50-315

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

September 18, 1996

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF AN EVENT AT D.C. COOK

Dear Mr. Fitzpatrick:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at D.C. Cook on September 12, 1995, (Enclosure 1), and was reported in Licensee Event Report (LER) No. 315/95-011. 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 1995. 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 information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 315/95-011, which documented the event.

Please contact me at (301) 415-3017 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:

John B. Hickman Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Preliminary ASP Analysis
2. Review Guidance
3. LER No. 315/95-011

cc w/encl: See next page

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# LER No. 315/95-011

Event Description: One Safety Injection pump unavailable for six months

Date of Event: September 12, 1995

Plant: D. C. Cook, Unit 1

## Event Summary

During a surveillance test with the unit shutdown in Mode 6, personnel determined that the West Centrifugal Charging Pump (CCP) had been inoperable for about six months. This was the result of an incorrect breaker relay calibration performed six months earlier. The unavailability of the West CCP primarily affects the units' response to steam generator tube rupture (SGTR) and small-break loss of coolant accident (SLOCA) events. The estimated *increase* in core damage probability for this event is $7.7 \times 10^{-6}$, over a nominal value for the same period of $2.9 \times 10^{-5}$.

## Event Description

On September 12, 1995, the plant was shutdown in Mode 6 when personnel started the West CCP in order to perform the "ECCS Full Flow Test" surveillance. The West CCP provides injection flow on the receipt of a Safety Injection (SI) signal. After operating at full flow for seven minutes, the pump tripped on motor overcurrent. Personnel determined that the pump tripped because the setting of the 1-51-TA8 time overcurrent relay was incorrectly set. It was determined that this relay was last calibrated on March 15, 1995 (180 days prior to the full flow test). This effectively rendered the West CCP inoperable for the preceding six months.

During the review of this event, the Instrumentation and Control (I&C) technicians involved in calibrating the relays demonstrated how they typically determine the relay pickup current. Because their technique was incorrect, the relays were miscalibrated. Both I&C technicians involved with the relay calibration were

trained and qualified within the D. C. Cook Nuclear Plant relay training program. However, it was determined that a significant amount of time elapsed between the end of the training program and the time that the 1-51-TA8 time overcurrent relay associated with the West CCP breaker was incorrectly calibrated.

## Additional Event-Related Information

During normal plant operation, both charging pumps (East and West) are configured for their charging function. One charging pump is sufficient to supply full charging flow and reactor coolant pump seal injection during normal leakage and normal letdown conditions. A third positive displacement charging pump is available, but is not normally used. On receipt of a valid SI signal, the CCPs operate in the high pressure injection mode.

D. C. Cook also has a separate SI system. This system, with two pumps operating in parallel, operates in an intermediate pressure injection mode. The two SI pumps deliver flow from the Refueling Water Storage Tank (RWST) at a maximum injection pressure of approximately 1100 psig. The residual heat removal (RHR) pumps can be aligned for recirculation from the containment sump to the suction of either the SI pumps or the CCPs.

The licensee indicated that the East CCP had been inoperable for less than 18 h during the six-month period that the West CCP was unavailable. Additionally, the emergency diesel generator (EDG) supporting the East CCP was unavailable for less than 50 hours during the six-month period that the West CCP was unavailable.

## Modeling Assumptions

The CCPs were subject to common cause failure during this six-month period based on incorrect maintenance practices. Because the success criteria in the IRRAS model assumes both CCPs are required for success of the CCP portion of the high pressure injection function in response to either a SLOCA or a SGTR, no further changes were required to model the increased potential for common cause failure. Success of one of the two SI pumps also ensures success of the HPI function in the IRRAS model, independent of the success of the

CCPs. This is not as stringent as the assumption of the plant Individual Plant Examination that one of two CCPs and one of two SI pumps are required in response to a SLOCA.

The IRRAS response to a SGTR was modified. Previously, a loss of the high pressure injection function lead directly to core damage. The possibility of lowering RCS pressure below the Steam Generator safety valve setpoint within 30 minutes was allowed following the loss of high pressure injection capability. Based on the operator burden under a short time constraint, a failure probability of 0.1 was assigned to the new basic event, PCS-XHE-DEPRES.

Loss of offsite power (LOOP) sequences were prominent in Case 2 with only one EDG available. LOOP probabilities 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.

