July 3, 2001

Mr. R. P. Necci
Vice President - Nuclear Technical Services
c/o Mr. David A. Smith
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SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF SEPTEMBER 2000 OPERATIONAL CONDITION - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 (TAC NO. MB2186)

Dear Mr. Necci:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) Program analysis of an operational condition (Enclosure 1) which was discovered at Millstone Nuclear Power Station, Unit No. 2, on September 20, 2000. The condition was documented in the Nuclear Regulatory Commission (NRC) Inspection Report No. 05000336/2000-011 dated October 30, 2000. The results of our preliminary analysis indicate that the condition may be a precursor (difference in core damage probability $\geq$ 1 x10^{-6}) in the ASP Program.

In assessing operational conditions, the NRC staff strives 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 our review process is to provide as realistic an analysis of the significance of the condition as possible.

In order for us to incorporate your comments, perform any required re-analysis, and prepare the final report of our analysis in a timely manner, we are requesting that you complete your review and to provide any comments within 60 calendar days from the date 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 condition is made publicly available. As soon as our final analysis of this condition has been completed, we will provide for your information the final precursor analysis and the resolution of your comments.

We have also enclosed information to facilitate your review (Enclosure 2). Enclosure 2 contains specific guidance for performing the requested review, identifies the criteria that 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 information that you should provide to support such a claim.

Please contact me at (301) 415-3199 if you have any questions regarding this request. This request is covered by the existing OMB clearance number (3150-0104) for NRC staff follow up review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

/RA/

John Harrison, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Preliminary ASP Analysis for 2000 Event
2. ASP Review Guidance

cc w/encls: See next page

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DATE	6/26/01	6/26/01	6/29/01

Official Record Copy

July 3, 2001

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Unit 2

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July 3, 2001

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July 3, 2001

Preliminary Precursor Analysis

Accident Sequence Precursor Program --- Office of Nuclear Regulatory Research

Millstone Unit 2	Failure of the turbine-driven auxiliary feedwater pump during a routine surveillance test	
Event Date 09/20/00	Inspection Report: 336/2000-011	ΔCDP = 7.2×10^{-6}

Condition Summary

On August 23, 2000, during a routine surveillance test, while raising the turbine-driven auxiliary feedwater (TDAFW) pump speed from approximately 1400 rpm to its rated speed of 4400 rpm, the control room noted that the turbine speed would at times not respond to motion of the speed control switch and at other times rise in spurts. Also during the start, a senior reactor operator in the pump room noted that at times the speed control servo motor was turning without any corresponding motion of the turbine governor steam valve. Engineering personnel and the Shift Manager evaluated the condition and concluded that the observed governor valve response was consistent with expected response in that, at certain points, substantial motion of the speed control servo motor is necessary to cause a perceptible change in governor steam valve position.

The next operation of the TDAFW pump was a regularly scheduled surveillance test performed on September 20, 2000. During the test, the turbine was started and warmed at its minimum operating speed of approximately 1400 rpm. Following the warm-up, control room operators were unable to increase turbine speed above its starting speed through operation of the TDAFW pump speed control switch. The discharge pressure of the pump at that speed was less than 200 psig, which was insufficient pressure for the pump to provide feedwater to the steam generators. (Refs. 1 and 2)

Concurrent with this condition, the “C” High Pressure Safety Injection (HPSI) pump had a low oil level from July 6 to August 3, 2000. Information from the pump vendor indicated that the as-found oil level would have allowed the pump to operate for an estimated 30 hours before failure. Because this time to failure exceeds the modeled mission time for HPSI of 24 hours, this additional condition was not included in the condition assessment. (Ref. 3)

Cause. Following the surveillance test failure, the licensee disassembled the speed control servo motor and the associated coupling. The mechanic performing the disassembly found the self-locking nut loose and the outward bend in the clutch spring sheared off. The apparent cause assigned by the licensee for pump failure was an age-related failure of a spring in the coupling that joined the servo-motor, which provided remote operation of the governor, to the turbine governor. The licensee also determined that the locking nut on the speed control knob, when tight, added tension to the spring inside the knob. The spring was part of the original equipment, since the plant went in service in 1975.

Condition duration. The condition duration was half of the July-to-August surveillance interval plus the full surveillance interval from August to September (1 ½ months or 42 days). See Modeling Assumptions section for further details.

