50-237

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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July 20, 1995

Mr. D. L. Farrar
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SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF PRELIMINARY ASP ANALYSIS OF OPERATIONAL CONDITION AT DRESDEN, UNIT 2

Dear Mr. Farrar:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational condition which occurred at Dresden, Unit 2, on June 8, 1994 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 237/94-018, Revision 1. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORNL). The results of this preliminary analysis indicate that this condition may be a precursor in the 1994 Annual Precursor Report. 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 to incorporate your comments and meet our schedule for issuance of the 1994 Precursor Report, you are requested to complete your review and to provide any comments within 30 days of receipt of this letter.

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 information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 237/94-018, Revision 1, which documented the event.

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The final resolution of each licensee's comment on the preliminary ASP analyses will be documented in a separate appendix of the 1994 Precursor Report, NUREG/CR-4674. Dresden, Unit 2, is on the distribution list for NUREG/CR-4674. 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.

If you have any questions regarding this request, please contact me at (301) 415-1345.

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-237

Enclosures: 1. ASP Analysis
2. Guidance
3. LER No. 237/94-018, Rev. 1

cc w/encls: see next page

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D. L. Farrar
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A.1 LER No. 237/94-018 Rev. 1

Event Description: Motor Control Center Trips Due to Improper Breaker Settings

Date of Event: June 8, 1994

Plant: Dresden 2

A.1.1 Summary

Following an unexpected trip of a motor control center (MCC) at Dresden 3 during surveillance testing, three MCCs were identified at Dresden 2 and Dresden 3 with improperly set feeder breakers. A review of MCC loading indicated that load additions since the original settings were determined had created an overload situation. For two of the MCCs, the overload condition would only have existed if an emergency diesel generator (EDG) was running following a reactor trip with offsite power available. Load shedding following a loss of offsite power (LOOP) would have precluded an overload condition for this initiating event. For one of the MCCs, the overload condition would also have existed following a LOOP.

The conditional core damage probability estimated for the event is 6.1×10^{-6}. The relative significance of this event compared to other postulated events at Dresden is shown below in Figure A.1.1 (to be provided in the final report).

A.1.2 Event Description

On June 8, 1994, Dresden Unit 2 was operating at 99% power, and Unit 3 was in refueling. The Unit 2/3 standby gas treatment (SBGT) system was in operation, and a 24 h endurance run for the EDG 3 was in progress, as was a Unit 2 high-pressure coolant injection (HPCI) surveillance.

Shortly after starting the Unit 2 HPCI auxiliary oil pump, MCC 39-2 tripped. As a result of the loss of power at MCC 39-2 (1) EDG 3 tripped on high temperature following loss of power to its cooling water pump and ventilation fan, (2) the 125-V dc and 250-V dc battery systems had to be realigned to alternate chargers, (3) a half-scram was generated as a result of loss of power to the RPS motor-generator, and (4) SBGT train A auto-started following loss of power to train B components.

MCC 39-2 loads were stripped, and the MCC feeder breaker was reclosed. MCC 39-2 loads were reenergized within 30 min of the breaker trip.

The trip of MCC 39-2 was caused by an incorrectly set feeder breaker. The feeder breaker for the MCC had a General Electric dashpot type EC-2A overcurrent trip device which was original equipment. The setting for this breaker was 400 A. A review of the original loading on the MCC indicated that the 400 A setting was adequate, but load additions made to the MCC over time had increased the available running load current above the 400 A setting.

Two other breakers were subsequently identified with similar problems—MCC 28-3 and 38-3. The EC-2A trip devices for both of these MCCs had been replaced with newer General Electric solid state type RMS-9 trip devices. Both of these MCCs were also set to trip at 400 A. The licensee noted in the LER that the setting for MCC 38-3 was chosen to be identical with the original breaker setting based on the assumption that MCC loading had not changed over time. However, since the loading had changed, the total connected load was greater than the protective device setting. At the time of the MCC 28-3 trip device replacement, it was recognized that the overcurrent setting was lower than the total connected load. However, it was assumed that the running load during accident conditions would be within the setting of the protective device.

Based on the loads associated with each MCC, the licensee concluded that MCCs 38-3 and 39-2 could be overloaded and trip during a safety actuation in which the associated EDG was running (e.g., for testing or following a spurious start) while offsite power was still available. For these MCCs, loads shed following a LOOP would preclude an overload condition. For MCC 28-3, however, the overload condition could exist for both LOOPs and other events in which the associated EDG was running.

