50-483

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 16, 1996

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF THE ICING EVENT AT WOLF CREEK

Dear Mr. Carns:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at Wolf Creek on January 30, 1996 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 482/96-001. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORNL). The results of this preliminary analysis indicate that this event may be a precursor for 1996. In assessing operational events, an effort was made to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures, or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of the review process is to provide as realistic an analysis of the significance of the event as possible.

In order for us to incorporate your comments, perform any required reanalysis, and prepare the final report of our analysis of this event in a timely manner, you are requested to complete your review and to provide any comments within 30 days of receipt of this letter. We have streamlined the ASP Program with the objective of significantly improving the time after an event in which the final precursor analysis of the event is made publicly available. As soon as our final analysis of the event has been completed, we will provide for your information the final precursor analysis of the event and the resolution of your comments. In previous years, licensees have had to wait until publication of the Annual Precursor Report (in some cases, up to 23 months after an event) for the final precursor analysis of an event and the resolution of their comments.

We have also enclosed several items to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, identifies the criteria which we will apply to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event, and describes the specific

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information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 482/96-001, which documented the event.

Please contact me at (301) 415-3063 if you have any questions regarding this request. This request is covered by the existing OMB clearance number (3150-0104) for NRC staff followup review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

Original Signed By

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Accident Sequence Precursor Analysis
2. Review Guidance
3. LER 482/96-001

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LER No. 482/96-001

Event Description: Reactor Trip with a loss of Train A of the Essential Service Water and the Turbine-Driven Auxiliary Feedwater Pump

Date of Event: January 30, 1996

Plant: Wolf Creek

Event Summary

With the unit at 98% power, ice began to block the circulating water system 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 approximately 80% power in anticipation of a loss of vacuum in the condenser. Later, the ice buildup also forced the 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 following the discovery of a packing leak after the reactor trip. The unavailability of the ESWS pump and the TDAFWP affected the units' response to a transient event; these unavailabilities would have affected the units' response to a loss of offsite power (LOOP) event. The conditional core damage probability estimated for this event is 4.8×10^{-5}.

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 fully insert. Indications showed that the five control rods stopped inserting between 3.75 and 11.25 inches 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 hour.

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) in order to prevent the water in the tanks from freezing. The lowest temperature that the water reached in the CST was not reported. The emergency backup water supply to the CST is the ESWS.

Four hours after the reactor trip, ice buildup on the trash rack in front of the traveling screen for the A ESWS pump intake bay forced that pump to be secured due to low water level in its intake bay. 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 ice buildup. At one point, the water level in the intake bay for the B ESWS pump was only 4 feet above the minimum water level required for sufficient net positive suction head (NPSH). Ultimately, the ESWS Train A was out of service for 37 hours, 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 due to the equipment problems, the cold weather, and the inefficient use of the cooldown procedure. The licensee used portable heaters and sparging air to eventually break up the ice blockage and restore flow in ESWS Train A. After evaluating the situation, the utility opted to enter the scheduled refueling outage early.

Additional Event-Related Information

During normal plant operation, the service water system (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.

The ESWS is a two-train safety related system that is 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 the event 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 also has the capability to provide a flow of warm water (via a "warming line") to the ESWS intake bay. After the initial indication of ice buildup in the circulating water bays, the operators manually started the B train of the ESWS system. Operators failed to properly align the ESWS and to isolate it from the SWS when, for expediency, they were directed to align the ESWS from memory. The improper alignment resulted in severely restricted warming line flow to the ESWS intake bays for the pumps. This allowed ice to build up similar to that in the circulating water pump intake bays when the circulating water pumps were operating. After the Train A ESWS pump became inoperable due to the buildup of ice, the water level in the Train B ESWS intake bay oscillated 6 to 15 feet below normal due to the ice buildup on its trash rack. This situation was not fully communicated to the shift supervisor.

The icing in the intake area was the result of a phenomenon known as frazil ice. 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 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 circulating water pumps allowed the tiny ice crystals to readily accumulate on the metal surface of the trash racks. Because the ice rapidly expanded, flow through the trash rack was blocked without a gradual increase in the differential pressure indication across the trash rack. Thus, little advance notice was given to the reactor operator that the frazil ice condition existed and that the intake screens were becoming blocked.

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 fully insert eventually drifted to the bottom. The model was not specifically altered to reflect the control rod problem. The five control rods were considered to be 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. This 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 recirculation mode of RHR would be impacted by the potential loss of heat removal through the RHR heat exchangers. This was accounted for in the models by increasing the common cause failure probability of the RHR heat exchangers (RHR-HTX-CF-ALL) to the beta value of 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. Since 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 considered very limited. The operator nonrecovery value (MFW-XHE-NOREC) was 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) was 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 were 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) were 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 was 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. Since ESWS provides cooling to the emergency diesel generators (EDG), the ESWS Train A failure caused the model to recognize the A EDG as failed. The failure probability for ESW-MDP-FC-1B was not adjusted from the nominal failure probability given in the IPE because there is no means to indicate that a component that is in jeopardy of imminent failure. However, two sensitivity studies were performed to explore the impact of the degraded operating condition of the B ESWS pump: first, the failure probability of ESWS Train B was increased by a factor of ten to 1.78×10^{-2}; second, the failure probability was changed to 0.1.

A common cause failure event was 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 event probability was increased to the system beta factor of 0.15 (based on the beta factor for the RHR pump). Finally, an event was added to account for the operator failure to recover the ESWS if it should fail (EWS-XHE-NOREC). Due to the extreme operating conditions, however, this probability was set to TRUE (i.e., no recovery).

The TDAFWP failure (AFW-TDP-FC-1C) was also set to TRUE. The pump may have operated for the entire mission time with the packing gland leak; however, it is difficult to predict how long the pump could have continued to provide feedwater flow to the steam generators. 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.

The *Reactor Safety Study* reports the probability of a LOOP being induced by a LOCA (transient) as 1.0×10^{-3} (*Reactor Safety Study*, WASH-1400, NUREG-75/014, Table II 5-3). Additionally, a search of the Sequence Coding and Search System for transient induced LOOPs over a ten-year period between 1984 and 1993 revealed 5 transient-induced LOOPs out of 3985 trips. This 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 NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*. The RCP seal LOCA models were developed as part of the NUREG-1150 PRA efforts. Both of these are described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. It is assumed that the initiating cause of a LOOP would be a grid-related disturbance caused by the plant trip. Due to the severe cold and wind, it was further assumed that if a LOOP were to occur as a result of the transient, offsite power would not be restored within 30 min.