Recovery opportunity. Because of the lack of engagement between the manual speed control knob and the governor shaft, the servo motor could not turn the governor shaft. Reference 1 concluded that this failure mechanism would not readily allow recovery of the pump by local manipulation of the speed control knob.

Analysis Results

- **Importance**[1]

  The risk significance of the TDAFW pump being unavailable is determined by subtracting the nominal core damage probability from the conditional core damage probability:

Conditional core damage probability (CCDP) =	7.9×10^{-6}
Nominal core damage probability (CDP) =	$-\underline{6.5\times10^{-7}}$
Importance (ΔCDP = CCDP - CDP) =	7.2×10^{-6}

  The estimated importance (CCDP-CDP) for the condition was 7.2×10^{-6}. This is an increase of 7.2×10^{-6} over the nominal CDP for the 42-day period when the TDAFW pump was not available.

  The Accident Sequence Precursor Program acceptance threshold is an importance (ΔCDP) of 1×10^{-6}.

- **Dominant sequence**

  The dominant core damage sequence for this condition is a station blackout (SBO) sequence (Sequence 23-28). The events and important component failures in this sequence (shown in Sequence 23, Figure 1, and Sequence 28, Figure 2) include:

  - a loss of offsite power initiating event,
  - successful reactor trip,
  - failure of the emergency power system due to independent and common cause failures of the emergency diesel generators,
  - failure of the auxiliary feedwater system, and
  - failure to recover offsite power in the short term (1-hour).

- **Results tables**

  - The conditional probability of the dominant sequence is shown in Table 1.
  - The event tree sequence logic for the dominant sequence is provided in Table 2a.

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[1] Since this condition did not involve an actual initiating event, the parameter of interest is the measure of the incremental increase between the conditional probability for the period in which the condition existed and the nominal probability for the same period but with the condition nonexistent and plant equipment available. This incremental increase or "importance" is determined by subtracting the CDP from the CCDP. This measure is used to assess the risk significance of hardware unavailabilities especially for those cases where the nominal CDP is high with respect to the incremental increase of the conditional probability caused by the hardware unavailability.

- The conditional cut sets for the dominant sequence are provided in Table 3.

Modeling Assumptions

- **Assessment summary**

  This event was modeled as an at-power condition assessment with the TDAFW pump unavailable for 42 days. The Revision 2QA of the Summer Standardized Plant Analysis Risk (SPAR) model (Ref. 6) was used for this assessment. The SPAR Revision 2QA model includes event trees for transients (including loss of feedwater and a transfer tree for anticipated transient without scram or ATWS), loss of offsite power (including a transfer tree for station blackout), small loss-of-coolant accident, and steam generator tube rupture. These event trees were used in the analysis. The discussion below provides the bases for significant changes to the model.

- **Condition duration**

  The condition duration was concluded to be half of the July-to-August surveillance interval plus the full surveillance interval from August to September (1 ½ months or 42 days). This conclusion is based on the following:

  - The TDAFW pump failed to deliver sufficient discharge pressure to provide feedwater to the steam generators during the September monthly surveillance test.

  - The erratic operations observed during the August surveillance could have indicated that the spring was broken at that time, and that the friction of the locking nut was providing connection between the spindle and the knob, or that the locking nut was loose and there was intermittent connection between the spring and the knob. This failure mode could suggest that this condition may have been in place prior to the August monthly surveillance test.

  - There was no evidence of the pump being in a degraded condition during the July monthly surveillance test.

  - Since the exact time of failure is unknown, the exposure time is half of the duration between the time at which the pump was last known to be in successful condition (July monthly surveillance test), and the time at which the pump functionality was suspect (August monthly surveillance test). Further, since actions taken to correct the erratic operations of the pump after the August test were determined to be unsuccessful during the September monthly surveillance test, the exposure period was extended to include the time up to the September monthly surveillance test failure.

- **Basic event probability changes**

  Table 4 provides the basic events that were modified to reflect the event condition being analyzed. The bases for these changes are as follows:

- ***Probability of failure of the TDAFW pump (AFW-TDP-FC-TDP).*** The probability that the pump would fail to start was set to TRUE (failure probability of 1.0) to reflect the failure of the train to provide flow.

