B.5 LER No. 282/96-012

Event Description: Loss of offsite power to safeguards buses on both units

Date of Event: June 29, 1996

Plant: Prairie Island 1 and 2

B.5.1 Event Summary

Both units were operating at 100% power on June 29, 1996, when strong isolated thunderstorms caused the failure of three 345-kV offsite transmission lines to the plant substation. Both unit reactors tripped in response to a loss of load. All four emergency diesel generators (EDGs) started as expected. Both Unit 2 safeguards buses and one Unit 1 safeguards bus were immediately powered by their respective EDGs. A single 345-kV transmission line continued to supply power to the plant substation, the Unit 1 normal buses, and one Unit 1 safeguards bus (bus 15). However, the voltage supplied by the single offsite power supply line was so unstable that approximately 7 min after the trip, bus 15 automatically transferred to its associated EDG. A stable offsite source was not reestablished for approximately 5 h, and both units were cooled by natural circulation cooling until offsite power was restored.[1] The estimated conditional core damage probability (CCDP) for this grid-based loss of offsite power (LOOP) is 5.3×10^{-5}. This CCDP estimate is applicable to both units.

B.5.2 Event Description

On June 29, 1996, at approximately 1418, the Blue Lake 345-kV transmission line tripped off line and remained off line following a single phase-to-ground fault resulting from a passing thunderstorm. Both units remained at 100% power following the loss of this offsite connection to 345-kV bus 1 in the plant substation, shown in Fig. B.5.1. Approximately 11 min later, at 1429, severe weather with "straight-line" winds destroyed several support structures, causing three phase faults in both Red Rock lines. Both Red Rock 345-kV lines tripped. At this point, only the Byron 345-kV line remained in service and powered 345-kV bus 2 in the Prairie Island switchyard. Both unit generators were aligned to 345-kV bus 1 and subsequently tripped from 100% power because of the loss of load. All reactor coolant pumps (RCPs) on both units tripped because of low frequency, resulting in decay heat being removed by natural circulation cooling only.

All four EDGs started as designed upon the LOOP. Safeguards buses 16 (Unit 1) and 25 and 26 (Unit 2) were immediately sequenced onto their respective EDGs. Safeguards bus 15 (Unit 1) continued to be supplied by offsite power via the Byron line. Voltage on the Byron line was low and unstable, and safeguards bus 15 automatically loaded onto EDG D1 approximately 7 min after the reactor trip. Unit 1 normal 4-kV buses (nonsafety related) transferred to the 1R transformer supplied from the 345-kV bus 2 via the Byron line and continued to be powered from this unstable source throughout the event. The two normal 4-kV buses at Unit 2 transferred to the 2R transformer, which is supplied power from the 345-kV bus 1; this bus was deenergized at the time. At this time, the licensee declared an Unusual Event.

At approximately 1800, the Unit 2 normal 4-kV buses were powered from 345-kV bus 2, which was still in a degraded voltage condition (below 330 kV). At 1925, voltage on the Byron line was restored within the normal range after the utility purchased additional power and reset transformer taps in the switchyard. Because all EDGs were operating satisfactorily, emphasis was placed on restoring forced circulation cooling in both units rather than transferring the safeguards bus power supply to the appropriate offsite source. This priority was established because, in case of an EDG failure with an offsite power source available, the load sequencers would have automatically transferred safeguards bus loads to the offsite power source. At 1953, forced circulation cooling was established at Unit 1. At 2028, forced circulation cooling was established at Unit 2.

On June 30, 1996, at 0135, the Blue Lake 345-kV line was restored. The process of transferring the safeguards buses from the EDGs to an offsite power source began at this point and was completed at 1035. The licensee exited the Unusual Event at this point.

B.5.3 Additional Event-Related Information

The diesel-driven cooling water pumps that provide plant service water started as designed during the event to supply a source of cooling water throughout the event. Additionally, the Unit 1 motor-driven cooling water pump operated throughout the event while powered from the degraded offsite power source. Therefore, a heat sink was available throughout the event for the component cooling water system, the EDGs, and the auxiliary feedwater (AFW) pumps.[2] All AFW pumps operated as designed.

