B.11 LER Nos. 482/96-001, -002

Event Description: Reactor trip with a loss of train A of essential service water and the turbine-driven auxiliary feedwater pump

Date of Event: January 30, 1996

Plant: Wolf Creek

B.11.1 Event Summary

With the unit at 98% power, cold air caused ice to build on the circulating water system traveling screens. The decreased circulating water flow caused the pressure in the condenser to increase. The operators began a controlled shutdown; however, they were eventually forced to manually trip the unit from ~80% power in anticipation of a loss of vacuum in the condenser. Later, a frazil ice buildup on the trash racks forced operators to secure the A essential service water system (ESWS) pump and declare ESWS Train A out of service. Unrelated to the icing conditions, the turbine-driven auxiliary feedwater pump (TDAFWP) was declared out of service for 9 h when the inboard packing failed following the reactor trip. The unavailability of the ESWS pump and the TDAFWP affected the unit's response to a transient event; these unavailabilities would have affected the units' response to a transient induced loss of offsite power (LOOP) event.[1-3] The conditional core damage (CCDP) probability estimated for this event is 2.1×10^{-4}.

B.11.2 Event Description

On January 30, 1996, the plant was operating at 98% power at the beginning of a coastdown to a refueling outage. Ice began to block the circulating water traveling screens, which caused the pressure in the condenser to increase. Approximately 1 h later, operators began a controlled shutdown. Operators manually tripped the reactor when a loss of vacuum in the condenser became imminent. The circulating water pumps were then secured due to the low water level in the intake bay for the circulating water pumps. During the reactor trip, five control rods failed to insert fully. Indications showed that the five control rods stopped inserting between 9.53 and 28.58 cm (3.75 and 11.25 in.) from the bottom of the core. As required by the emergency operating procedures, operators began an emergency boration of the core. All five control rods drifted to the bottom of the core over the next 80 min.

Approximately 90 min after the reactor trip, the TDAFWP was reported to have an inboard shaft gland leak. The pump was secured and declared out of service, and, as required by Technical Specifications, the operators proceeded to take the plant to Mode 4. The TDAFWP was repaired in 9 h and returned to a functional status, though operational testing was still required. Complicating the situation, the auxiliary boiler tripped on at least two occasions. The auxiliary boiler provides heating to both the reactor water storage tank (RWST) and the condensate storage tank (CST) to prevent the water in the tanks from freezing. The lowest temperature that the water in the CST reached was 8.40°C (47.1°F) as recorded by the plant computer. The emergency backup water supply to the CST is the ESWS.

Four hours after the reactor trip, inadequate ESWS warming line flow—caused by original design errors and further reduced by an improper system alignment—resulted in a frazil ice buildup on the trash racks for the A and B ESWS pump intake bays. This condition eventually forced the A ESWS pump to be secured due to low water level in its intake bay. Operations personnel were confused about the actual cause of the fluctuating water level in the ESWS intake bays and, after the water level in the intake bay for train A recovered, the operators attempted to continue pump operation. However, the water level in the intake bay did not remain high enough to allow continuous pump operation. The level in the B ESWS pump intake bay was also quite low due to a frazil ice buildup. At one point, the water level in the intake bay for the B ESWS pump was only 12 m (4 ft) above the minimum water level required for sufficient net positive suction head (NPSH) and likely was moments away from a failure of ESWS Train B. Fortunately, the control room operators increased the heat load on ESWS Train B at this point and the intake water level for Train B began to recover. Ultimately, ESWS Train A was out of service for 37 h, while ESWS Train B pump and one of three one-half-capacity service water pumps remained in operation.

The operators were unable to enter Mode 4 within the time required by Technical Specifications because of the inefficient use of the cooldown procedure. The licensee eventually melted the ice blockage and restored ESWS Train A to normal standby alignment using sparging air to better mix the ESWS return water with the cold water in the intake bay. Cooling flow to Train A components was eventually reestablished by the nonsafety-related service water system (SWS). After evaluating the situation, the utility opted to enter the scheduled refueling outage early.

B.11.3 Additional Event-Related Information

During normal plant operation, the nonsafety-related SWS provides cooling to the ESWS loads and the turbine building loads, including the main feedwater (MFW) pump lube oil coolers. The SWS draws water from the same intake bays as the circulating water pumps. The nonsafety-related SWS consists of three half-capacity pumps and one low-flow startup pump. One SWS half-capacity pump remained in operation after the trip and throughout the event.