A.1.3 Additional Event-Related Information

Three EDGs provide emergency power to the two Dresden units: EDG 2 provides power to Unit 2 bus 24-1, EDG 3 provides power to Unit 3 bus 34-1, and swing EDG 2/3 provides power to either Unit 2 bus 23-1 or Unit 3 bus 33-1 in the event of a LOOP on Unit 2 or Unit 3, respectively. In the event of a dual-unit LOOP with a loss of coolant accident (LOCA) on one unit, EDG 2/3 provides power to the unit with the LOCA. In the event of a dual-unit LOOP without a LOCA, EDG 2/3 powers the unit that suffers the LOOP first. Unit 2 bus 24-1 and Unit 3 bus 34-1 can be cross-tied by closing two normally open breakers.

Two 250 V-dc and two 125 V-dc batteries are shared between both units. The 250 V-dc batteries primarily power large loads, such as dc-powered pumps and valves, while the 125 V-dc batteries provide control power to components such as circuit breakers. Battery chargers that normally supply dc power and provide battery charging can be powered from buses associated with EDG 2 (Unit 2) or EDG 3 (Unit 3) or the swing EDG. Each battery is sized to power its respective loads for 4h.

A.1.4 Modeling Assumptions

Four possible situations were addressed in the analysis of this event. All three MCCs could have tripped following an initiating event in which emergency core cooling system (ECCS) actuation was required, offsite power was available, and the EDG associated with the MCC was running (e.g., for testing or following a spurious start). Analysis Case 1a addresses the situation in which one EDG was running. Analysis Case 1b addresses the situation in which two EDGs were running. In addition, MCC 28-3 could have tripped following a LOOP. Analysis Cases 2a and 2b consider a plant-centered LOOP at Unit 2 and dual-unit LOOPs at Units 2 and 3. In all cases, the MCCs were assumed to trip if they could have tripped. This assumption may be conservative.

Case 1a. Postulated initiating event with offsite power available and one EDG running. This situation could exist if a transient or small-break LOCA occurred and one of the two EDGs associated with a unit was undergoing monthly surveillance testing. The greatest potential impact is associated with MCCs 39-2 and 38-3 at Unit 3. These MCCs, in addition to supplying power to EDG components (and turning gear components for MCC 38-3), also supply power to containment cooling service water (CCSW) cubicle fans. CCSW provides decay heat removal for the containment cooling mode of low-pressure coolant injection. The analysis assumed the two CCSW trains associated with the running EDG would be unavailable after the MCC tripped. The probability of a running EDG was estimated to be 2.8×10^{-3}, based on an assumed 1-h surveillance run-time for each EDG per month.

The significance for this case was estimated by setting basic events associated with the two impacted CCSW trains to true (failed) and calculating the increase in core damage probability for non-LOOP (transient and small-break LOCA) initiating events over a 1-year period using the IRRAS-based ASP model for Dresden. Long-term unavailabilities such as this event have typically been modeled in the ASP program for a 1-year period, assuming the plant was at power 70% of the time; this is equal to 6132 h (365 d × 24 h/d × 0.7). The increase in core damage probability was multiplied by the probability that an EDG would be running to estimate the conditional probability for Case 1a. This conditional probability is less than 1.0×10^{-8}. Since this is substantially below the 1.0×10^{-6} documentation limit used in the ASP program, the calculational results are not included herein.

Case 1b. Postulated initiating event with offsite power available and two EDGs running. This situation could exist if a transient or a small-break LOCA occurred and both EDGs associated with a unit were spuriously started. The analysis for this case is similar to Case 1a, except all trains of CCSW were assumed to be unavailable. The probability of spurious EDG start was estimated using a Sequence Coding and Search System search of BWR automatic or manual reactor trips with spurious EDG starts. Three such events were identified in 573 trips from power, resulting in an estimated probability of spurious EDG actuation of 5.2×10^{-3}. The resulting conditional core damage probability is estimated to be 4.3×10^{-8}, also well below 1.0×10^{-6}. As for Case 1a, the calculational results are not included herein.

Case 2a. Postulated plant-centered LOOP at Unit 2. For a postulated plant-centered LOOP at Unit 2 only, offsite power remains available at Unit 3. Trip of MCC 28-3 will result in inoperability of swing EDG 2/3 and unavailability of power to 4-kV bus 23-1. Power can be recovered to bus 24-1 if EDG 2 fails by recovering offsite power or by closing the cross-tie from Unit 3 bus 34-1. Because of the shared dc system at Dresden, dc power will remain available for instrumentation even if Unit 2 batteries are depleted. Therefore, a sequence involving SRV reseat and isolation condenser or HPCI success following a postulated station blackout will not proceed to core damage (essentially all of sequence 44).

The probability of failing to recover power to bus 24-1 through closure of the cross-tie breakers from Unit 3 was assumed to be 0.12 (ASP nonrecovery class R3, see Appendix A, Sect. A.1 to the 1992 precursor report, NUREG/CR-4674, Vol. 17). This value was chosen because recovery appeared possible in the required time from the control room, but was not considered routine (the value chosen for this failure probability for this case is considered a bounding probability and does not substantially impact the overall analysis results). This value is used in lieu of the failure probability for EDG 3 in the IRRAS-based ASP models to reflect the failure to provide power from bus 34-1. The probability of EDG common-cause failure was set to false to reflect the unavailability of EDG 2/3 and the availability of power on bus 34-1.