The EDGs on each unit can be cross-tied to the opposite unit using two 4160-V breakers connected in series. Also the motor-driven AFW pump on each unit can be cross-tied to the opposite unit through two manually operated valves.[2]

B.5.4 Modeling Assumptions

Five hours passed before a stable offsite power supply at normal voltage was reestablished. This did not occur until the utility purchased additional power and reset switchyard transformer taps. If the EDGs had not started as expected, the utility may have expedited the return of the offsite Byron 345-kV line, but it is not known if this would have been possible. Therefore, the Byron line was not considered a viable offsite power source for the safeguards buses, and the event was modeled as a grid-centered LOOP on each unit. The probability of not recovering offsite power in the short term is included in the initiating event probability (IE-LOOP). This term was set to the probability for a grid-based LOOP assuming operators fail to recover offsite power in the short term (4.8×10^{-1}).

The grid-based LOOP probability of short-term and long-term offsite power recovery for a grid-centered LOOP and the probability of a RCP seal loss-of-coolant accident (LOCA) following a postulated station blackout (SBO) were developed based on data distributions contained in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*.[3] The RCP seal LOCA models were developed as part of the NUREG-1150 probabilistic risk assessment (PRA) efforts. Both are described in *Revised LOOP Recovery and PWR Seal LOCA Models*.[4] The probabilities for the following basic events (defined in Table B.5.1) are

based on these models: IE-LOOP, OEP-XHE-NOREC-6H, OPE-XHE-NOREC-BD, OPE-XHE-NOREC-SL, and RCS-MDP-LK-SEALS.

Each of the four EDGs at Prairie Island can supply the power requirements for the hot shutdown loads for its associated unit and one train of essential loads of the opposite unit in case of an SBO on the second unit.[5] This is accomplished through two manual cross-tie breakers between buses 15 and 25 or buses 16 and 26. A basic event was added (EPS-XHE-XE-XTIE) to the model to account for the failure of the operator to initiate the cross-tie between buses according to the established procedure.[6] This cross-tie basic event assumes that a safety injection signal does not exist on both units at the same time. EPS-XHE-XE-XTIE was set at 3.2×10^{-3} based on the individual plant examination (IPE) (Ref. 5, Table 3.3-3). A basic event was also added to account for the mechanical failure of the two cross-tie breakers in series (Ref. 5, Table 3.3-1); however, this event did not influence any of the significant core damage sequences. The base-case common-cause failure probability of the EDGs (EPS-DGN-CF-ALL) was adjusted from 1.6×10^{-3} to 7.0×10^{-4} to account for all four EDGs (Ref. 7, Table 5-9, $\alpha_{4s} = 0.0164$; and Table 5-12, $\alpha_{4r} = 0.0174$).

The IPE (Ref. 5, page 2-7) indicates that the first station battery will fail after 2 h. Because a stable offsite power source was not restored until about 5 h into the LOOP, basic event OEP-XHE-NOREC-2H was set to TRUE (i.e., the probability of this event is 1.0 given that power was not restored in the short term). This had little impact on the CCDP calculated for this event.

Each motor-driven AFW pump can be cross-tied to supply feedwater to the opposite unit. A basic event was added (AFW-XHE-XE-XTIE) to the model to account for the failure of the operator to initiate the cross-tie between units. AFW-XHE-XE-XTIE was set at 3.2×10^{-2} (Ref. 5, Table 3.3-3). A basic event was also added to account for the potential failure of the cross-tie valves.

B.5.5 Analysis Results

The CCDP for this event is estimated to be 5.3×10^{-5}. The dominant core damage sequence for this event (sequence 28 on Fig. B.5.2) involves

- a LOOP,
- a successful reactor trip,
- failure of the emergency power supplies (SBO),
- success of the AFW system,
- no challenge to the power-operated relief valves (PORVs),
- failure of the RCP seals during the LOOP, and
- failure to recover offsite power after the RCP seals fail.

This sequence accounts for about 38% of the total contribution to the CCDP. Sequence 37 is similar to LOOP sequence 28, except LOOP sequence 37 involves a PORV lift and successful reclosure. Combined, these two sequences account for 60% of the total contribution to the CCDP.