## Analysis Results

Determining the overall increase in the core damage probability involved determining the increase in the core damage probability for three different cases and then summing these cases together. The three cases are:

- Case 1 — the increase in the core damage probability due to the long term unavailability of the West CCP (4252 h).
- Case 2 — the increase in the core damage probability due to the unavailability of the opposite train EDG that was periodically unavailable while the West CCP was unavailable (50 h).
- Case 3 — the increase in the core damage probability due to the time that both CCPs were simultaneously unavailable because of various maintenance activities (18 h).

Combining the probability estimates for the three cases results in an overall *increase* in the core damage probability for the 180 day period of $7.7 \times 10^{-6}$. Most of the increase (56%) is driven by the long term

unavailability of the West CCP (Case 1). An additional 44% of the increase in core damage probability was added by Case 2. The dominant core damage sequence, highlighted as sequence number 6 on the event tree in Figure 1, contributes approximately 44% to the combined increase in the core damage probability estimate for all three modeled cases. Sequence number 6 involves:

- a SLOCA,
- the successful trip of the reactor,
- the successful operation of the Auxiliary Feedwater (AFW) system, and
- the failure of the High Pressure Injection (HPI) system to provide sufficient cooling flow.

The next most dominant sequence involved a LOOP and contributed approximately 13% to the combined increase in the core damage probability estimate for all three modeled cases.

The nominal core damage probability over a six-month period estimated using the Accident Sequence Precursor (ASP) models for D. C. Cook is approximately $2.9 \times 10^{-5}$. The failed West CCP increased this probability by 28% to $3.7 \times 10^{-5}$. The latter value ($3.7 \times 10^{-5}$) is the conditional core damage probability (CCDP) for the six-month period in which the West CCP was inoperable.

For most ASP analyses of conditions (equipment failures over a period of time during which postulated initiating events could have occurred), sequences and cut sets associated with the observed failure dominate the CCDP (i.e., the probability of core damage over the unavailability period, given the observed failures). The increase in core damage probability because of the failures is, therefore, essentially the same as the CCDP, and the CCDP can be considered a reasonable measure of the significance of the observed failures. However, for this event, sequences unrelated to the failure of the West CCP dominated the CCDP estimate. The increase in core damage probability given the West CCP inoperability, $7.7 \times 10^{-6}$, is, therefore, a better measure of the significance of the failure of the West CCP.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences for the condition assessment are shown in Table 2. The sequence logic associated with the sequences listed in Table 2 are given in Table 3. Table 4 describes the

system names associated with the dominant sequences for the condition assessment. Minimal cut sets associated with the dominant sequences for the condition assessment are shown in Table 5.

## Acronyms

| | |
|---|---|
| AFW | Auxiliary Feedwater |
| ASP | Accident Sequence Precursor |
| CCDP | Conditional Core Damage Probability |
| CCP | Centrifugal Charging Pump |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| HPI | High Pressure Injection |
| HPR | High Pressure Recirculation |
| IRRAS | Integrated Reliability and Risk Analysis System |
| I&C | Instrumentation and Control |
| LOCA | Loss-of-Coolant Accident |
| LOOP | Loss of Offsite Power |
| MOV | Motor-Operated Valve |
| PRA | Probabilistic Risk Assessment |
| RCP | Reactor Coolant Pump |
| RHR | Residual Heat Removal |
| RWST | Refueling Water Storage Tank |
| SGTR | Steam Generator Tube Rupture |
| SI | Safety Injection |
| SLOCA | Small-Break Loss-of-Coolant Accident |

## References

1. LER 315/95/011, Rev 0, "West Centrifugal Charging Pump Inoperable Due to Inability to Meet Design Basis Requirements for Six Months as a Result of Personnel Error During Relay Calibration," November 20, 1995.
2. Indiana Michigan Power Company, *Donald C. Cook Nuclear Plant Individual Plant Examination Summary Report.*
3. Indiana Michigan Power Company, *Donald C. Cook Nuclear Plant Final Safety Analysis Report.*