Analysis Results

The estimated conditional core damage probability (CCDP) associated with this event is 4.8×10^{-5}. The dominant core damage sequence, highlighted as sequence number 39 on the event tree in Fig. 1, contributes approximately 25% to the CCDP estimate. This event involves:

- given a loss of offsite power, the reactor successfully trips,
- both trains of emergency power fail, and
- AFW fails to provide sufficient flow.

This sequence is driven by the loss of the TDAFWP and a nominal failure of the B EDG. In an actual LOOP, 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 LOOP sequences contribute 53% of the total estimated CCDP. A transient sequence, highlighted as sequence number 20 on the event tree in Fig. 2, also contributes approximately 25% to the CCDP estimate. The transient sequence 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.

A sensitivity study was performed assuming the failure rate of ESWS Train B is increased by a factor of ten to 1.78×10^{-2}. The estimated CCDP associated with this case increases to 6.9×10^{-5}. The dominant LOOP sequence remained the same, but overall, the combined LOOP sequences become more dominant (now contributing 63% of the total CCDP).

A second sensitivity study was performed assuming the ESWS Train B pump failure probability to be 0.1. This sensitivity case was performed to reflect the potential for failure of the ESWS Train B pump when intake water level decreased to four feet above the minimum required to maintain NPSH and poor communications did not allow this information to be properly relayed to shift management. The estimated ccdp associated with this case increases to 1.7×10^{-4}. The dominant LOOP sequence remains the same as the base case (i.e., sequence 39), however, the combined LOOP sequences now contribute 78% of the total CCDP.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table 5.

Acronyms

AFW	Auxiliary Feedwater
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CST	Condensate Storage Tank
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESWS	Emergency Service Water System
HPI	High Pressure Injection
HPR	High Pressure Recirculation
IPE	Individual Plant Examination
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
NPSH	Net Positive Suction Head
MFW	Main Feedwater
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWST	Reactor Water Storage Tank
SBO	Station Blackout
SI	Safety Injection
SWS	Service Water System
TDAFWP	Turbine-Driven Auxiliary Feedwater Pump

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. NRC Inspection Report 50-482/96-05, March 7, 1996.
3. WASH-1400, NUREG-75/014, Table II 5-3, *Reactor Safety Study*, October, 1975.
4. Sequence Coding and Search System.
5. Wolf Creek Generating Station Individual Plant Examination Summary Report, September 1992.

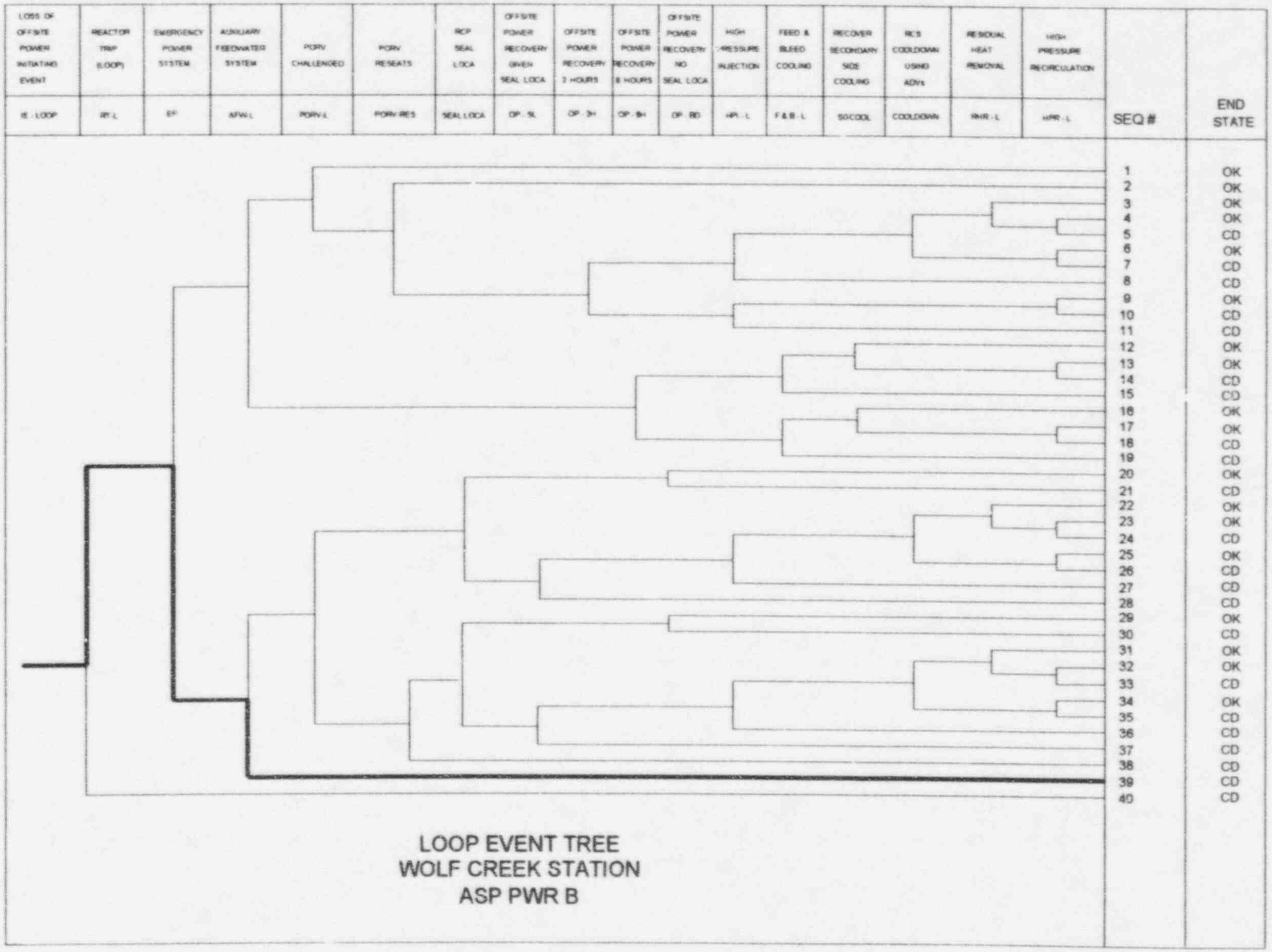

Fig. 1 Dominant core damage sequence given a LOOP for LER No. 482/96-001.