Sequences involving battery depletion (sequences 21 and 30) account for 16% of the total contribution to the CCDP. An SBO is involved in 98% of the dominant core damage sequences. Failure of the AFW system only occurs in 8% of the dominant core damage sequences.

Definitions and probabilities for selected basic events are shown in Table B.5.1. The conditional probabilities associated with the highest probability sequences are shown in Table B.5.2. Table B.5.3 lists the sequence logic associated with the sequences listed in Table B.5.2. Table B.5.4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table B.5.5.

In a sensitivity study, the probability of failing to recover offsite power in the short term (IE-LOOP) was set to TRUE [LOOP with no short-term (30 min) recovery possible]. SBO values were adjusted to account for the revised period based on the code associated with the *Revised LOOP Recovery and PWR Seal LOCA Models*.[4] The calculated CCDP in this case is 6.2×10^{-5}. The dominant sequence remained the same for this sensitivity study.

Finally, the IPE indicated that an additional 2 h was available to recover offsite power before core damage following battery depletion. An estimate of the CCDP assuming 2 h before battery depletion and an additional 2 h before uncovering the core is calculated to be 4.1×10^{-5}. Again, adjusted station blackout values were based on the code associated with the *Revised LOOP Recovery and PWR Seal LOCA Models*.[4] The dominant sequences remained the same as for the initial analysis.

B.5.6 References

1. LER 282/96-012, Rev. 0, "Loss of Offsite Power to Unit 2 and Degraded Offsite Power to Unit 1 Followed by Reactor Trips of Both Units," July 29, 1996.

2. Northern States Power Company, Prairie Island Nuclear Generating Plant, *Updated Safety Analysis Report*.

3. *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032.

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

5. Prairie Island Nuclear Generating Plant, *Individual Plant Examination*.

6. Prairie Island Procedure 1ECA-0.0, Rev. 11, "Loss of All Safeguards AC Power."

7. *Common-Cause Failure Data Collection and Analysis System*, INEL-94/0064, December, 1995.

Figure removed during SUNSI review.

Fig. B.5.1. Switchyard at Prairie Island.