The ESWS is a two-train safety-related system started and isolated from the SWS following a safety injection (SI) signal or a LOOP event. The ESWS loads are normally split and include the component cooling water (CCW) heat exchangers and the coolers for the diesel generators. The ESWS loads also include the room coolers for the emergency core cooling system (ECCS) pumps, the charging pumps, the CCW pumps, the auxiliary feedwater (AFW) pumps, the control room, switchgear rooms, and containment. Additionally, the ESWS provides the backup water supply to the AFW system in case of a condensate storage tank failure. The CCW system normally operates in a split mode and cools the residual heat removal (RHR) system heat exchanger, the RHR seal cooler, the charging pump bearing oil cooler, the safety injection pump bearing oil cooler, and the reactor coolant pumps.

The ESWS was intended to be able to provide a flow of warm water (via a "warming line") to the ESWS intake bay. Design input assumption errors resulted in inadequate warming line flow and lower warming line temperature than intended. After the initial indication of an ice buildup in the circulating water bays, the operators manually started the A and B trains of the ESWS system. Operators failed to align the ESWS properly and to isolate it from the SWS when, for expediency, they were directed to align the ESWS from

memory. Although the expedient lineup was appropriate for the circumstances, a subsequent independent verification of the ESWS lineup with the procedure was not performed as required. The improper alignment resulted in further reductions of warming line flow to the ESWS intake bays for the pumps. This allowed frazil ice to build up on the trash racks and added to the confusion of the operators. While the Train A ESWS pump was declared inoperable due to the ice buildup, the water level in the Train B ESWS intake bay oscillated 1.8 to 4.6 m (6 to 15 ft) below normal as a result of the ice buildup on its trash rack. This situation was not fully communicated to the shift supervisor.

The icing on the ESWS trash racks was the result of a phenomenon known as frazil ice. According to the Augmented Inspection Report, the process starts when a body of water having a large surface area, such as the intake bay area, is subcooled by a loss of heat (as can happen on a very clear, windy, cold night). This condition, which existed at Wolf Creek, allowed tiny crystals of ice to form on the surface of the water. The heavy wind that existed on January 30 propelled the ice crystals below the intake surface. The water flow induced by the running ESWS pumps allowed the tiny ice crystals to readily accumulate on the metal surface of the trash racks.

B.11.4 Modeling Assumptions

This event is modeled as a transient event with the TDAFWP and one train of the ESWS unavailable. The five control rods that failed to insert fully eventually drifted to the bottom of the core. The model was not specifically altered to reflect the control rod problem. The five control rods were considered fully inserted for modeling purposes based on their proximity to the bottom of the core and because the operators commenced an emergency boration as directed by the emergency procedures.

Room cooling for all the ECCS equipment is provided by the ESWS. Procedures are in place to provide alternate room cooling in the event of the loss of the normal room coolers. Considering the inclement cold and windy weather contributing to the event, the loss of any room cooler was not considered a factor in considering a component failed for analysis purposes. The ESWS also removes heat from the CCW system via the CCW heat exchangers. The loss of the ability to remove heat via the CCW heat exchangers causes a loss of bearing oil or seal cooling to the SI pumps and the RHR pumps. However, in the injection mode, ECCS pump cooling was presumed to be adequate based on the flow of cold RWST water into the core. The ECCS injection system operator nonrecovery basic events were maintained at their nominal values to reflect increased operator attention to these systems during continued operation with degraded cooling. A "place-holder" basic event (HPI-WS-FAIL) was used to reflect a failure of the HPI system as a result of the failure of the ESWS where appropriate for clarity. The recirculation mode of RHR would be impacted by the potential loss of heat removal through the RHR heat exchangers. This potential loss is accounted for in the models by increasing the common-cause failure probability of the RHR heat exchangers (RHR-HTX-CF-ALL) to 0.1 based on the failure of ESWS Train A and the similar failure symptoms affecting Train B.