- ***Nonrecovery probabilities for the auxiliary feedwater system*****.** Based on the failure cause (speed control mechanism), the TDAFW pump was not considered recoverable within the time period available for an SBO event (dominant sequence). The sequence nonrecovery probabilities for the dominant sequences were modified to account for the nonrecovery of the AFW system during a SBO (see Table 5).

- ***Other changes of sequence nonrecovery probabilities*****.** The generic sequence nonrecovery probabilities from the SPAR model were reviewed and modified, as necessary, to appropriately reflect the minimum cut sets of the important dominant sequences. Table 4 shows the sequence nonrecovery probabilities for the dominant sequences. Table 5 provides the bases for those probabilities.

● **Model update**

The SPAR model for Millstone 2 was updated to account for:

- updates of system/component failure probabilities and initiating event frequencies based on recent operating experience,

- changes in the probability of failing to recover offsite power in the short term to account for estimated core uncovery times for station blackout sequences (Ref. 7), and

- changes in the reactor coolant pump seal loss-of-coolant accident (LOCA) model (Ref. 8).

Bases for these updates are described in the footnotes to Table 4.

References

1. NRC Inspection Report 50-336/2000-011, 50-423/2000-011, October 30, 2000 (ADAMS Accession No. ML003764492).

2. EA-00-236, *Final Significance Determination for a White Finding and Notice of Violation at Millstone 2*, NRC Inspection Report No. 05000336/2000-011, December 6, 2000 (ADAMS Accession No. ML003774806).

3. NRC Inspection Report 50-336/2001-003, March 19, 2001 (ADAMS Accession No. ML010790130).

4. Reserved.

5. Reserved.

6. M. B. Sattison, et al., *Simplified Plant Analysis Risk Model for Millstone Unit 2,* Revision 2QA , Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID, December 1997.

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

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

9. Memorandum from Ashok C. Thadani to William D. Travers, “Closeout of Generic Safety Issue 23: Reactor Coolant Pump Seal Failure,” U.S. Nuclear Regulatory Commission, Washington, DC, November 8, 1999.

10. F. M. Marshall, et al., *Common-Cause Failure Parameter Estimations*, NUREG/CR-5497, U.S. Nuclear Regulatory Commission, Washington, DC, October 1998.

11. G. M. Grant, et al., *Reliability Study: Emergency Diesel Generator Power System, 1987-1993*, NUREG/CR-5500, Vol. 5, U.S. Nuclear Regulatory Commission, Washington, DC, September 1999.

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

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

Table 1. Conditional probabilities associated with the highest probability sequences

Event tree name	Sequence no.	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP - CDP)[2]
LOOP	23-28	6.3E-006	1.0E-007	—
Total (all sequences)[1]		**7.9E-006**	**6.5E-007**	**7.2E-006**

Notes:
1. Total CCDP and CDP includes all sequences (including those not shown in this table).
2. Importance is calculated using the total CDP and total CDP from all sequences. Sequence level importance measures are not additive.
3. (File name: GEM 336-00-S01 5-18-2001 081033.WPD)

Table 2a. Event tree sequence logic for dominant sequence

Event tree name	Sequence no.	Logic (“/” denotes success; see Table 2b for top event names)
LOOP	23-28	/RT-L, EP, AFW-L, ACP-ST

Table 2b. Definitions of fault trees listed in Table 2a[1]

ACP-ST	OFFSITE POWER RECOVERY IN SHORT TERM
AFW-L	NO OR INSUFFICIENT AUXILIARY/EMERGENCY FEEDWATER FLOW
EP	EMERGENCY POWER SYSTEM FAILS
RT-L	REACTOR FAILS TO TRIP DURING LOSS OF OFFSITE POWER