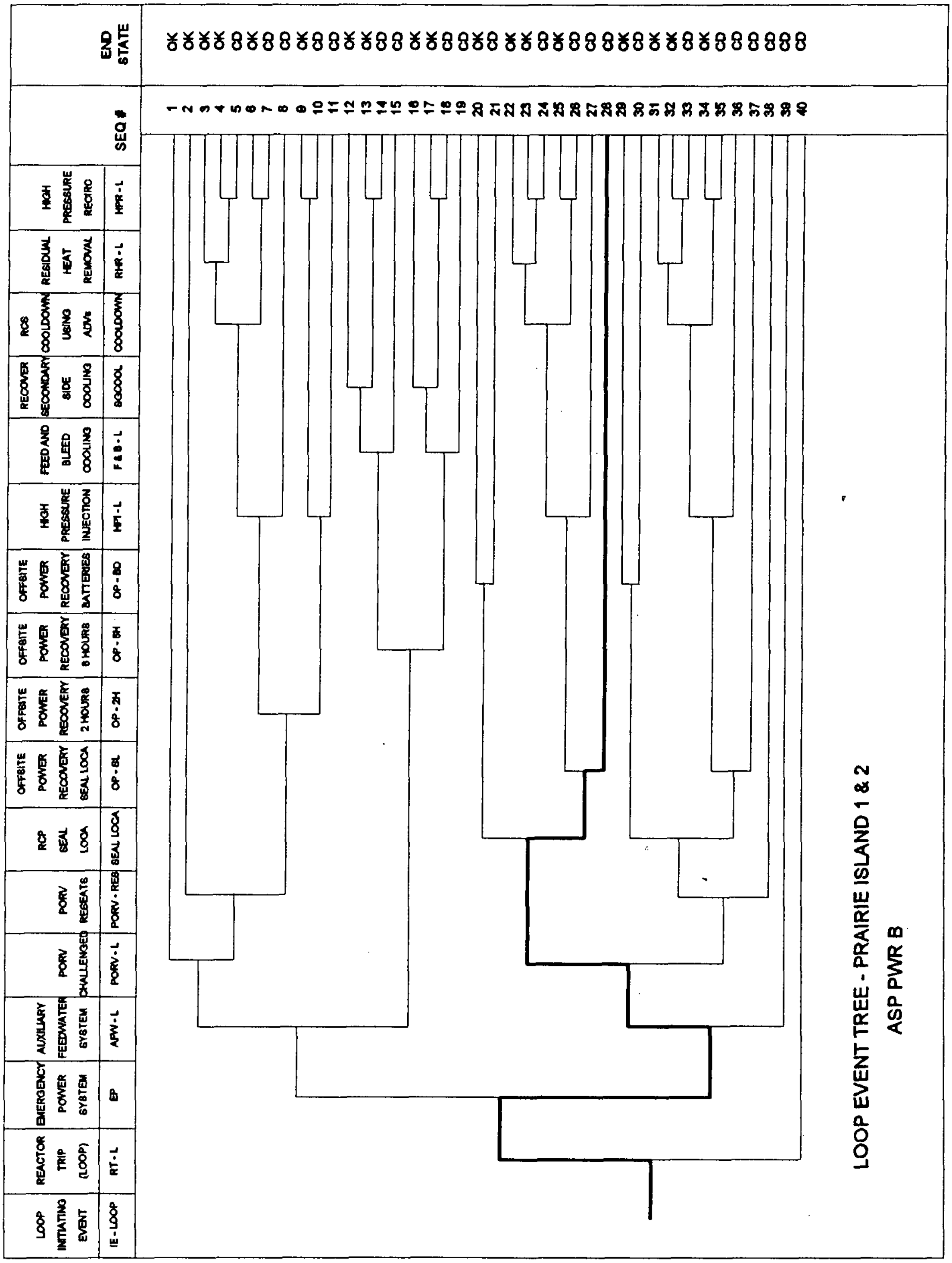

Fig. B.5.2. Dominant core damage sequence for LER No. 282/96-012.

Table B.5.1. Definitions and Probabilities for Selected Basic Events for LER No. 282/96-012

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–LOOP	5.9 E-006	4.8 E-001	GRID LOOP	Yes
IE-SGTR	Initiating Event–Steam Generator Tube Rupture	1.0 E-006	0.0 E+000	IGNORE	Yes
IE-SLOCA	Initiating Event–Small-Break LOCA	1.0 E-006	0.0 E+000	IGNORE	Yes
IE-TRANS	Initiating Event–Transient (TRANS)	5.3 E-004	0.0 E+000	IGNORE	Yes
AFW-TDP-FC-TDP	Turbine-Driven AFW Pump Fails	3.5 E-002	3.5 E-002		No
AFW-XHE-NOREC-EP	Operator Fails to Recover AFW System During an SBO	3.4 E-001	3.4 E-001		No
AFW-XHE-XE-XTIE	Operator Fails to Cross-Tie the Motor-Driven AFW Pump	3.2 E-002	3.2 E-002	NEW	No
EPS-DGN-CF-ALL	Common-Cause Failure of EDGs	7.0 E-004	7.0 E-004		No
EPS-DGN-FC-1	EDG 1 Fails	4.2 E-002	4.2 E-002		No
EPS-DGN-FC-2	EDG 2 Fails	4.2 E-002	4.2 E-002		No
EPS-XHE-NOREC	Operator Fails to Recover Emergency Power	1.0 E+000	1.0 E+000	TRUE	No
EPS-XHE-XE-XTIE	Operator Fails to Cross-Tie the Safeguards ac Buses	3.2 E-003	3.2 E-003	NEW	No
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 h	2.1 E-001	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 h	9.9 E-002	3.6 E-004	GRID LOOP	Yes
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Battery Depletion	6.1 E-002	3.3 E-002	GRID LOOP	Yes
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power After a Seal LOCA	5.9 E-001	4.5 E-001	GRID LOOP	Yes
PPR-SRV-CO-SBO	PORVs Lift During an SBO	3.7 E-001	3.7 E-001		No
PPR-SRV-OO-1	PORV 1 Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No

Table B.5.1. Definitions and Probabilities for Selected Basic Events for LER No. 282/96-012 (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
RCS-MDP-LK-SEALS	RCP Seals Fail Without Cooling and Injection	2.3 E-001	2.1 E-001	GRID LOOP	Yes