The SWS system provides cooling to the feedwater pump lube oil coolers. Because only one of three one-half-capacity SWS pumps remained in operation and the water supply from the intake bay was seriously threatened, the ability of the operators to recover main feedwater was not considered possible. The operator nonrecovery value (MFW-XHE-NOREC), therefore, is raised to 1.0 from the nominal value of 0.34 to reflect the inability of the operators to recover the main feedwater system if it were to fail. [The probability of the

main feedwater system tripping (MFW-SYS-TRIP) is not changed because the lube oil coolers for the MFW pumps are cooled by the SWS, which still had one half-capacity pump running.]

Two basic events are added to the Wolf Creek model to account for both trains of the ESWS. Both events (EWS-MDP-FC-1A and EWS-MDP-FC-1B) are assigned a nominal failure probability of 1.78×10^{-3} based on data from the Wolf Creek individual plant examination (IPE). Basic event ESW-MDP-FC-1A is set to "TRUE" (i.e., failed) based on the unavailability of the ESWS Train A pump due to the inability to maintain the water level in the ESWS Train A pump intake bay. Because ESWS provides cooling to the emergency diesel generators (EDG), the ESWS Train A failure causes the model to recognize the A EDG as failed. The failure probability for ESW-MDP-FC-1B is not adjusted from the nominal failure probability given in the IPE because there is no means to indicate that an otherwise operable component is in jeopardy of imminent failure from an external cause. However, two sensitivity studies explore the impact of the degraded operating condition of the B ESWS pump: (1) the failure probability of ESWS Train B is increased by a factor of ten to 1.78×10^{-2} and (2) the failure probability is changed to 0.1.

A common cause failure event is also added for the ESWS (EWS-MDP-CF-ALL). Based on the failure of the ESWS Train A pump and the operating condition of the ESWS Train B pump, the basic event probability is increased to 0.15 (based on the beta factor for the RHR pump). Finally, an event is added to account for the operator's failure to recover the ESWS if it should fail (EWS-XHE-NOREC). Because of the extreme operating conditions, however, this probability is set to TRUE (i.e., no recovery).

The TDAFWP failure (AFW-TDP-FC-1C) is also set to TRUE. The pump may have operated for the entire 24-h mission time with the inboard packing failure; however, predicting how long the pump could have continued to provide feedwater flow to the steam generators is difficult. Therefore, considering the pump was physically disabled for repairs after the operators declared the pump to be out of service, it is appropriate to consider the pump to be failed. Based on the initial operability of the TDAFW pump (97 min) and the subsequent unavailability of the pump (537 min), a combined operator nonrecovery value for a station blackout (SBO) (AFW-XHE-NOREC-EP) is projected as follows:

$$\text{P(Operator fails to recover AFW)} = [97(0.34) + 537(1.0)] \div 634 = 0.899,$$

where the nominal value of AFW-XHE-NOREC-EP is 0.34; while the pump is undergoing repairs, the value can be assumed to be 1.0 for an SBO. If the entire 24-h mission time is considered, AFW-XHE-NOREC-EP can be projected to be 0.586 by a similar calculation. A sensitivity study is explored for this value. For the case with no LOOP, the operator nonrecovery value (AFW-XHE-NOREC) is left at its nominal value of 0.26. This method is appropriate considering both motor-driven AFW pumps were available and the TDAFWP was available for at least the initial 97 min of the event.

The emergency power system is directly affected by the ESWS support system. As previously noted, the event circumstances made it appear appropriate to adjust the ESWS operator nonrecovery value (EWS-XHE-NOREC) to 1.0. This adjustment seems to dictate that the operator nonrecovery value for the emergency power system (EPS-XHE-NOREC) also be adjusted to 1.0. However, not all possible EDG failures involve a loss of ESWS support. Therefore, EPS-XHE-NOREC is left at the nominal value of 0.80. A sensitivity case was reviewed adjusting the emergency power operator nonrecovery value to 1.0.