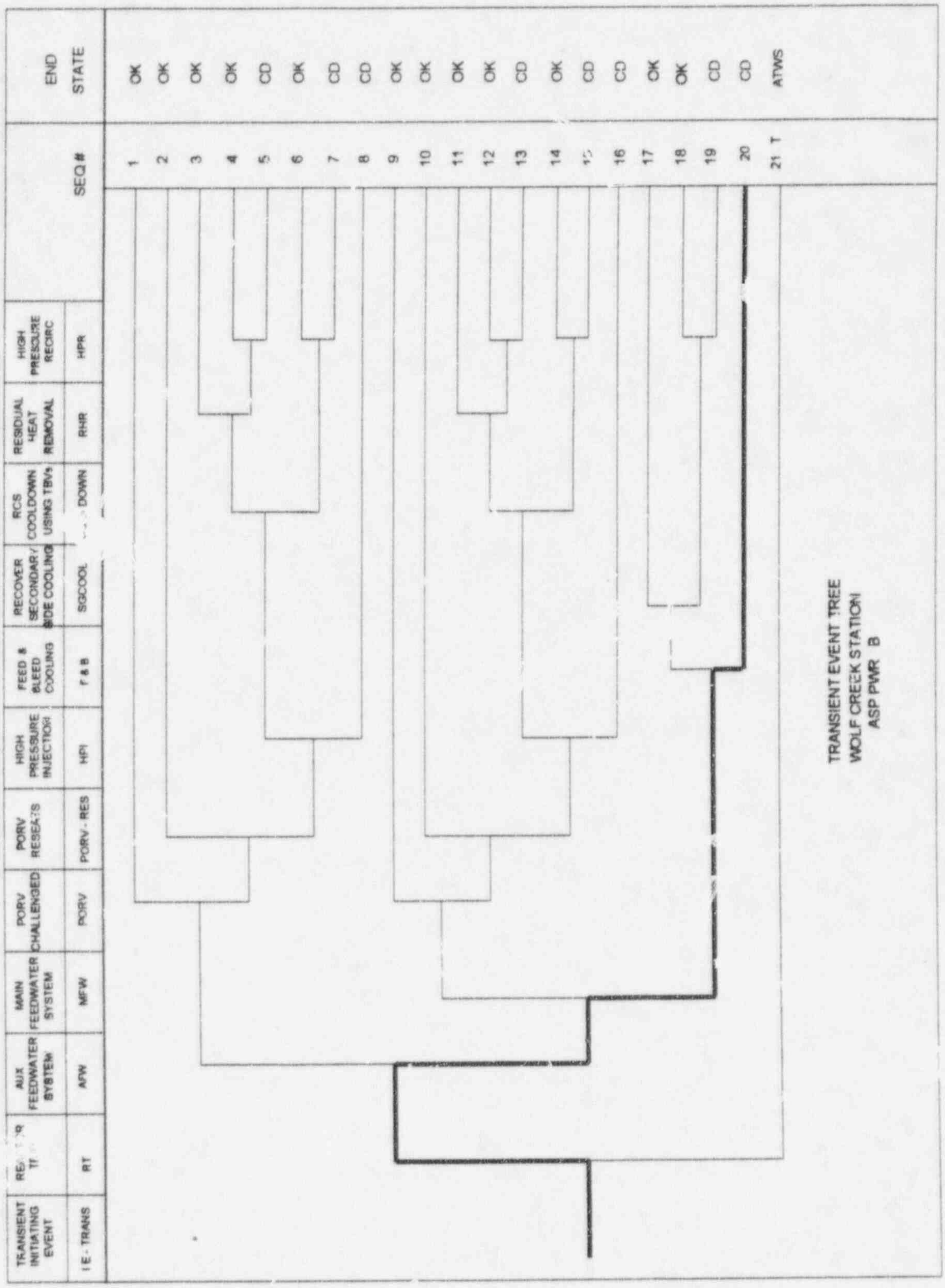

Fig. 2 Dominant core damage sequence given a transient for LER No. 482/96-001.

Table 1. Definitions and probabilities for selected basic events for LER 482/96-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss Of Offsite Power Initiating Event	6.9 E-006	1.0 E-003		Yes
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.6 E-006	0.0 E+000	IGNORE	No
IE-SLOCA	Small Loss Of Coolant Accident Initiating Event	1.0 E-006	0.0 E+000	IGNORE	No
IE-TRANS	Transient Initiating Event	5.3 E-004	1.0 E+000	TRUE	Yes
AFW-MDP-CF-AB	Common Cause Failure of All Motor Driven 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 Condensate Storage Tank	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 Station Blackout	3.4 E-001	3.4 E-001		No
AFW-XHE-XA-ESW	Operator Fails to Initiate Backup Water Source	1.0 E-003	1.0 E-003		No
EPS-DGN-CF-ALL	Common Cause Failure of Diesel Generators	1.6 E-003	1.6 E-003		No
EPS-DGN-FC-1B	Diesel Generator 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	EWS MDP Common Cause Failure	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 1. Definitions and probabilities for selected basic events for LER 482/96-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
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 HPR System	1.0 E+000	1.0 E+000		No
MFW-SYS-TRIP	Main Feedwater System Trips	2.0 E-001	2.0 E-001		No
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	3.4 E-001	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 Hours	2.2 E-001	1.1 E-001		Yes
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 Hours	6.7 E-002	3.6 E-004		Yes
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Batteries Deplete	5.8 E-002	3.6 E-003		Yes
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power (Seal LOCA)	5.7 E-001	4.4 E-001		Yes
PCS-XHE-XO-SEC	Operator Fails to Establish Secondary Cooling	2.0 E-001	2.0 E-001		No
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
PPR-SRV-CO-TRAN	PORVs Open During Transient	4.0 E-002	4.0 E-002		No
PPR-SRV-OO-1	PORV 1 Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No

Table 1. Definitions and probabilities for selected basic events for LER 482/96-001

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	7.3 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-XHE-NOREC	Operator Fails to Recover the RHR System	1.0 E+000	1.0 E+000		No