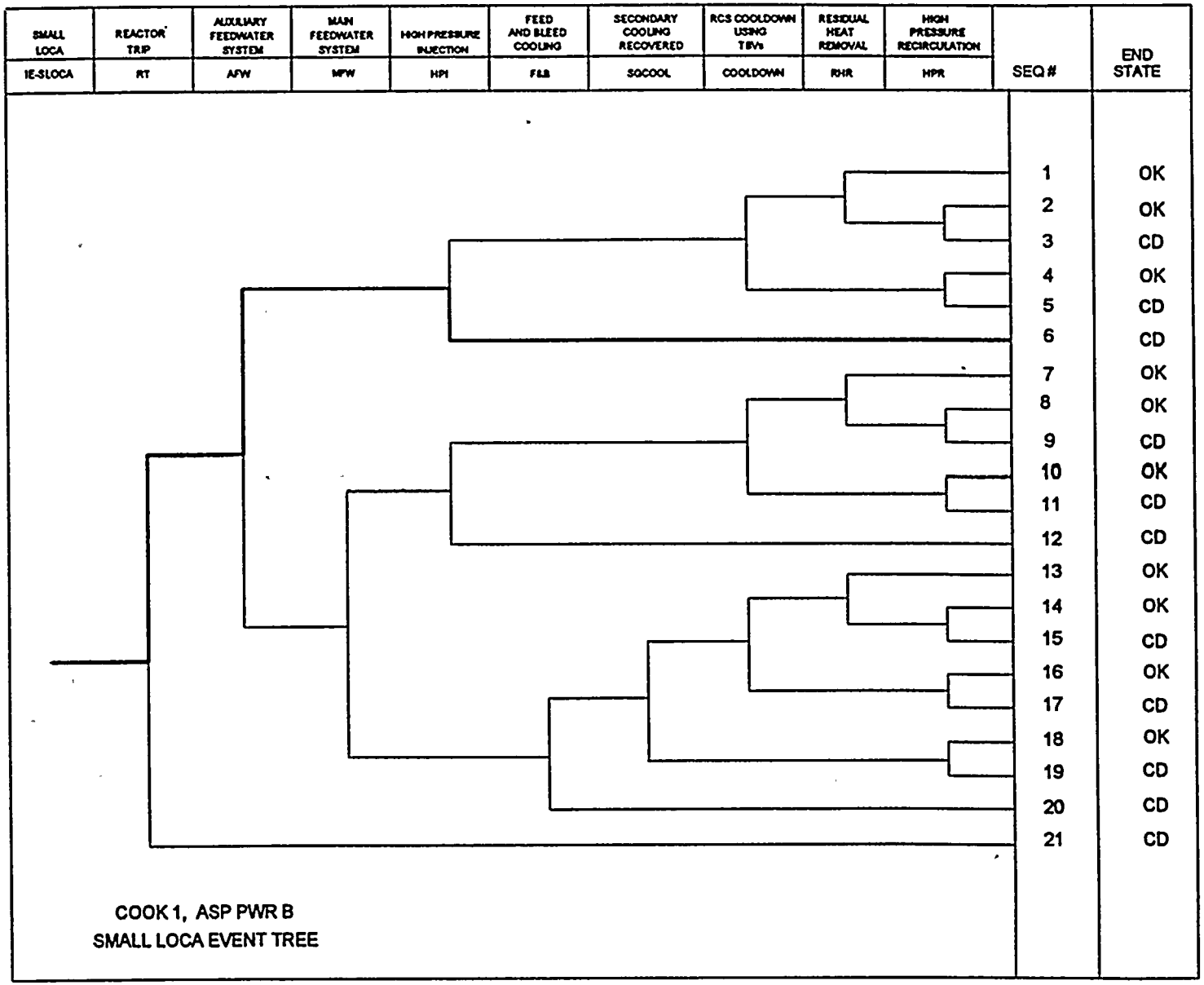


Fig. 1 Dominant core damage sequences for LER no. 315/95-011.