Note:
1. Modifications to other fault trees not listed in this table were made in accordance with guidance provided in Reference 9. The SPAR model was modified to replace the existing reactor coolant pump (RCP) seal LOCA model with the Rhodes Model (Ref. 8). In order to replace the RCP seal LOCA model without modifying the station blackout event tree, top event OP-SL was set to “False” (basic event OEP-XHE-NOREC-SL). To account for offsite power recovery, the nonrecovery probabilities for offsite power AND emergency diesel generators (EDGs) were added to the sequence-specific nonrecovery probabilities for the RCP seal LOCA sequences in the station blackout event tree (see Table 5). Based on the Rhodes Model, the time available to prevent core damage by high-pressure injection if RCP seals fail is 4 hours. Therefore, the nonrecovery probabilities for EDGs and offsite power were modified to reflect the 4-hour recovery time to avert core damage (see Table 5). Finally, Event Tree Linking Rule Nos. 4 and 5 (Ref. 6, Table 2-1), which are triggered by the success of top event OP-SL, were negated by substituting fault tree HPI for HPI-L in LOOP Sequences 23-11 and 23-23 and HPR for HPR-L in LOOP Sequences 23-06, 23-09, 23-18, and 23-21. High temperature seals were assumed to be installed on all RCPs.

Table 3. Conditional cut sets for Sequence 23-28

CCDP	Percent contribution	Minimal cut sets[1]	
Event Tree: LOOP, Sequence 23-28			
4.8E-006	77.3	EPS-DGN-FC-DGA OEP-XHE-NOREC-ST	EPS-DGN-FC-DGB LOOP-23-28-NREC
1.4E-006	22.5	EPS-DGN-CF-AB LOOP-23-28-NREC	OEP-XHE-NOREC-ST
6.3E-006	**Total[2]**		

Notes:
1. See Table 4 for definitions and probabilities for the basic events.
2. Total CCDP includes all cut sets (including those not shown in this table).

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

Event name	Description	Probability/ Frequency	Modified
AFW-TDP-FC-TDP	AFW TURBINE-DRIVEN PUMP FAILURE	TRUE	YES[1]
EPS-DGN-CF-AB	COMMON CAUSE FAILURE OF DIESEL GENERATORS	8.8E-004	YES[2]
EPS-DGN-FC-DGA	DIESEL GENERATOR A FAILS	5.5E-002	YES[3]
EPS-DGN-FC-DGB	DIESEL GENERATOR B FAILS	5.5E-002	YES[3]
IE-LOOP	LOSS OF OFFSITE POWER (LOOP) INITIATING EVENT	1.0E-05/hr	YES[4]
IE-SGTR	STEAM GENERATOR TUBE RUPTURE (SGTR) INITIATING EVENT	8.0E-07/hr	YES[5]
IE-SLOCA	SMALL LOSS OF COOLANT ACCIDENT INITIATING EVENT	3.4E-07/hr	YES[5]
IE-TRAN	TRANSIENT (TRANS) INITIATING EVENT	1.6E-04/hr	YES[5]
LOOP-22-NREC	LOOP SEQUENCE 22 NONRECOVERY PROBABILITY	8.4E-001	YES[6]
LOOP-23-06-NREC	LOOP SEQUENCE 23-06 NONRECOVERY PROBABILITY	5.0E-002	YES[7]
LOOP-23-09-NREC	LOOP SEQUENCE 23-09 NONRECOVERY PROBABILITY	5.0E-002	YES[7]
LOOP-23-11-NREC	LOOP SEQUENCE 23-11 NONRECOVERY PROBABILITY	5.0E-002	YES[7]
LOOP-23-18-NREC	LOOP SEQUENCE 23-18 NONRECOVERY PROBABILITY	5.0E-002	YES[7]
LOOP-23-21-NREC	LOOP SEQUENCE 23-21 NONRECOVERY PROBABILITY	5.0E-002	YES[7]
LOOP-23-23-NREC	LOOP SEQUENCE 23-23 NONRECOVERY PROBABILITY	5.0E-001	YES[7]
LOOP-23-28-NREC	LOOP SEQUENCE 23-28 NONRECOVERY PROBABILITY	8.0E-001	YES[6]
OEP-XHE-NOREC-SL	OPERATOR FAILS TO RECOVER OFFSITE POWER BEFORE REACTOR COOLANT PUMP (RCP) SEAL LOCA	FALSE	YES[8]
OEP-XHE-NOREC-ST	OPERATOR FAILS TO RECOVER OFFSITE POWER IN SHORT TERM	2.0E-001	YES[9]
RCS-MDP-LK-SEALS	RCP SEALS FAIL W/O COOLING AND INJECTION	2.2E-001	YES[8]