Table 2. Sequence conditional probabilities for LER 482/96-001

Event tree name	Sequence name	Conditional core damage probability (CCDP)	% Contribution
LOOP	39	1.2 E-005	25.2
TRANS	20	1.2 E-005	24.7
TRANS	08	4.2 E-006	8.7
TRANS	05	3.5 E-006	7.2
LOOP	28	3.4 E-006	7.0
LOOP	37	3.4 E-006	7.0
LOOP	38	2.1 E-006	4.4
TRANS	19	1.7 E-006	3.5
LOOP	27	1.2 E-006	2.4
LOOP	36	1.2 E-006	2.4
LOOP	24	1.0 E-006	2.2
LOOP	33	1.0 E-006	2.2
Total (all sequences)		4.8 E-005	

Table 3. Sequence logic for dominant sequences for LER 482/96-001

Event tree name	Sequence name	Logic
LOOP	39	/RT-L, EP, AFW-L-EP
TRANS	20	/RT, AFW, MFW, F&B
TRANS	08	/RT, /AFW, PORV, PORV-RES, HPI
TRANS	05	/RT, /AFW, PORV, PORV-RES, /HPI, /COOLDOWN, RHR, HPR
LOOP	28	/RT-L, /EP, /AFW-L-EP, PORV-SBO, SEALLOCA, OP-SL
LOOP	37	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, OP-SL
LOOP	38	/RT-L, EP, /AFW-L-EP, PORV-SBO, PORV-EP
TRANS	19	/RT, AFW, MFW, /F&B, SGCOOL, HPR
LOOP	27	/RT-L, EP, /AFW-L-EP, /PORV-SBO, SEALLOCA, /OP-SL, HPI
LOOP	36	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, /OP-SL, HPI
LOOP	24	/RT-L, EP, /AFW-L-EP, /PORV-SBO, SEALLOCA, /OP-SL, /HPI, /COOLDOWN, RHR, HPR
LOOP	33	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, /OP-SL, /HPI, /COOLDOWN, RHR, HPR

Table 4. System names for LER 482/96-001

System name	Logic
AFW-L	No or Insufficient AFW Flow During LOOP
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 High Pressure Injection System
HPR	No or Insufficient High Pressure Recirculation Flow
MFW	Failure of the Main Feedwater System
OP-SL	Operator Fails to Recover Offsite Power (Seal LOCA)
PORV	Power Operated Relief Valves (PORVs) Open During Transient
PORV-EP	PORVs Fail to Reclose (no Electric Power)
PORV-RES	PORVs Fail to Reseat
PORV-SBO	PORVs Open During Station Blackout
RHR	No or Insufficient Flow from the RHR System
RT	Reactor Fails to Trip During Transient
RT-L	Reactor Fails to Trip During LOOP
SEALLOCA	RCP Seals Fail During LOOP
SGCOOL	Failure of Secondary Cooling

Table 5. Conditional cut sets for higher probability sequences for LER 482/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets
LOOP Sequence 39		1.2 E-005	
1	92.6	1.1 E-005	EPS-DGN-FC-1B, EPS-XHE-NOREC, EWS-MDP-FC-1A, AFW-TDP-FC-1C, AFW-XHE-NOREC-EP
2	3.9	4.8 E-007	EPS-XHE-NOREC, EWS-MDP-FC-1B, EWS-MDP-FC-1A, AFW-TDP-FC-1C, AFW-XHE-NOREC-EP
3	3.5	4.4 E-007	EPS-DGN-CF-ALL, EPS-XHE-NOREC, AFW-TDP-FC-1C, AFW-XHE-NOREC-EP
TRANS Sequence 20		1.2 E-005	
1	54.2	6.6 E-006	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, EWS-MDP-CF-ALL, EWS-XHE-NOREC, 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-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-XHE-NOREC
4	4.3	5.2 E-007	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, HPI-XHE-XM-FB
5	2.7	3.3 E-007	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-2
6	2.7	3.3 E-007	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-CC-1
7	2.2	2.7 E-007	AFW-TNK-FC-CST1, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, EWS-MDP-CF-ALL, EWS-XHE-NOREC, 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 08		4.3 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-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-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-XHE-NOREC

Table 5. Conditional cut sets for higher probability sequences for LER 482/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets
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-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
LOOP Sequence 28		3.5 E-006	
1	92.6	3.2 E-006	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
2	3.9	1.4 E-007	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
3	3.5	1.2 E-007	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 37		3.5 E-006	
1	92.6	3.2 E-006	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
2	3.9	1.4 E-007	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
3	3.5	1.2 E-007	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 38		2.2 E-006	
1	46.3	1.0 E-006	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2
2	46.3	1.0 E-006	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
3	2.0	4.3 E-008	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2

Table 5. Conditional cut sets for higher probability sequences for LER 482/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets
4	2.0	4.3 E-008	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
5	1.8	3.8 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
6	1.8	3.8 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2
TRANS Sequence 19		1.7 E-006	
1	60.1	1.0 E-006	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XO-SEC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
2	16.8	2.9 E-007	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XO-SEC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
3	12.6	2.2 E-007	AFW-TDP-FC-1C, AFW-MDP-CF-AB, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XO-SEC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
4	2.5	4.3 E-008	AFW-TNK-FC-CST1, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XO-SEC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
5	1.1	1.9 E-008	AFW-XHE-XA-ESW, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XO-SEC, EWS-MDP-FC-1A, EWS-MDP-FC-1B, HPR-XHE-NOREC
LOOP Sequence 27		1.2 E-006	
1	75.7	9.1 E-007	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-XHE-NOREC
2	21.4	2.6 E-007	EWS-MDP-FC-1A, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, EWS-MDP-FC-1B, EWS-XHE-NOREC, HPI-XHE-NOREC
3	2.9	1.9 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-XHE-NOREC
LOOP Sequence 36		1.2 E-006	
1	75.7	9.1 E-007	EPS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-XHE-NOREC

Table 5. Conditional cut sets for higher probability sequences for LER 482/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets
2	21.4	2.6 E-007	EPS-MDP-FC-1A, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, EWS-MDP-FC-1B, EWS-XHE-NOREC, HPI-XHE-NOREC
3	2.9	3.5 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, EWS-MDP-CF-ALL, EWS-XHE-NOREC, HPI-XHE-NOREC
LOOP Sequence 24		1.1 E-006	
1	66.4	7.2 E-007	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
2	28.1	3.0 E-007	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, EWS-XHE-NOREC, HPR-XHE-NOREC
3	2.5	2.7 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
LOOP Sequence 33		1.1 E-006	
1	66.4	7.2 E-007	EWS-MDP-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, RHR-HTX-CF-ALL, HPR-XHE-NOREC
2	28.1	3.0 E-007	EWS-MDP-FC-1A, EWS-MDP-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, HPR-XHE-NOREC
3	2.5	2.7 E-008	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, RHR-XHE-NOREC, RHR-HTX-CF-ALL HPR-XHE-NOREC
Total (all sequences)		**4.8 E-005**	

a. The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability of the initiating events are given in Table 1 and begin with the designator "IE". The probabilities for the basic events are also given in Table 1.

GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event that occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include actual initiating events, such as a loss of off-site power (LOOP) or loss-of-coolant accident (LOCA), degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event.

Modeling Techniques

The models used for the analysis of 1995 and 1996 events were developed by the Idaho National Engineering Laboratory (INEL). The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The models are based on linked fault trees. Four types of initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) losses of offsite power (LOOPs), and (4) steam generator tube ruptures (PWR only). Fault trees were developed for each top event on the event trees to a supercomponent level of detail. The only support system currently modeled is the electric power system.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Description" section accurately describe the event as it occurred?

- Does the "Additional Event-Related Information" section provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?

- Does the "Modeling Assumptions" section accurately describe the modeling done for the event? Is the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions? This also includes assumptions regarding the likelihood of equipment recovery.

Appendix H of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER, AIT, or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the equipment and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures,*
- piping and instrumentation diagrams (P&IDs),*
- electrical one-line diagrams,*
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulator),* etc.

Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) of the use of the specific recovery measure on:

- the sequence of events,
- the timing of events,
- the probability of operator error in using the system or equipment, and
- other systems/processes already modeled in the analysis (including operator actions).

For example, Plant A (a PWR) experiences a reactor trip, and during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW modeling would be patterned after information gathered either from the plant FSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be

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* Revision or practices at the time the event occurred.

mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or accounted for in the IPE,
- procedures for using the system during recovery existed at the time of the event,
- the plant operators had been trained in the use of the system prior to the event,
- a clear diagram of the system is available (either in the FSAR, IPE, or supplied by the licensee),
- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis,
- the effects of using the standby feedwater system on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event.

- The specific LER, augmented inspection team (AIT) report, or other pertinent reports.

- A summary of the calculation results. An event tree with the dominant sequence(s) highlighted. Four tables in the analysis indicate: (1) a summary of the relevant basic events, including modifications to the probabilities to reflect the circumstances of the event, (2) the dominant core damage sequences, (3) the system names for the systems cited in the dominant core damage sequences, and (4) cut sets for the dominant core damage sequences.

Schedule

Please refer to the transmittal letter for schedules and procedures for submitting your comments.

References

1. L. N. Vanden Heuvel et al., Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volumes 21 and 22, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory and Science Applications International Corp., December 1995.

Enclosure 3

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Otto L. Maynard
Vice President Plant Operations

February 28, 1996

WO 96-0029

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Licensee Event Report 96-001-00

Gentlemen:

The attached Licensee Event Report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) concerning an Engineered Safety Features actuation and 10 CFR 50.73 (a)(2)(i)(B) for a violation of Technical Specifications. Both incidents are part of the same event and are therefore being documented in one report.

If you should have any questions regarding this submittal, please contact me at (316) 364-8831 extension 4450, or William M. Lindsay at extension 8760.

Very truly yours,

Otto L. Maynard

Otto L. Maynard

OLM/jad

Attachment

cc: L. J. Callan (NRC), w/a
W. D. Johnson (NRC), w/a
J. F. Ringwald (NRC), w/a
J. C. Stone (NRC), w/a

JE22 | 1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
WOLF CREEK GENERATING STATION	05000482	1 OF 12

TITLE (4)

Loss of Circulating Water Due to Icing on Traveling Screens Causes Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
01	30	96	96	001	00	02	28	96	FACILITY NAME	DOCKET NUMBER

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	1							
			20.402(b)		20.405(c)	x	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	98.3%		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)	73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)	
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)	
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
William M. Lindsay Manager Performance Assessment	316-364-8831

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
C	KE	SCRN	N/A	N/A					
D	BA	P	I-RAND	YES					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, completed EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT:

On January 30, 1996, at 0337 CST, operators in the Wolf Creek Generating Station (WCGS) Control Room manually tripped the reactor due to ice build-up on the Circulating Water (CW) system traveling screens. At the time of the reactor trip, all engineered safety features and the reactor protection system performed as expected except five control rods did not indicate full insertion into the reactor core. The Essential Service Water System (ESWS) train A was declared inoperable at 0747 CST, causing the A Motor Driven Auxiliary Feedwater Pump (MDAFWP) to be declared inoperable. The Turbine Driven Auxiliary Feedwater Pump (TDAFWP) auto-started as a result of the reactor trip, but was declared inoperable due to a packing leak at 0514 CST. With the TDAFWP and the A MDAFWP inoperable, Technical Specification 3.7.1.2.b requires the plant be in Hot Standby (MODE 3) in 6 hours and be in Hot Shutdown (MODE 4) within the following 6 hours. This requirement to be in MODE 4 was not met. The events surrounding the inoperability of ESWS Train A are detailed in LER 96-002-00.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT CONDITIONS AT THE TIME OF EVENT

MODE:	1
Power level:	98.3%
RCS Temperature:	584.6° Fahrenheit (F)
RCS Pressure:	2235 psig
Wind Speed:	10-25 mph
Lake Level:	1086.4
Circulating Water Condenser Inlet Temperature:	32.4°F
Outside Air Temperature:	7°F
Dew Point:	1°F
Wind Chill:	-12°F to -33°F

The A and C Circulating Water (CW) pumps [KE-P] were running and the B CW pump was tagged out for preventive maintenance. The A and C Service Water (SW) pumps [KG-P] were supplying service water. The standby SW pump B was tagged out for maintenance. The standby low flow SW pump was available. The Circulating Water Screenhouse (CWSH) traveling screens [KE-SCN] were in manual operation on slow speed in accordance with procedure STN GP-001, "Plant Winterization."