**Table 1. Definitions and probabilities for selected basic events for LER No. 315/95-011**

| Event name | Description | Base probability | Current probability | Type | Modified for this event |
|---|---|---|---|---|---|
| CVC-MDP-FC-1A | Failure of Charging Pump A | 9.0 E-004 | 1.0 E+000 | TRUE | Yes |
| HPI-MDP-CF-ALL | HPI Pump Common Cause Failures | 7.8 E-004 | 7.8 E-004 | | No |
| HPI-MDP-FC-1A | HPI Motor-Driven Pump A Fails | 3.9 E-003 | 3.9 E-003 | | No |
| HPI-MDP-FC-1B | HPI Motor-Driven Pump B Fails | 3.9 E-003 | 3.9E-003 | | No |
| HPI-MOV-OC-SUC | HPI Serial Component Failures | 1.4 E-004 | 1.4 E-004 | | No |
| HPI-MOV-OO-RWST | Failure to Isolate RWST From HPI System | 3.0 E-003 | 3.0 E-003 | | No |
| HPI-XHE-NOREC | Operator Fails to Recover the HPI System | 8.4 E-001 | 8.4 E-001 | | No |
| HPR-XHE-NOREC | Operator Fails to Recover the HPR System | 1.0 E+000 | 1.0 E+000 | | No |
| PCS-XHE-DEPRES | Failure to depressurize the RCS within 30 minutes | 1.0 E-001 | 1.0 E-001 | | No |
| RHR-MDP-CF-ALL | RHR Pump Common Cause Failures | 4.5 E-004 | 4.5 E-004 | | No |
| RHR-MDP-FC-1A | RHR Motor-Driven Pump 1A Fails | 4.1 E-003 | 4.1 E-003 | | No |
| RHR-MDP-FC-1B | RHR Motor-Driven Pump 1B Fails | 4.1 E-003 | 4.1 E-003 | | No |
| RHR-MOV-CC-SUC1 | Failure of RHR Hot Leg Suction MOV A | 3.0 E-003 | 3.0 E-003 | | No |
| RHR-MOV-CC-SUC2 | Failure of RHR Hot Leg Suction MOV B | 3.0 E-003 | 3.0 E-003 | | No |
| RHR-MOV-OO-RWST | Failure to Isolate the RWST During RHR | 3.0 E-003 | 3.0 E-003 | | No |
| 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 No. 315/95-011**

| Event tree name | Sequence name | CCDP | CDP | Change to CCDP (Importance) | Percent Contribution to Importance |
|---|---|---|---|---|---|
| SLOCA | 06 | 3.3 E-06 | 2.9 E-08 | 3.3 E-006 | 77.2 |
| SGTR | 08 | 5.4 E-07 | 4.8 E-09 | 5.4 E-007 | 12.5 |
| SLOCA | 03 | 2.2 E-06 | 2.0 E-06 | 1.6 E-007 | 3.7 |
| TRANS | 08 | 8.9 E-08 | 7.9 E-10 | 8.8 E-008 | 2.0 |
| SLOCA | 05 | 9.6 E-08 | 3.2 E-08 | 6.3 E-008 | 1.4 |
| Subtotal Case 1 (shown)[a] | | 6.3 E-06 | 2.0 E-06 | 4.3 E-006 | |
| Subtotal Case 2[b] | | 3.6 E-06 | 2.4 E-07 | 3.4 E-006 | |
| Subtotal Case 3[c] | | 2.7 E-08 | 8.9 E-09 | 1.8 E-008 | |
| Total (all sequences) | | 9.9 E-06 | 2.2 E-06 | 7.7 E-006 | |

[a] Case 1 represents the increase in the core damage probability due to the long term unavailability of the West CCP (4252 h).

[b] Case 2 represents the increase in the core damage probability due to the unavailability of the opposite train EDG that was periodically unavailable while the West CCP was unavailable (50 h).

[c] Case 3 represents the increase in the core damage probability due to the time that both CCPs were simultaneously unavailable because of various maintenance activities (18 h).

Table 3. Sequence logic for dominant sequences for LER No. 315/95-011 (Case 1 only)

| Event tree name | Sequence name | Logic |
|---|---|---|
| SLOCA | 06 | /RT, /AFW, HPI |
| SGTR | 08 | /RT, /AFW, HPI, RCS-SG-H |
| SLOCA | 03 | /RT, /AFW, /HPI, /COOLDOWN, RHR, HPR |
| TRANS | 08 | /RT, /AFW, PORV, PORV-RES, HPI |
| SLOCA | 05 | /RT, /AFW, /HPI, COOLDOWN, HPR |

Table 4. System names for LER No. 315/95-011 (Case 1 only)

| System name | Logic |
|---|---|
| AFW | No or Insufficient AFW Flow |
| COOLDOWN | RCS Cooldown to RHR Pressure Using Turbine-Bypass Valves, etc. |
| HPI | No or Insufficient Flow from HPI System |
| HPR | No or Insufficient HPR Flow |
| RHR | No or Insufficient Flow from RHR System |
| RCS-SG-H | Failure to depressurize the RCS below the SG safety valve setpoint without HPI |
| RT | Reactor Fails to Trip During Transient |