After elimination of sequence 44 (since it does not proceed to core damage for a single-unit plant-centered LOOP) a conditional core damage probability of 1.6×10^{-8} is estimated. As for Cases 1a and 1b, the calculational results are not included herein.

Case 2b. Dual-unit LOOP at Units 2 and 3. For a postulated dual-unit LOOP (primarily grid- and weather-related LOOPs), offsite power is unavailable to both units. If the LOOP occurs at Unit 2 first, trip of MCC 28-3 will result in unavailability of swing EDG 2/3. EDG 3 will be required to power Unit 3 loads, leaving only EDG 2 to supply power to Unit 2 loads.

The frequency of a dual-unit LOOP and the probability of failing to recover offsite power in the short-term and before battery depletion were estimated to be 1.7 × 10^{-2}/year, 0.66, and 0.21, respectively, based on models described in *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. These models are based on the results of data distributions contained in *Evaluation of Station Blackout at Nuclear Power Plants*, NUREG-1032. The probability of the dual-unit LOOP occurring first at Unit 2 was assumed to be 0.5. The failure probability for EDG 2/3 was set to true to reflect its unavailability following trip of MCC 28-3. The common-cause failure probability for the EDGs was revised to 4.4 × 10^{-3} to reflect the unavailability of EDG 2/3.

A.1.5 Analysis Results

The conditional core damage probability estimated for this event is 6.1 × 10^{-6}. The dominant core damage sequence, highlighted on the event tree in Figure A.1.3 involves a postulated dual-unit LOOP (primarily grid- or weather-related) with subsequent failure of all three Dresden EDGs and failure to recover offsite power prior to battery depletion. In the dominant sequence, EDG 2/3 fails due to MCC 28-3 trip following its alignment to Unit 2 (the postulated dual-unit LOOP affects Unit 2 first), and EDG 2 and 3 fail for unspecified reasons (random or common-cause failures).

The calculational results for Cases 1a, 1b, and 2a were not included since they do not provide a significant contribution to the conditional core damage probability for the event. The calculational results for Case 2b are shown in Tables A.1.1 through A.1.5. Definitions and probabilities for basic events are shown in Table A.1.1. The conditional probabilities associated with the highest probability sequences are shown in Table A.1.2. Table A.1.3 lists the sequence logic associated with the sequences listed in Table A.1.2. Table A.1.4 describes the system names associated with the dominant sequences. Cutsets associated with each sequence are shown in Table A.1.5.