BASIS FOR REPORTABILITY

On January 30, 1996, at 0337 CST, Wolf Creek Generating Station Control Room operators manually tripped the reactor [RCT] which resulted in actuation of the reactor protection system [JD] and multiple engineered safety features [JE]. The decision to trip the reactor was based on icing conditions on the traveling screens preventing flow to the circulating water pumps and inhibiting the supply of CW water to the condenser [KE-COND]. The Reactor Protection System and Engineered Safety Features actuations are reportable pursuant to 10 CFR 50.73 (a)(2)(iv).

The Turbine Driven Auxiliary Feedwater Pump (TDAFWP) [BA-P] auto-started as a result of the trip, but was declared inoperable at 0514 CST, due to a packing leak. The A Motor Driven Auxiliary Feedwater Pump (MDAFWP) [BA-P]was declared inoperable at 0747 CST, due to the A Essential Service Water System (ESWS) [BI] train being out of service. Operators entered Technical Specification 3.7.1.2, Action Statement "b", which requires the plant be in Hot Shutdown (MODE 4) within six hours (i.e., 1347 CST) of being in Hot Standby (MODE 3) when two of the three auxiliary feedwater pumps are inoperable. At approximately 1000 CST, the Manager Operations was informed that the six hour Limiting Condition for Operation (LCO) might be exceeded. The Vice President of Operations and NRC Resident Inspector were notified shortly thereafter.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

The operators were cautioned by the Manager Operations to reach Mode 4 safely and in a timely manner, and not rush to try and meet the LCO time requirement. The plant entered MODE 4 at 1531 CST. This is reportable as a Technical Specification violation pursuant to 10 CFR 50.73 (a)(2)(i)(B). Refer to Root Cause for Missed Technical Specification Action Statement on page 10 for more information.

The icing conditions which affected the CW System also affected the ESWS and will be detailed in Licensee Event Report 96-002-00.

Background Information on the Circulating Water (CW) System

The CWSH has three suction bays which are physically separated by concrete walls. There are four 12" pipe sleeves which penetrate the walls that divide the A and B suction bays and the B and C suction bays. The level fluctuations in one bay are noticed to some degree in the others. (Refer to Figure One on Page 13)

Bar grate type trash racks [KE-RCK]are installed at the front of each suction bay. A solid apron plate is welded to the bar grates and extends from the top of the trash racks down to the 1075'-6" water level. The traveling screens are located 11'-2" from the trash racks.

A 42" CWSH keepwarm line routes some of the CW discharged from the plant to a header installed at the bottom of the CW bays, just in front of the trash racks. The warming line header in front of the trash racks has two discharge nozzles per traveling screen (12 total) for even distribution of flow. Per design criteria, the warming line is to maintain water temperature at a minimum of 34°F in the bays with lake temperature at 32°F. The warming line isolation valve, 1CW002, is to be either full open during winter or full closed during summer operation. The apron plate on the trash racks and the keep warm line are design features which are to prevent frazil ice formation on the trash racks and ice intrusion into the traveling screens.

The accumulation of frazil ice starts when the water becomes supercooled or drops below its freezing temperature. The water will supercool first at the surface, and when turbulence is present, will mix through the entire intake depth. Small crystals of ice - frazil ice - will be carried along with the supercooled water. Because the crystals are supercooled, and rapidly grow in size, they stick to any object they come into contact with, including trash racks (as long as these objects are at a temperature below freezing).

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

The traveling screens consist of basket assemblies installed onto a carrier chain. When the traveling screens on the upstream side emerge from the water, they are backwashed by spray as they travel through the fiberglass enclosure on the operating deck. " traveling screens are located outside the enclosed portion of the CWSH. Spray wash water is supplied by SW at about 80 psi. The traveling screen drive motors are interlocked, so they will not energize unless a pressure switch is actuated just downstream of a pneumatically operated spray wash control valve. Each traveling screen backwash header is supplied by its own spray wash control valve.

The traveling screen wash system can operate in an automatic or manual mode. In the automatic mode, the traveling screen wash system is started by either a timer or on high differential bay level. Differential level for each bay is continuously plotted on chart recorders on the local traveling screen wash control panel.

The following are the differential level settings for the control system:

- 4" differential level: traveling screen slow speed wash cycle auto starts and a SLOW SCREEN WASH alarm annunciates locally and in the control room (CWSH SCREEN TROUBLE annunicator) indicating the traveling screens are operating in slow speed.

- 6" differential level: traveling screen fast speed wash cycle auto starts and a FAST SCREEN WASH alarm annunciates in the control room (CWSH SCREEN TROUBLE annunicator) and locally, indicating the traveling screens are operating in fast speed.

- 10" differential level: TRAVELING SCREEN BLOCKED and CWSH SCREEN TROUBLE alarms annunciate in the Control Room and locally.

- 60" differential level: TRAVELING SCREEN DP VERY HI alarm annunciates locally. TRAVELING SCREEN EMERGENCY and CWSH SCREEN TROUBLE alarms annunciate in the control room, CW pumps should be immediately de-energized at this point.

- 1075' elevation: BAY LEVEL LOW alarm annunciates locally. CWSH BAY 1(2)(3) EMERG annunciates in the Control Room.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

DESCRIPTION OF EVENT

On January 30, 1996, at 0149 CST, the WCGS Control Room received CWSH SCREEN TROUBLE and TRAVELING SCREEN BLOCKED alarms. The Site Watch was dispatched to investigate. The alarms indicated increased differential pressure across the traveling screens (lake level vs. bay level). At 0158 CST, the Control Room received a bay 1 screen emergency alarm. The Site Watch reported the traveling screens to bays 1 and 3 were frozen and not rotating. The Site Watch also reported that bay 1 and bay 3 levels were approximately 8 feet lower than the normal water mark which could be seen on the wall of the bay. The Control Room then received an "ELECTRIC FIRE PUMP 1FP01PA RUN" annunicator, indicating an auto-start of the electric fire pump [KF-P]due to low SW pressure. The B ESWS pump was started with the intent to separate the ESWS from the SW system. The Control Room noticed that the turbine lube oil temperature was 120°F and increasing as well as other components' temperatures that are cooled by service water. The Turbine Building Watch was dispatched to locally control affected temperatures.

After discussions with Electrical Maintenance, the B CW pump clearance order was removed, and the B CW pump and the low flow service water pump were started (The B CW pump had been tagged out of service for preventative maintenance). The A SW pump and the A CW pump were secured to help reduce the differential pressure across the CW traveling screen. Bay 1 level was nine feet low and decreasing. Prior to starting the B CW pump, the traveling screens were placed in manual fast speed to attempt to prevent them from freezing. Shortly after the swap from the A to B CW pumps, the Site Watch reported bay 2 traveling screens were frozen, and the bay 1 and bay 2 levels were decreasing.