The *Reactor Safety Study*[4] reports the probability of a LOOP being induced by a LOCA (transient) as 1.0×10^{-3}. Additionally, a search of the Sequence Coding and Search System for transient-induced LOOPs over a 10-year period between 1984 and 1993 revealed five transient-induced LOOPs out of 3,985 trips. This calculation yields a rate of 1.25×10^{-3} per transient, which tends to substantiate the WASH-1400 value. The grid-based LOOP probability of short-term and long-term offsite power recovery and the probability of a reactor coolant pump (RCP) seal LOCA following a postulated station blackout were developed based on data distributions contained in *Evaluation of Station Blackout Accidents at Nuclear Power Plants*.[7] The RCP seal LOCA models were developed as part of the NUREG-1150 PRA efforts. Both are described in *Revised LOOP Recovery and PWR Seal LOCA Models*.[8] The initiating cause of a LOOP is assumed to be a grid-related disturbance caused by the plant trip. Because of the severe cold and wind, it was further assumed that if a LOOP were to occur because of the transient, offsite power would not be restored within 30 min. The possibility of a LOOP is added to the transient initiating event tree following a reactor trip (OFFSITE Fault tree). This possibility leads to a transfer to an event tree similar to the LOOP initiating event tree (TRANLOOP).

B.11.5 Analysis Results

The estimated CCDP associated with this event is 2.1×10^{-4}. The dominant core damage sequence, highlighted as sequence number 21-39 on the event tree in Figs. B.11.1 and B.11.2, contributes approximately 66% to the CCDP estimate. This event involves

- a successful reactor trip,
- subsequent LOOP,
- both trains of emergency power fail, and
- AFW fails to provide sufficient flow.

This sequence is driven by the loss of the TDAFWP and the common-cause failure of the ESWS pumps. In an actual station blackout, the operators would likely have continued to operate the TDAFWP with the gland leak until it failed, while working in parallel to restore emergency power. The combined transient-induced LOOP sequences contribute 89% of the total estimated CCDP.

The most significant transient sequence that does not involve a LOOP contributes approximately 6% to the CCDP estimate. This transient sequence (sequence number 20 on the event tree in Fig. B.11.1) involves

- a successful reactor trip,
- failure of AFW,
- failure of MFW, and
- failure of feed-and-bleed.

This transient sequence is driven by the common-cause failure of the ESWS pumps and by a failure of the operator to establish make-up water to the CST (AFW-XHE-XA-MW).

A sensitivity study assumes the failure rate of ESWS Train B is increased by a factor of ten to 1.78×10^{-2}. This was an attempt to reflect the increased (time dependent) possibility of the remaining ESWS pump failing

when intake water levels decreased to 1.2 m (4 ft) above the minimum required to maintain NPSH, and poor communications did not allow this information to be properly relayed to shift management. The ESWS pump common-cause failure factor due to the frazil ice buildup was left at the previously increased value (0.15). The estimated CCDP associated with this case increases to 2.3×10^{-4}. The dominant transient sequence remains the same, and the relative contribution that transient-induced LOOP sequences add to the estimated CCDP remain about the same.

Further sensitivity studies indicate the increasing likelihood of core damage as the reliability of the second pump is decreased. If the probability of ESWS Train B pump failure is assumed to be 0.1, the estimated CCDP associated with this case increases to 3.4×10^{-4}. Further, if the probability of ESWS Train B pump failure is assumed to be 0.5, the estimated CCDP associated with this case increases to 6.9×10^{-4}. Both of these cases are intended to capture the time sensitivity of the situation regarding the possible imminent failure of the B ESWS pump. Again, the ESWS pump common-cause failure factor due to the frazil ice buildup was left at the previously increased value (0.15). This sensitivity case is calculated to reflect the potential for failure of the ESWS Train B pump. The dominant transient sequence remains the same as the base case (i.e., sequence 21-39) for both cases.

When the operator nonrecovery value for emergency power (EWS-XHE-NOREC) is assumed to be 1.0, the CCDP increases to 2.6×10^{-4} compared with the base case value of 2.1×10^{-4}. If the operator nonrecovery value for AFW during a station blackout (SBO) (AFW-XHE-NOREC-EP) is reduced to 0.586 based on the TDAFWP availability during the 24-h mission period, then the CCDP is reduced to 1.6×10^{-4}.

Definitions and probabilities for selected basic events are shown in Table B.11.1. The conditional probabilities associated with the highest probability sequences are shown in Table B.11.2. Table B.11.3 lists the sequence logic associated with the sequences listed in Table B.11.2. Table B.11.4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table B.11.5.

B.11.6 References

1. LER 482/96-001, Rev. 0, "Loss of Circulating Water Due to Icing on Traveling Screens Causes Reactor Trip," February 28, 1996.

