+03
3	2.72E-03	5.40E-03	5.18E-03	8.81E-03	5.16E-03	8.21E+00	1.51E+03
4	1.29E-03	3.27E-03	3.05E-03	5.98E-03	3.13E-03	4.97E+00	1.52E+03
5	1.51E-03	3.63E-03	3.41E-03	6.48E-03	3.81E-03	5.52E+00	1.52E+03
6	2.21E-03	4.67E-03	4.45E-03	7.86E-03	4.96E-03	7.11E+00	1.52E+03

Total Number of Independent Failure Events: 866.0

Total Number of Common Cause Failure Events: 71

Application Note:
Centrifugal ESW pumps. Includes the MDP, MOT, and PMP components FR.

Event Summary Statistics

Application ESW MDP FR | Component
Total Events 71 | Failure Mode
CCCG Size 6

CCCG SIZE [4](26), [2](16), [6](9), [5](10), [3](9), [12](1)

YEARS [1976](1), [1980](2), [1981](5), [1982](8), [1983](3), [1984](3), [1985](14), [1986](8), [1987](3), [1988](5), [1989](2), [1990](6), [1991](2), [1992](1), [1993](4),

[1994](4)

SHOCK TYPES NL(59), L(12)

COMPONENT TYPES PMP(64), MOT(7)

FAILURE MODES FR(71)

COUPLING FACTORS OMTC(24), HDSC(19), EI(14), OMTP(4), OOOP(5), EE(2), QIC(1), HQMM(1), OMTS(1), OOOS(1)

PROXIMATE CAUSES IC(39), DE(18), PA(4), HA(3), QP(3), IE(2), DC(1), DM(1), HT(1)

SYSTEMS ESW(71)

PLANTS SURRY 1(9), SURRY 2(9), FORT CALHOUN 1(4), PILGRIM 1(4), HATCH 1(3), INDIAN POINT 2(3), OCONEE 2(3), SALEM 2(3), BRUNSWICK 1(2), MILLSTONE 2(2), OCONEE 1(2), OCONEE 3(2), PALISADES(2), QUAD CITIES 1(2), SALEM 1(2), ARKANSAS 2(1), BROWNS FERRY 1(1), CALVERT CLIFFS 1(1), COOPER STATION(1), DRESDEN 3(1), FARLEY 1(1), FARLEY 2(1), GRAND GULF 1(1), HATCH 2(1), INDIAN POINT 3(1), KEWAUNEE(1), LASALLE 2(1), NINE MILE PT 1(1), PEACH BOTTOM 2(1), ROBINSON 2(1), SEABROOK 1(1), ST LUCIE 2(1), SUSQUEHANNA 1(1), TURKEY POINT 4(1)

OP ST (WHEN DET) O(61), D(10)

OP STATUS (MODE) BO(69), OP(2)
EVENT TYPE CCF(71)

COMP. GROUP PMP(71)

DETECTION MTHD. [BLANK](71)

DEGREE OF FAIL. P(55), C(12), A(4)

DEFENSE MECHAN. FSB(27), MAI(26), MON(15), PBR(2), DIV(1)

PLANT TYPES PWR(51), BWR(20)

VENDOR TYPES WE(33), GE(20), CE(11), BW(7)

Appendix G
Simplified Diagrams

No changes made.

Appendix H
HRA Worksheets

HRA Worksheets Index

Basic Event	Description	Page
COND-XHE-XM-IA	***Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure (new worksheet added)***	
COND-XHE-XM-IA1	***Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure (dependent) (new worksheet added)***	
COND-XHE-XM-IA1	***Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure (dependent) (new worksheet added)***	

SPAR Model Human Error Worksheet (Page 1 of 3)

Plant: Point Beach **Event Name:** COND-XHE-XM-IA

Task Error Description: Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure

Does this task contain a significant amount of diagnosis activity ? YES ___ NO X

If Yes, Use Table 1 below to evaluate the PSFs for the Diagnosis portion of the task before going to Table 2. If No, go directly to Table 2.

Table 1. Diagnosis worksheet.