Notes:
1. Basic event was changed to reflect condition being analyzed. TRUE has a failure probability of 1.0.
2. Base case model was updated using data from NUREG/CR-5497, Tables 5-2 and 5-5 (Ref. 10). Updated value uses an 8-hour mission time for the diesel generator, which is the 95% probability of recovering offsite power for the weighted average of all LOOP events (Ref. 6, Table 6.1).
3. Base case model was updated using data from NUREG/CR-5500, Vol. 5, Tables C4, C6, and C7 (Ref. 11). See note 2 for additional information.
4. Base case model was updated using data from NUREG/CR-5750, Table H3 (Ref. 12) and NUREG/CR 5496 Table B4 (Ref. 13).
5. Base case model was updated using data from NUREG/CR-5750, Table 3-1 (Ref. 12).
6. Basic event was changed to reflect condition being analyzed. Sequence nonrecovery probabilities were modified to reflect the nonrecovery of AFW; see Table 5.
7. Base case model was updated. See Table 5 for basis.
8. Base case model was updated to reflect the Rhodes Model. (See foot note to Table 2b.)
9. Base case model was updated to reflect the nonrecovery of offsite power within 1 hour; from SPAR 2QA model, Table 6-1 (Ref. 6). For the condition assessment evaluated in this event, TDAFW pump unavailable, the dominating core damage sequence is an SBO with no auxiliary feedwater. For this sequence, core uncoverying is estimated to occur in approximately 1.7 hours (Ref. 7, Table 7.1, 6200 sec). The actual time for recovering offsite power is reduced 30 minutes to 1.2 hours, to allow sufficient time for the operator to perform the necessary system recovery actions. The probability of not recovering offsite power, for the weighted average of all types of LOOPs, within 1 hour is 0.2 (Ref. 6, Figure 6-1). Therefore, OEP-XHE-NOREC-ST is set to 0.2.

Table 5. Basis for the probabilities of sequence-specific recovery actions

Seq. no. and basic event	Failed systems and recovery time[1,2]	Nonrecovery probability	Combined failure probability	Modification remarks (also see footnotes)
22 LOOP-22-NREC	AFW-L F&B-L	1.0 0.84	0.84	TDAFW pump is nonrecoverable
23-28 LOOP-23-28-NREC	EP AFW-L ACP-ST	0.8 1.0 n/a	0.8	TDAFW pump is nonrecoverable
23-06 LOOP-23-06-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model
23-09 LOOP-23-09-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model
23-11 LOOP-23-11-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model
23-18 LOOP-23-18-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model
23-21 LOOP-23-21-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model
23-23 LOOP-23-23-NREC	EDG (4 hours) Offsite Power (4 hours)	0.5 0.1	0.05	Include Rhodes RCP seal LOCA model

Notes:

1. Based on the SPAR model (Ref. 6), nonrecovery probability for an EDG is $exp(-0.173t)$, where t is recovery time in hours. When multiple EDGs are failed, only one EDG is considered for recovery, since operators would attempt to recover only one EDG.
2. Recovery times used in the SPAR model are as follows:
   - 1 hour--core uncovery due to loss of heat removal during a station blackout (Ref. 7, Table 7.1)
   - 4 hours--core uncovery due to RCP seal LOCA (update based on Rhodes Model, Reference 9)