At 0211 CST, the A ESWS pump was started intending to separate ESWS and SW System, such that SW System was supplying only the Turbine Building loads. At 0252 CST, Control Room operators noticed that condenser vacuum was decreasing and started all three condenser vacuum pumps. At 0256 CST, Control Room operators began inserting control rods. At 0300 the Site Watch was directed to throttle the B and C CW pump discharge valves to establish a discharge pressure of 27 psig to stop the bay levels from further decreasing. Control Room operators started reducing turbine load, as required by the Alarm Response Procedure for CW Bay Emergency. The Control Room received reports that the CW bay levels were stable at approximately 6 - 8 feet below normal and vacuum was recovering as the turbine load was reduced.

At 0324 CST, the SS discussed plant status with the Control Room staff and the decision was made to trip the plant if CW bay levels decreased any further or any other plant problems arose. At 0337 CST, the Control Room received word from the Site Wat' that the CW bay levels were down to 12-13 feet below normal and that the low flow SW pump was vibrating. Control Room operators manually tripped the reactor, secured CW pumps, broke condenser vacuum, fast-closed the MSIVs [SB-ISV] and controlled the reactor coolant system [AB] temperature with the steam generator atmospheric relief valves [SB-RV].

Events occurring after the Reactor Trip

After the trip, all engineered safety features systems and reactor protection systems functioned as expected with the exception of control rods H-2, F-6, K-10, K-6 and H-8, which failed to fully insert into the reactor core (Refer to Additional Information section on page 12 for more information).

At 0503 CST, while stabilizing the plant in MODE 3, the Control Room was notified that the TDAFWP was spraying water. At 0505 CST, the Turbine Building Watch reported to the Control Room that the TDAFWP shaft gland packing was leaking and notified Maintenance. At 0514 CST, the TDAFWP inboard seal packing was determined to have failed and the pump was declared inoperable. Steam generator water levels were being maintained by the two MDAFWPs. The packing was replaced and the TDAFWP was declared functional at 1411 CST, on January 30, 1996. The required surveillances for operability will be completed during start-up from Refuel VIII.

At 0747 CST, the Control Room entered the following Technical Specifications: 3.7.4 for an inoperable ESWS A train; 3.8.1.1 for an inoperable diesel generator due to the unavailability of the ESWS A train; and, 3.7.1.2 action statement "b" for two inoperable auxiliary feedwater pumps. The TDAFWP and the A Motor Driven Auxiliary Feedwater Pump (MDAFWP) were out of service. Technical Specification 3.7.1.2 Action Statement "b" was entered at 0747 CST, and requires that with two auxiliary feedwater pumps inoperable, the plant be in at least Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours. The plant should have reached Hot Shutdown by 1347 CST, but did not make the MODE change until 1531. Refer to Root Cause for Missed Technical Specification Action Statement on page 10 for more information.

Also, at 0747 CST, the Control Room received a report that ice was developing in the A bay of ESWS. The A ESWS pump was secured due to low discharge pressure and high differential pressure across the strainer [BI-STR]. An administrative decision was made to declare a Notification of Unusual Event (NUE) and heighten plant staff awareness of the icing conditions in the ESWS bays. Further details on the NUE and the ESWS events that followed the reactor trip can be found in LER 96-002-00.

On January 30, 1996, at approximately 0530 CST, with all CW traveling screens and CW pumps secured, an inspection of the traveling screens was performed. As Maintenance personnel rotated the traveling screens to inspect for damage, a thick layer of ice was noticed packed on the traveling screens as they emerged from the water. The ice formed on the traveling screens in very thin scale-like pieces oriented mainly in a vertical direction and perpendicular to the traveling screen. The ice was densely packed onto the traveling screen.

During this inspection of the CWSH, the C SW pump and the electric motor driven fire protection pump were running. Levels had stabilized as soon as the CW pumps were secured. The SW and fire protection pumps running did not cause bay levels to drop.

Shear pins on the gear reducer drive chain sprocket for all six traveling screens failed during the event. Traveling screens A, C, and D incurred relatively minor damage. Some baskets were replaced on these traveling screens and they were returned to service by February 2, 1996. The B traveling screen frame was bent near the joint between the head section and the upper intermediate frame. The E and F traveling screens incurred substantial damage and could not be immediately repaired.

Diver inspections performed on February 1, 1996, in the outer bay between the traveling screens and trash racks indicated that the traveling screens were free of ice. The inspection for icing was performed just prior to commencing the air sparge. The air bubbler, (introducing air into the water) was implemented in accordance with Temporary Modification 96-009-ZC to prevent further ice build-up on the traveling screens.

ROOT CAUSE

Circulating Water System

The root cause of the event was that the design of the Circulating Water Intake Structure and associated traveling screens did not account for the harsh environmental conditions imposed on them during the events of January 30, 1996. A potential contributing factor to the event was the spraying of near freezing water onto the screens exposed to atmospheric conditions. This spray water caused an initial ice build-up on the screens, which grew when exposed to the water from the lake.

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APPROVED BY OMB NO. 3150-0104 . EXPIRES 5/31/95

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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Wolf Creek Generating Station	05000482	96	001	00	8 OF 12

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Detailed Evaluation

Warming line Flow

The CWSH is provided with a 42" warming line supplied from the CW Discharge Header and routed to the area just in front of the structure's trash racks. During normal operation, the line delivers approximately 34,000 gpm of water heated 30°F above lake temperature, during 3 Circulating Water Pump operation, and approximately 30,000 gpm of water heated to 40°F above lake temperature, for two pump operation. The head pressure which drives the warming line flow is a factor of CW Discharge Weir water level. This level varies by less than 3' between 2 pump and 3 pump operation. Thus, the change in warming line flow between 2 and 3 pumps is relatively minor. Since throttling of the CW pump discharge valves to prevent pump runout does not reduce the weir level by any significant amount, this practice actually helps increase the heat input to the bay by increasing the condenser outlet temperature even more. Heat balance calculations show that there is sufficient warming line heat input to prevent frazil ice during either mode of Circulating Water Pump operation with the plant on-line. These calculations were verified by measuring the warming line flow rate during two pump operation using an ultrasonic flow meter. The lack of any substantial build-up of ice on the trash racks also indicates that the warming line's design is adequate.