PSFs	PSF Levels	Multiplier for Diagnosis	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	
	Barely adequate < 20 m	10	
	Nominal ≈ 30 m	1	
	Extra > 60 m	0.1	
	Expansive > 24 h	0.01	
2. Stress	Extreme	5	
	High	2	
	Nominal	1	
3. Complexity	Highly	5	
	Moderately	2	
	Nominal	1	
4. Experience/ Training	Low	10	
	Nominal	1	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
	Diagnostic/symptom oriented	0.5	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1	
8. Work Processes	Poor	2	
	Nominal	1	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

SPAR Model Human Error Worksheet (Page 2 of 3)

Table 2. Action worksheet.

PSFs	PSF Levels	Multiplier for Action	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	Recovery of instrument air occurs at the latest possible time to avert core uncovery.
	Time available ≈ time required	10✔	
	Nominal	1	
	Available > 50x time required	0.01	
2. Stress	Extreme	5	Establishing MFW is the option of last resort. Failure to accomplish task will result in core damage.
	High	2✔	
	Nominal	1	
3. Complexity	Highly	5	Establishing MFW is not difficult to perform. However, stabilizing the plant at saturated conditions with MFW and potential SI from overcooling will add to the complexity compared to MFW initiation immediately following loss of AFW.
	Moderately	2✔	
	Nominal	1	
4. Experience/ Training	Low	3	
	Nominal	1✔	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1✔	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1✔	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1✔	
8. Work Processes	Poor	2	
	Nominal	1✔	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

Table 3. Task failure probability without formal dependence worksheet.

Task Portion	Nom. Prob.	Time	Stress	Compl.	Exper./ Train.	Proced.	Ergon.	Fitness	Work Process	Prob.
Diag.	1.0E-2									na
Action	1.0E-3	10	2	2	1	1	1	1	1	4.0E-2
Total										4.0E-2

SPAR Model Human Error Worksheet (Page 3 of 3)

For all tasks, except the first task in the sequence, use the table and formulae below to calculate the Task Failure Probability With Formal Dependence.

Table 4. Dependency condition worksheet.

Condition Number	Crew (same or different)	Location (same or different)	Time (close in time or not close in time)	Cues (additional or not additional)	Dependency	Number of Human Action Failures Rule
1	s	s	c	–	complete	If this error is the 3rd **error in the sequence**, then the dependency is at least **moderate**. If this error is the 4th **error in the sequence**, then the dependency is at least **high**. This rule may be ignored only if there is compelling evidence for less dependence with the previous tasks.
2	s	s	nc	na	high	
3	s	s	nc	a	moderate	
4	s	d	c	–	high	
5	s	d	nc	na	moderate	
6	s	d	nc	a	low	
7	d	s	c	–	moderate	
8	d	s	nc	na	low	
9	d	s	nc	a	low	
10	d	d	c	–	moderate	
11	d	d	nc	na	low	
12	d	d	nc	a	low	
13					zero	

Using P = Task Failure Probability Without Formal Dependence (calculated on page 2):

For Complete Dependence the probability of failure = 1.0
For High Dependence the probability of failure = (1 + P)/2
For Moderate Dependence the probability of failure = (1 +6P)/7
For Low Dependence the probability of failure = (1 + 19P)/20
✔For Zero Dependence the probability of failure = P

Task Failure Probability With Formal Dependence = (1 + (____ * _____)) / _____ =_______

Additional Notes:
The following list is dependency correction factors that are required in order to take into account the dependency of the operator failures in a sequence cut set. By using value events, the original operator action names are not removed or renamed to take account for their dependencies.

SPAR Model Human Error Worksheet (Page 1 of 3)

Plant: Point Beach **Event Name:** COND-XHE-XM-IA1

Task Error Description: Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure (dependent)

Does this task contain a significant amount of diagnosis activity ? YES ___ NO X

If Yes, Use Table 1 below to evaluate the PSFs for the Diagnosis portion of the task before going to Table 2. If No, go directly to Table 2.

Table 1. Diagnosis worksheet.

PSFs	PSF Levels	Multiplier for Diagnosis	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	
	Barely adequate < 20 m	10	
	Nominal ≈ 30 m	1	
	Extra > 60 m	0.1	
	Expansive > 24 h	0.01	
2. Stress	Extreme	5	
	High	2	
	Nominal	1	
3. Complexity	Highly	5	
	Moderately	2	
	Nominal	1	
4. Experience/ Training	Low	10	
	Nominal	1	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
	Diagnostic/symptom oriented	0.5	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1	
8. Work Processes	Poor	2	
	Nominal	1	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

SPAR Model Human Error Worksheet (Page 2 of 3)

Table 2. Action worksheet.