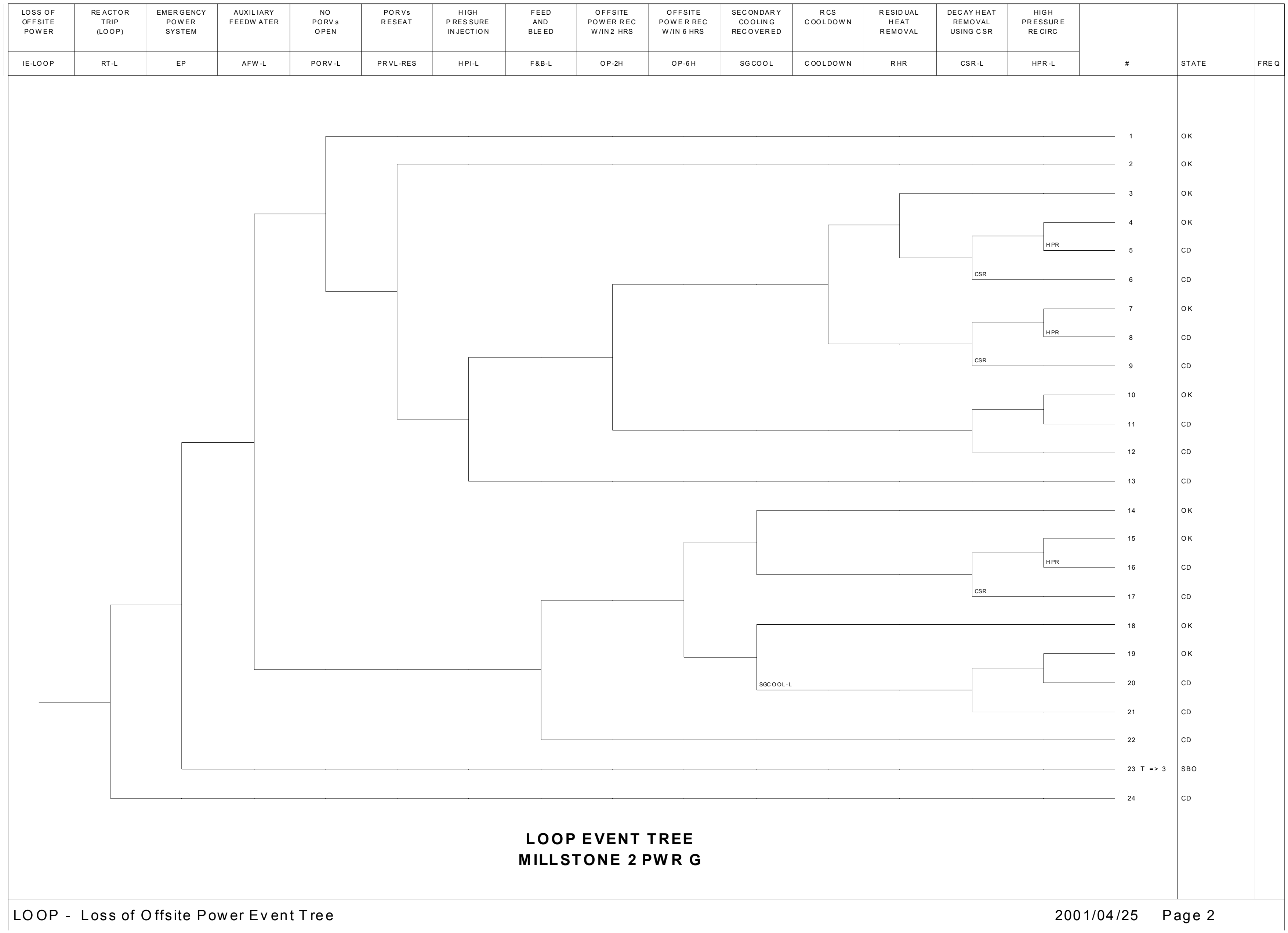

Figure 1 Millstone 2 loss of offsite power event tree showing sequence 23 transfer to station blackout event tree (Fig. 2)