2. LER 482/96-002, Rev. 0, "Loss of A Train Essential Service Water Due to Icing on the Trash Racks," February 29, 1996.

3. NRC Inspection Report 50-482/96-05, March 7, 1996.

4. WASH-1400, NUREG-75/014, Table II 5-3, *Reactor Safety Study*, October 1975.

5. *Sequence Coding and Search System for Licensee Event Reports*, NUREG/CR-3905 LD, August 1984.

6. *Wolf Creek Generating Station Individual Plant Examination Summary Report*, September 1992.

7. P. W. Baranowsky, *Evaluation of Station Blackout at Nuclear Power Plants*, NUREG-1032, U.S. Nuclear Regulatory Commission, June, 1988.

8. *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-9/11, August 1989.

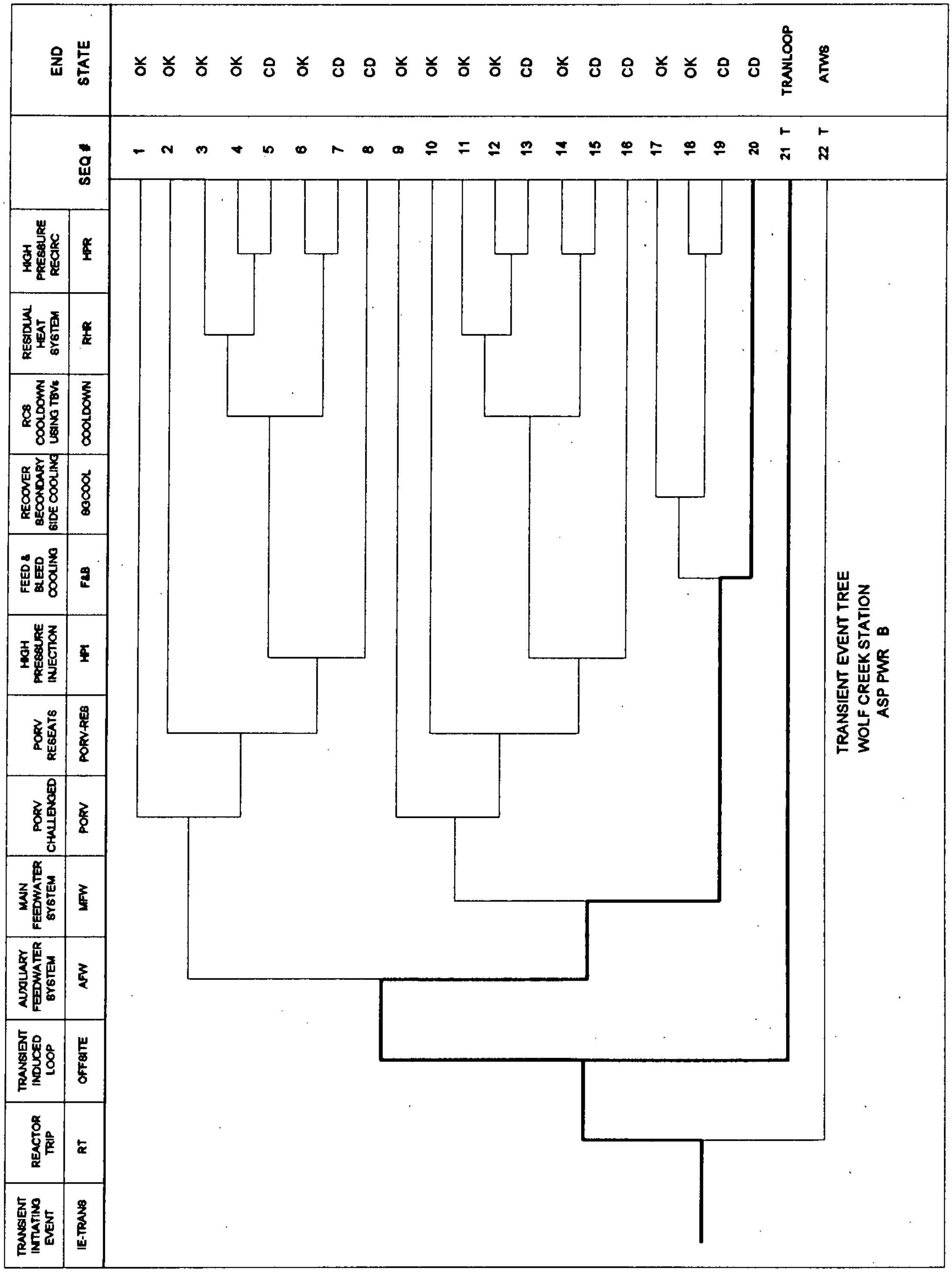

Fig. B.11.1. Dominant core damage sequence given a LOOP for LER Nos. 482/96-001, -002.