Traveling Screen Operation

The CW traveling screens are a series of stainless steel wire mesh basket assemblies with 3/8" openings. The screens are chain driven by a two speed motor (1800/450 RPM). In the slow speed operation (whether manual or automatic), 1 complete basket cycle takes 36 minutes. In fast speed operation (whether manual or automatic) 1 complete cycle takes 9 minutes.

While in manual slow speed operation, as procedurally required, the screens are exposed to the atmospheric conditions for approximately 7 minutes during the 36 minute cycle. While exposed to the atmosphere, the screens are backwashed by sprays supplied by SW at approximately 80 psi. This spray, when exposed to the atmospheric conditions present on January 30, 1996, caused an initial ice build-up on the steel mesh. As the traveling screens continued through their cycle, they were exposed to the water coming into the CW Bay. This incoming water was believed to be warmed sufficiently to prevent ice buildup on the trash racks in front of the screens, yet near enough to freezing that the initial ice build-up caused a rapid growth of ice to block the screens completely.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

CORRECTIVE ACTIONS

Circulating Water Screenhouse

Corrective Actions Completed:

Procedures SYS SW-121, "Circ Water Screen Wash System," and STN GP-001, "Plant Winterization," were revised on February 6, 1996, to delete the requirement to operate traveling screens continuously in the slow manual mode during cold weather or unusual icing conditions. These revisions allow the screens to be operated in an automatic mode in which the screens remain stationary without sprays until the system is started either by timer or on high differential level. The automatic mode will cycle the screens through two/thirds of one complete cycle before stopping, providing there is no high differential level still present across the screens. The screens will be automatically started again by either the timer or a high differential level. This mode of operation is recommended by Engineering when icing conditions are present. The winterization procedure previously had the screens started in slow, as a conservative action, during cold weather in anticipation of surface icing or of moss or other aquatic vegetation floating into the screens.

Long Term Corrective Actions

The traveling screens will be enclosed in a heated environment. This modification will be completed by October 1, 1996.

Root Cause

Missed Technical Specification Action Statement

Technical Specification 3.7.1.2, Action Statement "b", requires that, with two auxiliary feedwater pumps inoperable, the plant be in at least Hot Standby (MODE 3) within the next 6 hours and in Hot Shutdown (MODE 4) within the following 6 hours. This Technical Specification is applicable in MODES 1, 2, and 3.

The TDAFWP was declared inoperable at 0514 CST, on January 30, 1996, due to a packing leak, and the A MDAFWP was declared inoperable at 0747 CST, that same day, based on the A ESWS train being out of service.

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The root cause for late entry into MODE 4 was the amount of time it took to complete procedure, GEN 00-005, "Plant Ops from 20% Min Load to Hot Standby," Attachment "A" before a cooldown of the plant could begin. Many of the steps contained in Attachment "A" could have been delayed or GEN 00-006, "Hot Standby to Cold Shutdown," could have been performed concurrently with GEN 00-005 in order to begin the cooldown in a more timely manner.

Contributing Factors:

There are various reasons which contributed to the late entry into MODE 4:

1. This is the first time WCGS has had to perform a rapid cooldown to Mode 4. While the simulator training the operators received helped prepare them for this evolution, certain unanticipated conditions existed (e.g., loss of circulating water, one train of ESW, loss of the TDAFWP) and slowed the process.

2. The procedure did not provide guidance on which steps are required to be performed and which steps can be delayed for the purpose of a rapid cooldown.

3. Adherence to procedures is management's expectation and focused the Operator on ensuring every step of the procedure was completed.

CORRECTIVE ACTIONS

Technical Specification Action Statement Not Met

Guidance on which steps are mandatory and which actions may be delayed in an accelerated shutdown condition will be provided in appropriate operating procedures. This corrective action will be completed by the end of Refuel VIII.

Simulator training will be conducted to provide the operating crews with the opportunity to operate under conditions which require a forced cooldown in 6 hours. This training will be completed by the end of the third training cycle following Refuel VIII.

SAFETY SIGNIFICANCE

The CW and SW systems are not safety-related systems. They provide a source of heat rejection for plant auxiliary systems which require cooling during normal plant operation and normal plant shutdown. The systems also supply cooling water to the safety-related ESWS during normal operation.

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LICENSEE EVENT REPORT (LER)

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During this event, the A ESWS train experienced icing conditions on its trash racks, which caused it to be declared inoperable. More detail on the ESWS event can be found in LER 96-002-00.

Previous similar occurrences

Licensee Event Report 85-069-00 details an event involving high differential pressure across the traveling screens. Strong south winds blew dead pondweed growth into the traveling screens. The traveling screen wash control system malfunctioned and the traveling screens failed to backwash. Some of the traveling screens collapsed and a controlled shutdown was commenced. There have been no previous events due to icing conditions.

Additional Information

After the trip, all systems functioned as expected with the exception of control rods in core locations H-2, F-6, K-10, K-6 and H-8, which failed to fully insert immediately into the core. A review of the computer data captured from the Digital Rod Position Indicator (DRPI) system indicates that the first position at which the control rods stopped was H-2 at 30 steps, F-6 at 18 steps, K-10 at 12 steps, K-6 at 18 steps and H-8 at 18 steps.

The fuel assemblies under the subject control rods were all from Region 8 of the core, and were initially inserted into the reactor for Cycle 6. These fuel assemblies are of the Westinghouse V5H fuel design. These fuel assemblies were examined for fuel rod defects by ultrasonic testing at the end of Cycle 6 and 7. No defects were identified. Four of the control rods absorber material were made of Silver/Indium/Cadmium and one was made of Hafnium (location H-2). The hafnium control rod that did not fully insert is scheduled for discharge during Refueling Outage VIII.

The control rods reached the bottom of their travel after a short period of time, with the last rod indicating fully inserted approximately one hour and 20 minutes after the tri Rod drop testing was performed on February 2-3, 1996. Preliminary results from these tests indicate that only control rods in high burnup regions of the core are experiencing insertion problems. The data from these tests is being analyzed as part of WCNOC's continuing investigation efforts. Further testing is scheduled to occur after all fuel assemblies are offloaded from the core.

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Wolf Creek Generating Station	05000482	96	001	00	12 OF 12

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FIGURE 1

Circulating Water System Diagram