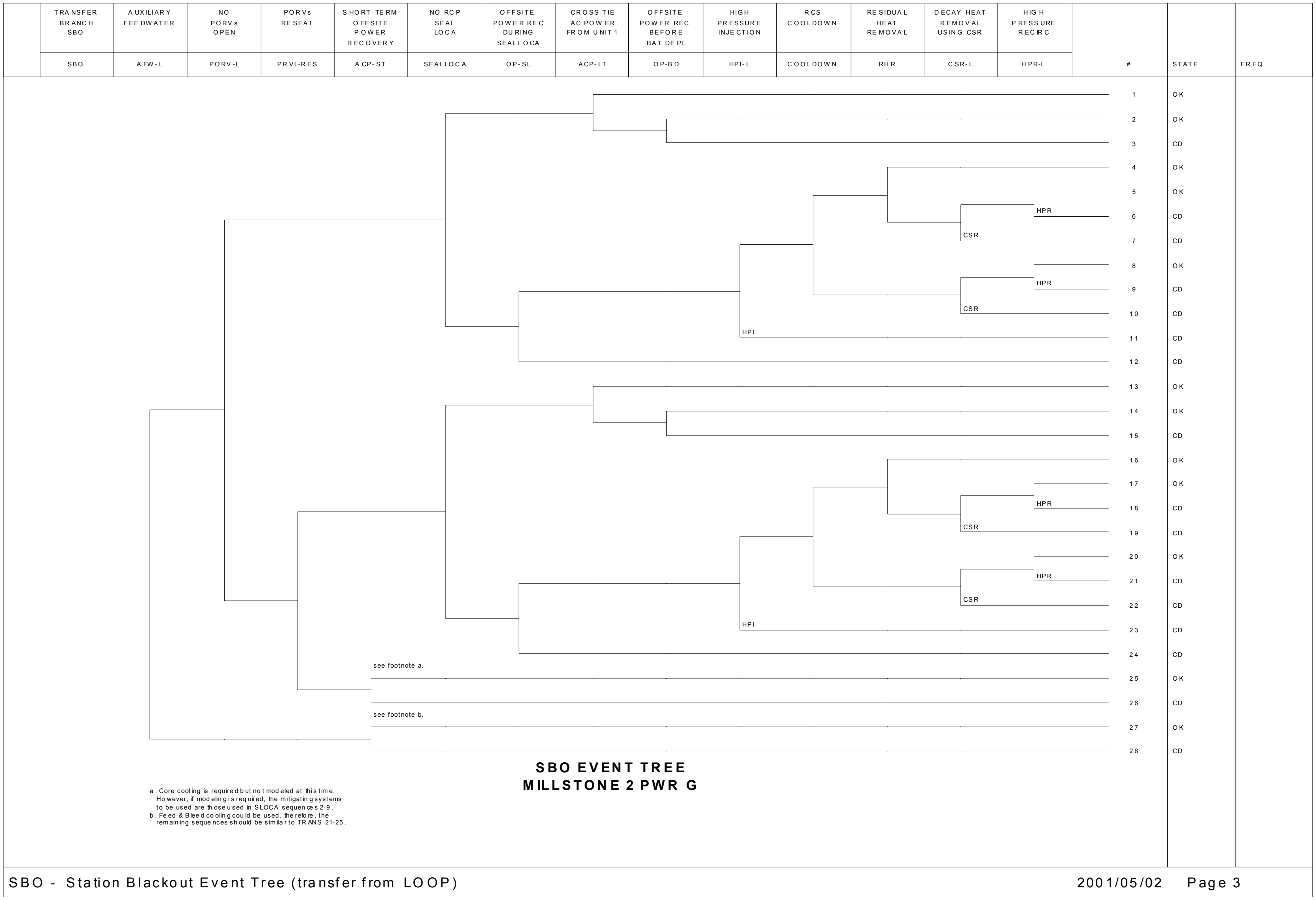

Figure 2 Millstone 2 station blackout event tree showing sequence 28

GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an event or condition 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 (LOSP) 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 other pertinent reports, such as the licensee event report (LER) and/or NRC inspection reports.

Modeling Techniques

The models used for the analysis of events were developed by the Idaho National Engineering and Environmental Laboratory (INEEL). The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The developed models are called Standardized Plant Analysis Risk (SPAR) models. The SPAR models are based on linked fault trees. Fault trees were developed for each top event on the event trees to a super component level of detail.

SPAR Version 2 models have four types of initiating events: (1) transients, (2) small loss-of-coolant accidents (LOCAs), (3) steam generator tube rupture (PWR only), and (4) loss of offsite power (LOSP). The only support system modeled in Version 2 is the electric power system. The SPAR models have transfer events trees for station blackout and anticipated transient without scram.

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 Summary" section:
  - accurately describe the event as it occurred; and
  - provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?

ENCLOSURE 2

- Does the "Modeling Assumptions" section:

  – accurately describe the modeling done for the event;

  – accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions; and

  – includes assumptions regarding the likelihood of equipment recovery?

Appendix G 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 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 simulation).

This documentation must be the revision or cover the practices at the time of the event occurrence. 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).

An Example of a Recovery Measure Evaluation

A pressurized-water reactor plant experiences a reactor trip. 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 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, and

- 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 event or condition:

- Preliminary ASP analysis.

- Specific LER, NRC inspection report, or other pertinent reports for each preliminary ASP analysis.

Schedule

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

Reference

1. R. J. Belles et al., “Precursors to Potential Severe Core Damage Accidents: 1997, A Status Report,” USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volume 26, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, and Science Applications International Corp., Oak Ridge, Tennessee, November 1998.

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