Table B.11.1. Definitions and Probabilities for Selected Basic Events for LER Nos. 482/96-001, -002

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–LOOP	6.9 E-006	0.0 E+000	IGNORE	Yes
IE-SGTR	Initiating Event–Steam Generator Tube Rupture	1.6 E-006	0.0 E+000	IGNORE	Yes
IE-SLOCA	Initiating Event–Small Loss-of-Coolant Accident	1.0 E-006	0.0 E+000	IGNORE	Yes
IE-TRANS	Initiating Event–Transient	5.3 E-004	1.0 E+000	TRUE	Yes
AFW-MDP-CF-AB	Common-Cause Failure of All Motor-Driven AFW Pumps	2.1 E-004	2.1 E-004		No
AFW-PMP-CF-ALL	Common-Cause Failure of AFW Pumps	2.8 E-004	2.8 E-004		No
AFW-TDP-FC-1C	AFW Turbine-Driven Pump Fails	3.2 E-002	1.0 E+000	TRUE	Yes
AFW-TNK-FC-CST1	Failure of the CST	4.1 E-005	4.1 E-005		No
AFW-XHE-NOREC	Operator Fails to Recover AFW System	2.6 E-001	2.6 E-001		No
AFW-XHE-NOREC-EP	Operator Fails to Recover AFW During a SBO	3.4 E-001	9.0 E-001		Yes
AFW-XHE-XA-MW	Operator Fails to Initiate Makeup Water	1.0 E-003	1.0 E-003		No
CVC-MDP-FC-1A	Charging Train A Fails	3.9 E-003	3.9 E-003		No
CVC-MDP-FC-1B	Charging Train B Fails	6.8 E-003	6.8 E-003		No
EPS-DGN-FC-1B	EDG B Fails	4.2 E-002	4.2 E-002		No
EPS-XHE-NOREC	Operator Fails to Recover Emergency Power	8.0 E-001	8.0 E-001		No
EWS-MDP-CF-ALL	Common-Cause Failure of ESWS Motor-Driven Pumps	2.6 E-004	1.5 E-001		Yes
EWS-MDP-FC-1A	Failure of ESWS Train A	1.7 E-003	1.0 E+000	TRUE	Yes
EWS-MDP-FC-1B	Failure of ESWS Train B	1.7 E-003	1.7 E-003		No
EWS-XHE-NOREC	Operator Fails to Recover ESWS	8.4 E-001	1.0 E+000	TRUE	Yes

Table B.11.1. Definitions and Probabilities for Selected Basic Events for LER Nos. 482/96-001, -002 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPI-EWS-FAIL	High Pressure Injection (HPI) System Fails in Injection Mode Given ESWS is Failed	1.0 E+000	1.0 E+000	TRUE	Yes
HPI-XHE-NOREC	Operator Fails to Recover the HPI System	8.4 E-001	8.4 E-001		No
HPI-XHE-XM-FB	Operator Fails to Initiate Feed-and-Bleed Cooling	1.0 E-002	1.0 E-002		No
HPR-XHE-NOREC	Operator Fails to Recover the High Pressure Recirculation (HPR) System	1.0 E+000	1.0 E+000		No
LOOP	Transient-Induced LOOP	1.0 E-003	1.0 E-003		No
MFW-SYS-TRIP	MFW System Trips	2.0 E-001	2.0 E-001		No
MFW-XHE-NOREC	Operator Fails to Recover MFW	3.4 E-001	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 h	2.2 E-001	1.1 E-001		Yes
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 h	6.7 E-002	3.6 E-004		Yes
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Batteries Deplete	2.5 E-002	3.6 E-003		Yes
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power (Seal LOCA)	5.8 E-001	4.4 E-001		Yes
PPR-MOV-OO-BLK1	PORV 1 Block Valve Fails to Close	3.0 E-003	3.0 E-003		No
PPR-MOV-OO-BLK2	PORV 2 Block Valve Fails to Close	3.0 E-003	3.0 E-003		No
PPR-SRV-CC-1	PORV 1 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CC-2	PORV 2 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CO-SBO	PORVs Open During SBO	1.0 E+000	1.0 E+000		No