PSFs	PSF Levels	Multiplier for Action	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	Recovery of instrument air occurs at the latest possible time to avert core uncovery.
	Time available = time required	10✔	
	Nominal	1	
	Available > 50x time required	0.01	
2. Stress	Extreme	5	Establishing MFW is the option of last resort. Failure to accomplish task will result in core damage.
	High	2✔	
	Nominal	1	
3. Complexity	Highly	5	Establishing MFW is not difficult to perform. However, stabilizing the plant at saturated conditions with MFW and potential SI from overcooling will add to the complexity compared to MFW initiation immediately following loss of AFW.
	Moderately	2✔	
	Nominal	1	
4. Experience/ Training	Low	3	
	Nominal	1✔	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1✔	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1✔	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1✔	
8. Work Processes	Poor	2	
	Nominal	1✔	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

Table 3. Task failure probability without formal dependence worksheet.

Task Portion	Nom. Prob.	Time	Stress	Compl.	Exper./ Train.	Proced.	Ergon.	Fitness	Work Process	Prob.
Diag.	1.0E-2									na
Action	1.0E-3	10	2	2	1	1	1	1	1	4.0E-2
Total										4.0E-2

SPAR Model Human Error Worksheet (Page 3 of 3)

For all tasks, except the first task in the sequence, use the table and formulae below to calculate the Task Failure Probability With Formal Dependence.

Table 4. Dependency condition worksheet.

Condition Number	Crew (same or different)	Location (same or different)	Time (close in time or not close in time)	Cues (additional or not additional)	Dependency	Number of Human Action Failures Rule
1	s	s	c	–	complete	If this error is the 3rd **error in the sequence**, then the dependency is at least **moderate**. If this error is the 4th **error in the sequence**, then the dependency is at least **high**. This rule may be ignored only if there is compelling evidence for less dependence with the previous tasks.
2	s	s	nc	na	high	
3	s	s	nc	a	moderate	
4	s	d	c	–	high	
5	s	d	nc	na	moderate	
6	s	d	nc	a	low	
7	d	s	c	–	moderate	
8	d	s	nc	na	low	
9	d	s	nc	a	low	
10	d	d	c	–	moderate	
11	d	d	nc	na	low	
12	d	d	nc	a	low	
13					zero	

Using P = Task Failure Probability Without Formal Dependence (calculated on page 2):

	For Complete Dependence the probability of failure	= 1.0
	For High Dependence the probability of failure	= (1 + P)/2
	For Moderate Dependence the probability of failure	= (1 +6P)/7
✔	For Low Dependence the probability of failure	= (1 + 19P)/20
	For Zero Dependence the probability of failure	= P

Task Failure Probability With Formal Dependence = (1 + (19 * 0.04)) / 20 = 0.088

Additional Notes:

The following list is dependency correction factors that are required in order to take into account the dependency of the operator failures in a sequence cut set. By using value events, the original operator action names are not removed or renamed to take account for their dependencies.

SPAR Model Human Error Worksheet (Page 1 of 3)

Plant: Point Beach **Event Name:** COND-XHE-XM-IA2

Task Error Description: Manual initiation of MFW using EOP-H.1 following recovery of instrument air pressure (dependent)

Does this task contain a significant amount of diagnosis activity ? YES ___ NO _X_

If Yes, Use Table 1 below to evaluate the PSFs for the Diagnosis portion of the task before going to Table 2.
If No, go directly to Table 2.

Table 1. Diagnosis worksheet.

PSFs	PSF Levels	Multiplier for Diagnosis	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	
	Barely adequate < 20 m	10	
	Nominal ≈ 30 m	1	
	Extra > 60 m	0.1	
	Expansive > 24 h	0.01	
2. Stress	Extreme	5	
	High	2	
	Nominal	1	
3. Complexity	Highly	5	
	Moderately	2	
	Nominal	1	
4. Experience/ Training	Low	10	
	Nominal	1	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
	Diagnostic/symptom oriented	0.5	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1	
8. Work Processes	Poor	2	
	Nominal	1	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

SPAR Model Human Error Worksheet (Page 2 of 3)

Table 2. Action worksheet.