**Table 5. Conditional cut sets for higher probability sequences for LER No. 315/95-011**

| Cut set No. | Percent Contribution | CCDP (Importance)[a] | Cut sets |
|---|---|---|---|
| SLOCA Sequence 06 | | 3.3 E-006 | |
| 1 | 83.1 | 2.8 E-006 | HPI-MDP-CF-ALL, HPI-XHE-NOREC |
| 2 | 14.9 | 5.0 E-007 | HPI-MOV-OC-SUC, HPI-XHE-NOREC |
| 3 | 1.6 | 5.4 E-008 | HPI-MDP-FC-1A, HPI-MDP-FC-1B, HPI-XHE-NOREC |
| SGTR Sequence 08 | | 5.4 E-007 | |
| 1 | 83.1 | 4.5 E-007 | HPI-MDP-CF-ALL, HPI-XHE-NOREC, PCS-XHE-DEPRES |
| 2 | 14.9 | 8.0 E-008 | HPI-MOV-OC-SUC, HPI-XHE-NOREC, PCS-XHE-DEPRES |
| 3 | 1.6 | 8.6 E-009 | HPI-MDP-FC-1A, HPI-MDP-FC-1B, HPI-XHE-NOREC, PCS-XHE-DEPRES |
| SLOCA Sequence 03 | | 1.6 E-007 | |
| 1 | 86.4 | 1.5 E-007 | RHR-MDP-CF-ALL, RHR-XHE-NOREC, HPR-XHE-NOREC |
| 2 | 3.2 | 6.2 E-009 | RHR-MDP-FC-1A, RHR-MDP-FC-1B, RHR-XHE-NOREC, HPR-XHE-NOREC |
| 3 | 1.7 | 3.6 E-009 | HPI-MOV-OO-RWST, RHR-MOV-CC-SUC2, RHR-XHE-NOREC, HPR-XHE-NOREC |
| 4 | 1.7 | 3.6 E-009 | HPI-MOV-OO-RWST, RHR-MOV-OO-RWST, RHR-XHE-NOREC, HPR-XHE-NOREC |
| 5 | 1.7 | 3.6 E-009 | HPI-MOV-OO-RWST, RHR-MOV-CC-SUC1, RHR-XHE-NOREC, HPR-XHE-NOREC |
| TRANS Sequence 08 | | 8.8 E-008 | |
| SLOCA Sequence 05 | | 6.3 E-008 | |
| Subtotal Case 1[b] (shown above) | | 4.3 E-006 | |
| Subtotal Case 2[c] | | 3.4 E-006 | |
| Subtotal Case 3[d] | | 1.8 E-008 | |

## Table 5. Conditional cut sets for higher probability sequences for LER No. 315/95-011

| Cut set No. | Percent Contribution | CCDP (Importance)[a] | Cut sets |
|---|---|---|---|
| Total (all sequences) | | 7.7 E-006 | |

[a] The change in conditional probability (importance) is determined by calculating the conditional probability for the period in which the condition existed and given the condition, and subtracting the conditional probability for the same period but with plant equipment assumed to be operating nominally. The conditional probability for each cut set within a sequence is determined by multiplying the probability that the portion of the sequence that makes the precursor visible (e.g., the system with a failure is demanded) will occur during the duration of the event by the probabilities of the remaining basic events in the minimal cut set. This can be approximated by $1 - e^{-p}$, where p is determined by multiplying the expected number of initiators that occur during the duration of the event by the probabilities of the basic events in that minimal cut set. The expected number of initiators is given by $\lambda t$, where $\lambda$ is the frequency of the initiating event (given on a per hour basis), and t is the duration time of the event. This approximation is conservative for precursors made visible by the initiating event. The frequencies of interest for this event are: $\lambda_{TRANS} = 5.3 \times 10^{-4}$/h, $\lambda_{LOOP} = 3.8 \times 10^{-6}$/h, $\lambda_{SLOCA} = 1.0 \times 10^{-4}$/h, and $\lambda_{SGTR} = 1.6 \times 10^{-4}$/h.

[b] Case 1 represents the increase in the core damage probability due to the long term unavailability of the West CCP (4252 h).

[c] Case 2 represents the increase in the core damage probability due to the unavailability of the opposite train EDG that was periodically unavailable while the West CCP was unavailable (50 h).

[d] Case 3 represents the increase in the core damage probability due to the time that both CCPs were simultaneously unavailable because of various maintenance activities (18 h).

# 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.