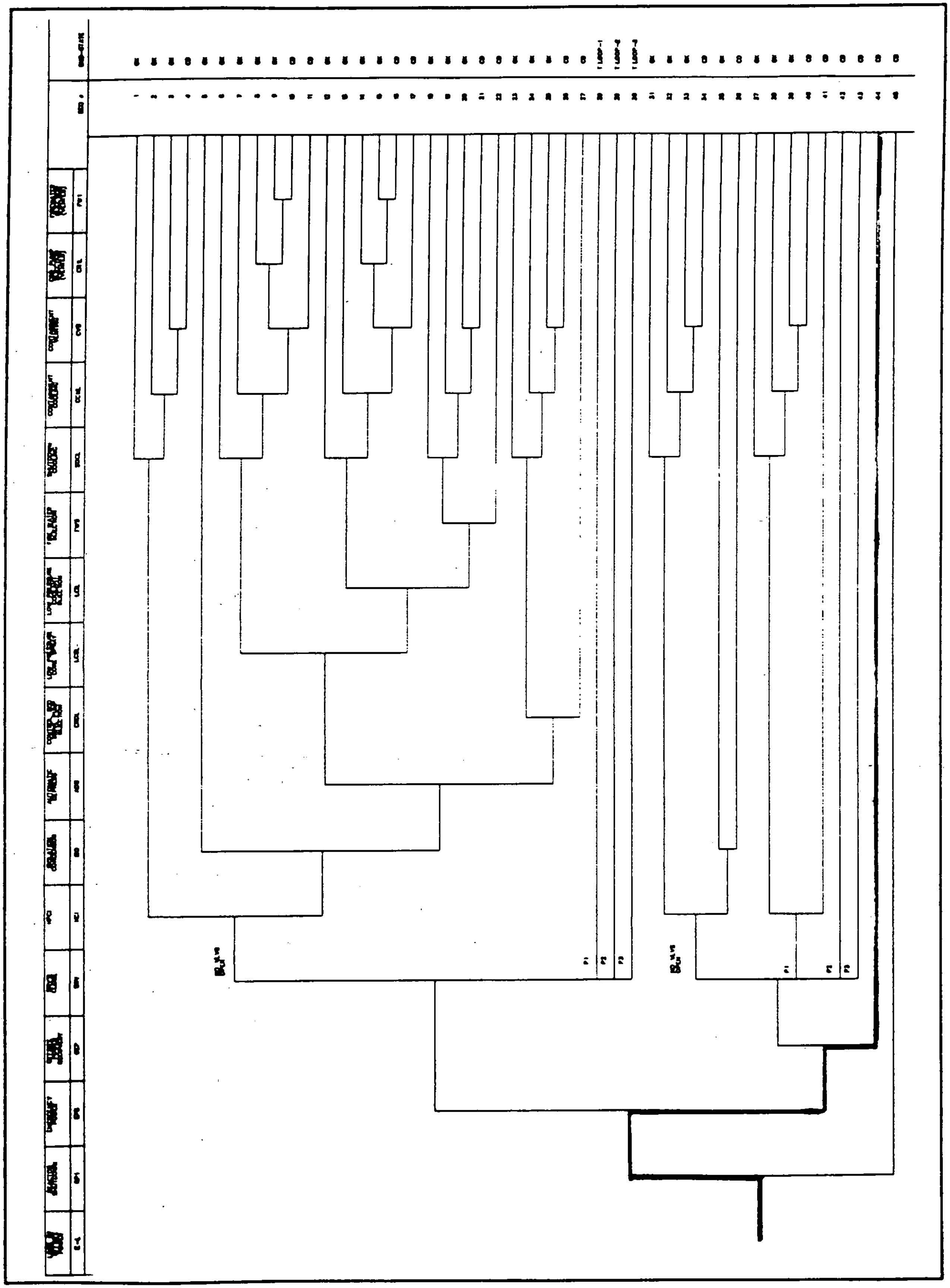

Figure A.1.2 Dominant core damage sequence for LER 237/94-018.

Table A.1.1. Definitions and probabilities for selected basic events for LER 237/94-018					
Event name	Description	Base probability	Current probability	Type	Modified for this event?
EPS-DGN-CF-DGNS	COMMON CAUSE FAILURE OF DIESEL GENERATORS	1.2E-003	4.4E-003		Y
EPS-DGN-FC-DG2	UNIT 2 GENERATOR FAILS	4.4E-002	4.4E-002		N
EPS-DGN-FC-DG3	UNIT 3 DIESEL GENERATOR FAULRE	4.4E-002	4.4E-002		N
EPS-DGN-FC-DG23	SWING DIESEL GENERATOR FAILS	4.4E-002	1.0E+000	TRUE	Y
EPS-XHE-XE-NOREC	OPERATOR FAILS TO RECOVER EMERGENCY POWER	8.0E-001	8.0E-001		N
IE-LOOP	LOSS OF OFFSITE POWER INITIATOR	9.1E-007	5.6E-003		Y
IE-SLOCA	SMALL LOCA INITIATOR	1.7E-006	0.0E+000	IGNORE	Y
IE-TRAN	TRANSIENT INITIATOR	3.4E-004	0.0E+000	IGNORE	Y
OEP-XHE-XE-NOREC	OPERATOR FAILS TO RECOVER OFFSITE POWER	2.1E-001	2.1E-001		N

Table A.1.2. Sequence conditional probabilities for LER 237/94-018					
Event tree name	Sequence name	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP-CDP)	% Contribution
LOOP	44	5.9E-006	3.5E-006	2.3E-006	96.7
Total (All Sequences)		6.1E-006			

Table A.1.3. Sequence logic for dominant sequences for LER 237/94-018		
Event tree name	Sequence name	Logic
LOOP	44	/RP1, EPS, OEP

Table A.1.4. System names for LER 237/94-018	
System name	Description
EPS	EMERGENCY POWER SYSTEM FAILS
OEP	OFFSITE POWER RECOVERY
RP1	REACTOR SHUTDOWN FAILS

Table A.1.5. Conditional cut sets for higher probability sequences for LER 237/94-018			
Cutset No.	% Contribution	Frequency	Cut sets
LOOP Sequence: 44		6.0E-006	
1	69.5	4.1E-006	EPS-DGN-CF-DGNS, EPS-XHE-XE-NOREC, OEP-XHE-XE-NOREC
2	30.6	1.8E-006	EPS-XHE-XE-NOREC, OEP-XHE-XE-NOREC, EPS-DGN-FC-DG2, EPS-DGN-FC-DG3

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 1994 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 initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) loss 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 E 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 components 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 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,

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*Revision or practices at the time the event occurred.

- the effects of using the standby feedwater system have 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 calculational 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 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: 1993. A Status Report*. USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Volumes 19 and 20), Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory and Science Applications International Corp., September 1994.