Table B.11.1. Definitions and Probabilities for Selected Basic Events for LER Nos. 482/96-001, -002 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
PPR-SRV-OO-2	PORV 2 Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No
PPR-XHE-NOREC	Operator Fails to Close Block Valve	1.1 E-002	1.1 E-002		No
RCS-MDP-LK-SEALS	RCP Seals Fail Without Cooling and Injection	2.7 E-002	2.1 E-001		Yes
RHR-HTX-CF-ALL	Common-Cause Failure of RHR Heat Exchangers	1.4 E-005	1.0 E-001		Yes
RHR-MDP-FC-1A	RHR Train A Fails	3.9 E-003	3.9 E-003		No
RHR-XHE-NOREC	Operator Fails to Recover the RHR System	1.0 E+000	1.0 E+000		No

Table B.11.2. Sequence Conditional Probabilities for LER Nos. 482/96-001, -002

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Percent contribution
TRANS	21-39	1.4 E-004	65.9
TRANS	21-36	2.1 E-005	10.0
TRANS	21-37	1.4 E-005	6.7
TRANS	20	1.2 E-005	5.6
TRANS	21-38	9.3 E-006	4.4
TRANS	08	4.2 E-006	2.0
TRANS	21-33	4.1 E-006	1.9
TRANS	05	3.5 E-006	1.6
Total (all sequences)		**2.1 E-004**	

Table B.11.3. Sequence Logic for Dominant Sequences for LER Nos. 482/96-001, -002

Event tree name	Sequence name	Logic
TRANS	21-39	/RT-L, OFFSITE, EP, AFW-L-EP
TRANS	21-36	/RT-L, OFFSITE, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, /OP-SL, HPI
TRANS	21-37	/RT-L, OFFSITE, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, OP-SL
TRANS	20	/RT, /OFFSITE, AFW, MFW, F&B
TRANS	21-38	/RT-L, OFFSITE, EP, /AFW-L-EP, PORV-SBO, PORV-EP
TRANS	08	/RT, /OFFSITE, /AFW, PORV, PORV-RES, HPI
TRANS	21-33	/RT-L, OFFSITE, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, /OP-SL, /HPI, /COOLDOWN, RHR, HPR
TRANS	05	/RT, /OFFSITE, /AFW, PORV, PORV-RES, /HPI, /COOLDOWN, RHR, HPR

Table B.11.4. System Names for LER Nos. 482/96-001, -002

System name	Logic
AFW	No or Insufficient AFW
AFW-L-EP	No or Insufficient AFW Flow During Station Blackout
COOLDOWN	Reactor Coolant System Cooldown to RHR Pressure Using Turbine Bypass Valves, etc.
EP	Failure of Both Trains of Emergency Power
F&B	Failure of Feed-and-Bleed Cooling
HPI	No or Insufficient Flow from the HPI System
HPR	No or Insufficient Flow from the HPR System
MFW	Failure of the MFW System
OFFSITE	Transient-Induced LOOP
OP-SL	Operator Fails to Recover Offsite Power (Seal LOCA)
PORV	PORVs Open During Transient
PORV-EP	PORVs Fail to Reclose (No Electric Power)
PORV-RES	PORVs Fail to Reseat
PORV-SBO	PORVs Open During SBO
RHR	No or Insufficient Flow from the RHR System
RT	Reactor Fails to Trip During Transient
SEALLOCA	RCP Seals Fail During LOOP

Table B.11.5. Conditional Cut Sets for Higher Probability Sequences for LER Nos. 482/96-001, -002