PSFs	PSF Levels	Multiplier for Action	If non-nominal PSF levels are selected, please note specific reasons in this column
1. Available Time	Inadequate	1.0[a]	Recovery of instrument air occurs at the latest possible time to avert core uncovery.
	Time available = time required	10✔	
	Nominal	1	
	Available > 50x time required	0.01	
2. Stress	Extreme	5	Establishing MFW is the option of last resort. Failure to accomplish task will result in core damage.
	High	2✔	
	Nominal	1	
3. Complexity	Highly	5	Establishing MFW is not difficult to perform. However, stabilizing the plant at saturated conditions with MFW and potential SI from overcooling will add to the complexity compared to MFW initiation immediately following loss of AFW.
	Moderately	2✔	
	Nominal	1	
4. Experience/ Training	Low	3	
	Nominal	1✔	
	High	0.5	
5. Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1✔	
6. Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1✔	
	Good	0.5	
7. Fitness for Duty	Unfit	1.0[a]	
	Degraded Fitness	5	
	Nominal	1✔	
8. Work Processes	Poor	2	
	Nominal	1✔	
	Good	0.8	

a. Task failure probability is 1.0 regardless of other PSFs.

Table 3. Task failure probability without formal dependence worksheet.

Task Portion	Nom. Prob.	Time	Stress	Compl.	Exper./ Train.	Proced.	Ergon.	Fitness	Work Process	Prob.
Diag.	1.0E-2									na
Action	1.0E-3	10	2	2	1	1	1	1	1	4.0E-2
Total										4.0E-2

SPAR Model Human Error Worksheet (Page 3 of 3)

For all tasks, except the first task in the sequence, use the table and formulae below to calculate the Task Failure Probability With Formal Dependence.

Table 4. Dependency condition worksheet.

<table>
<tr><th>Condition Number</th><th>Crew (same or different)</th><th>Location (same or different)</th><th>Time (close in time or not close in time)</th><th>Cues (additional or not additional)</th><th>Dependency</th><th>Number of Human Action Failures Rule</th></tr>
<tr><td>1</td><td>s</td><td>s</td><td>c</td><td>–</td><td>complete</td><td rowspan="13">If this error is the 3rd error in the sequence, then the dependency is at least moderate.

If this error is the 4th error in the sequence, then the dependency is at least high.

This rule may be ignored only if there is compelling evidence for less dependence with the previous tasks.</td></tr>
<tr><td>2</td><td>s</td><td>s</td><td>nc</td><td>na</td><td>high</td></tr>
<tr><td>3</td><td>s</td><td>s</td><td>nc</td><td>a</td><td>moderate</td></tr>
<tr><td>4</td><td>s</td><td>d</td><td>c</td><td>–</td><td>high</td></tr>
<tr><td>5</td><td>s</td><td>d</td><td>nc</td><td>na</td><td>moderate</td></tr>
<tr><td>6</td><td>s</td><td>d</td><td>nc</td><td>a</td><td>low</td></tr>
<tr><td>7</td><td>d</td><td>s</td><td>c</td><td>–</td><td>moderate</td></tr>
<tr><td>8</td><td>d</td><td>s</td><td>nc</td><td>na</td><td>low</td></tr>
<tr><td>9</td><td>d</td><td>s</td><td>nc</td><td>a</td><td>low</td></tr>
<tr><td>10</td><td>d</td><td>d</td><td>c</td><td>–</td><td>moderate</td></tr>
<tr><td>11</td><td>d</td><td>d</td><td>nc</td><td>na</td><td>low</td></tr>
<tr><td>12</td><td>d</td><td>d</td><td>nc</td><td>a</td><td>low</td></tr>
<tr><td>13</td><td></td><td></td><td></td><td></td><td>zero</td></tr>
</table>

Using P = Task Failure Probability Without Formal Dependence (calculated on page 2):

	For Complete Dependence the probability of failure	= 1.0
	For High Dependence the probability of failure	= (1 + P)/2
✔	For Moderate Dependence the probability of failure	= (1 +6P)/7
	For Low Dependence the probability of failure	= (1 + 19P)/20
	For Zero Dependence the probability of failure	= P

Task Failure Probability With Formal Dependence = (1 + (6 * 0.04)) / 7 = 0.18

Additional Notes:
The following list is dependency correction factors that are required in order to take into account the dependency of the operator failures in a sequence cut set. By using value events, the original operator action names are not removed or renamed to take account for their dependencies.

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