Table B.5.2. Sequence Conditional Probabilities for LER No. 282/96-012

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Percent contribution
LOOP	28	2.0 E-005	37.9
LOOP	37	1.1 E-005	22.2
LOOP	38	7.5 E-006	14.1
LOOP	21	5.5 E-006	10.4
LOOP	39	4.0 E-006	7.5
LOOP	30	3.2 E-006	6.1
Total (all sequences)		**5.3 E-005**	

Table B.5.3. Sequence Logic for Dominant Sequences for LER No. 282/96-012

Event tree name	Sequence number	Logic
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
LOOP	21	/RT-L, EP, /AFW-L-EP, /PORV-SBO, /SEALLOCA, OP-BD
LOOP	39	/RT-L, EP, AFW-L-EP
LOOP	30	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, /SEALLOCA, OP-BD

Table B.5.4. System Names for LER No. 282/96-012

System name	Logic
AFW-L-EP	No or Insufficient AFW Flow During an SBO
EP	Failure of Both Trains of Emergency Power
OP-BD	Operator Fails to Recover Offsite Power Before Battery Depletion
OP-SL	Operator Fails to Recover Offsite Power After a Seal LOCA
PORV-EP	PORVs Fail to Reclose (No Electric Power)
PORV-SBO	PORVs Open During an SBO
RT-L	Reactor Fails to Trip During a LOOP
SEALLOCA	RCP Seals Fail During a LOOP

Table B.5.5. Conditional Cut Sets for Higher Probability Sequences for LER No. 282/96-012

Cut set number	Percent contribution	CCDP[a]	Cut sets[b]
LOOP Sequence 28		2.0 E-005	
1	98.8	2.0 E-005	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, /PPR-SRV-CO-SBO, OEP-XHE-NOREC-SL
2	0.8	1.6 E-007	EPS-DGN-FC-1, EPS-DGN-FC-2, EPS-XHE-XE-XTIE, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, /PPR-SRV-CO-SBO, OEP-XHE-NOREC-SL
LOOP Sequence 37		1.2 E-005	
1	98.8	1.2 E-005	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, PPR-SRV-CO-SBO, OEP-XHE-NOREC-SL
2	0.8	1.0 E-007	EPS-DGN-FC-1, EPS-DGN-FC-2, EPS-XHE-XE-XTIE, EPS-XHE-NOREC, PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 38		7.5 E-006	
1	49.4	3.7 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-1
2	49.4	3.7 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-2
LOOP Sequence 21		5.6 E-006	
1	98.8	5.5 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, /RCS-MDP-LK-SEALS, /PPR-SRV-CO-SBO, OEP-XHE-NOREC-BD
2	0.8	4.4 E-008	EPS-DGN-FC-1, EPS-DGN-FC-2, EPS-XHE-XE-XTIE, EPS-XHE-NOREC, /RCS-MDP-LK-SEALS, /PPR-SRV-CO-SBO, OEP-XHE-NOREC-BD
LOOP Sequence 39		4.0 E-006	
1	99.4	4.0 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, AFW-TDP-FC-TDP, AFW-XHE-NOREC-EP

Table B.5.5. Conditional Cut Sets for Higher Probability Sequences for LER No. 282/96-012 (Continued)

Cut set number	Percent contribution	CCDP[a]	Cut sets[b]
LOOP Sequence 30		3.3 E-006	
1	98.8	3.2 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, PPR-SRV-CO-SBO, /RCS-MDP-LK-SEALS, OEP-XHE-NOREC-BD
2	0.8	2.6 E-008	EPS-DGN-FC-1, EPS-DGN-FC-2, EPS-XHE-XE-XTIE, EPS-XHE-NOREC, PPR-SRV-CO-SBO, /RCS-MDP-LK-SEALS, OEP-XHE-NOREC-BD
Total (all sequences)		**5.3 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 probabilities for the initiating events and the basic events are given in Table B.5.1.

[b]Basic event EPS-XHE-NOREC is a type TRUE event and these type of events are normally not included in the output of fault tree reduction programs, but has been added to aid in understanding the sequences to potential core damage associated with the event.