Cut set number	Percent contribution	Conditional probability	Cut sets[a]
TRANS Sequence 21-39		1.4 E-004	
1	76.8	1.1 E-004	LOOP, EWS-MDP-CF-ALL, EWS-XHE-NOREC, EPS-XHE-NOREC, AFW-TDP-FC-1C, AFW-XHE-NOREC-EP
2	21.5	3.0 E-005	LOOP, EPS-DGN-FC-1B, EWS-MDP-FC-1A, EPS-XHE-NOREC, AFW-TDP-FC-1C, AFW-XHE-NOREC-EP
TRANS Sequence 21-36		2.1 E-005	
1	98.8	2.1 E-005	LOOP, EWS-MDP-CF-ALL, EWS-XHE-NOREC, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, HPI-EWS-FAIL, HPI-XHE-NOREC
2	1.2	2.5 E-007	LOOP, EPS-MDP-FC-1A, EWS-MDP-FC-1B, EWS-XHE-NOREC, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, HPI-EWS-FAIL, HPI-XHE-NOREC
TRANS Sequence 21-37		1.4 E-005	
1	76.8	1.1 E-005	LOOP, EWS-MDP-CF-ALL, EWS-XHE-NOREC, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
2	21.5	3.1 E-006	LOOP, EPS-DGN-FC-1B, EWS-MDP-FC-1A, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
TRANS Sequence 20		1.2 E-005	
1	54.2	6.6 E-006	AFW-XHE-XA-MW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
2	15.2	1.8 E-006	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
3	11.4	1.4 E-006	AFW-MDP-CF-AB, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
4	4.3	5.2 E-007	AFW-XHE-XA-MW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
5	2.7	3.3 E-007	AFW-XHE-XA-MW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
6	2.7	3.3 E-007	AFW-XHE-XA-MW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1

Table B.11.5. Conditional Cut Sets for Higher Probability Sequences for LER Nos. 482/96-001, -002 (Continued)

Cut set number	Percent contribution	Conditional probability	Cut sets[a]
7	2.2	2.7 E-007	AFW-TNK-FC-CSTI, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
8	1.2	1.5 E-007	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
TRANS Sequence 21-38		9.3 E-006	
1	38.4	3.6 E-006	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2
2	38.4	3.6 E-006	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
3	10.7	1.0 E-006	LOOP, EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2
4	10.7	1.0 E-006	LOOP, EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
TRANS Sequence 08		4.2 E-006	
1	38.8	1.7 E-006	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
2	38.8	1.7 E-006	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
3	10.6	4.5 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-MOV-OO-BLK2, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
4	10.6	4.5 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-MOV-OO-BLK1, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-EWS-FAIL, HPI-XHE-NOREC
TRANS Sequence 21-33		4.1 E-006	
1	63.1	2.5 E-006	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-HTX-CF-ALL, RHR-XHE-NOREC, HPR-XHE-NOREC
2	17.6	7.1 E-007	LOOP, EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-HTX-CF-ALL, RHR-XHE-NOREC, HPR-XHE-NOREC

Table B.11.5. Conditional Cut Sets for Higher Probability Sequences for LER Nos. 482/96-001, -002 (Continued)

Cut set number	Percent contribution	Conditional probability	Cut sets[a]
3	7.4	3.0 E-007	LOOP, EWS-MDP-FC-1A, EWS-MDP-FC-1B, EWS-XHE-NOREC, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, HPR-XHE-NOREC
4	4.2	1.7 E-007	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, CVC-MDP-FC-1B, HPR-XHE-NOREC
5	2.4	9.8 E-008	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-MDP-FC-1A, RHR-XHE-NOREC, HPR-XHE-NOREC
6	2.4	9.8 E-008	LOOP, EWS-MDP-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, CVC-MDP-FC-1A HPR-XHE-NOREC
TRANS Sequence 05		3.5 E-006	
1	37.1	1.3 E-006	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-XHE-NOREC, RHR-XHE-NOREC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
2	37.1	1.3 E-006	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-XHE-NOREC, RHR-XHE-NOREC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
3	10.1	3.6 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, RHR-XHE-NOREC, PPR-MOV-OO-BLK2, RHR-HTX-CF-ALL, HPR-XHE-NOREC
4	10.1	3.6 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, RHR-XHE-NOREC, PPR-MOV-OO-BLK1, RHR-HTX-CF-ALL, HPR-XHE-NOREC
Total (all sequences)		**2.1 E-004**	

[a]Basic events AFW-TDP-FC-1C, EWS-MDP-FC-1A, EWS-XHE-NOREC, and MFW-XHE-NOREC are all type TRUE events which are not normally included in the output of fault tree reduction programs but have been added to aid in understanding the sequences to potential core damage associated with the event.