50-272/311

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

April 30, 1997

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric and Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF OPERATIONAL CONDITION DISCOVERED AT SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2

Dear Mr. Eliason:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which was discovered at the Salem Nuclear Generating Station, Units 1 and 2, on January 10, 1996 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 272/96-002. 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 for 1996. In assessing operational events, an effort was made to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures, or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of the review process is to provide as realistic an analysis of the significance of the event as possible.

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

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

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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. 272/96-002, which documented the event.

Please contact me at (301) 415-1419 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,

(Original signed by)

Lenny N. Olshan, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/311

Enclosures: 1. Preliminary Accident Precursor Analysis
2. Guidance for Performing Review
3. LER No. 272/96-002

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LER Nos. 272/96-002

Event Description:	Charging pump suction valves from the RWST potentially unavailable due to pressure locking
Date of Event:	January 10, 1996
Plant:	Salem 1 and 2

Event Summary

During an evaluation for potential pressure locking and thermal binding in power-operated gate valves, as required by Generic Letter (GL) 95-07, personnel determined that the following valves on both units were subject to pressure locking (Fig. 1):

Valves SJ1 and SJ2	Valves on the Refueling Water Storage Tank (RWST) supply line to the charging/safety injection pump suction
Valves 1SJ113 and 2SJ113	Valves on the cross tie connection from the suction of the charging pumps to the suction of the safety injection (SI) pumps
Valves 1CS2 and 2CS2	Isolation valves on the containment spray header

Pressure locking of valves SJ1 and SJ2 could prevent these valves from opening when required for safety injection. Pressure locking of valves 1SJ113 and 2SJ113 could prevent these valves from opening if required during the recirculation phase of a loss-of-coolant accident (LOCA). Pressure locking of valves 1CS2 and 2CS2 could impact the containment spray function prior to the recirculation phase of a LOCA.

In addition, the following valves were determined to be susceptible to thermal binding:

Valves PR6 and PR7	Power Operated Relief Valve (PORV) block valves

Thermal binding of these block valves could render the associated PORV unavailable for feed-and-bleed operations if the block valve were to be closed prior to the existence of an accident condition.

Both units were shut down and defueled at the time of the evaluation. This analysis assumes the susceptible valves could impact the plant response to a small-break LOCA, a steam generator tube rupture (SGTR), a PORV lifting and failing to reseat, and a reactor coolant pump seal package failure. An increase in the core damage probability (CDP) for a one-year period of 5.8×10^{-6} was calculated over a nominal value for the same period of 3.0×10^{-5}. This increase in CDP is applicable to each unit.

Enclosure 1

Event Description

On January 10, 1996, both units were shutdown and defueled. At that time, Public Service Electric and Gas Company determined that the RWST supply valves to the charging/SI pump suction, the valves on the cross tie connection from the suction of the charging pumps to the suction of the SI pumps, and the containment spray header isolation valves on both units were subject to pressure locking. Additionally, the PORV block valves on both units were determined to be subject to thermal binding.

Pressure locking occurs when the fluid in the valve bonnet is at a higher pressure than the adjacent piping at the time of the valve opening. The two most likely scenarios for elevating the pressure in the valve bonnet relative to the pressure in the valve system are given below.

1. Thermal pressure locking (or bonnet heatup) can occur when an incompressible fluid is trapped in the valve bonnet (e.g., during valve closure), followed by heating-up of the volume in the bonnet. The bonnet heatup scenarios include heating the valve bonnet by an increase in the temperature of the environment during an accident, heatup due to an increase in the temperature of the process fluid on either side of the valve, etc. (Normal ambient temperature variation is not considered, because it occurs over a long time period and pressure changes tend to be alleviated through extremely small amounts of leakage. Further, operating experience shows that normal temperature variations are not a source of pressure locking events.)

2. Hydraulic pressure locking (or pressure-trapping) can occur when an incompressible fluid is trapped in the valve bonnet, followed by depressurization of the adjacent piping prior to valve opening. Examples of hydraulic pressure locking scenarios include back-leakage past check valves, and system operating pressures that are higher than the system pressure when the valve is required to open.

Pressure locking is of concern because the pressure in the space between the two discs of a gate valve can become pressurized above the pressure assumed when sizing the valve's motor operator. This prevents the valve operator from opening the valve when required.

Thermal binding is a phenomenon where temperature changes of the valve internal components cause the valve stem to expand after closure. This results in a higher required opening thrust that may be above the opening thrust assumed when sizing the valve motor operator.

The original plant designs at Salem 1 and 2 did not account for pressure locking and thermal binding effects. In 1977, plant personnel modified double-disc gate valves based on recommendations by Westinghouse. In 1986, a review of flexible wedge gate valves in response to INPO SOER 84-7 determined that the valves listed in this licensee event report (LER) were not susceptible to pressure locking or thermal binding. The more stringent requirements of GL 95-07 reversed this earlier conclusion for the above listed valves.

Licensee personnel determined that valves SJ1 and SJ2 were subject to the "pressure-trapping" effect. A maximum bonnet pressure of 96 psig was estimated, based on quarterly surveillance testing that recirculated water from the residual heat removal (RHR) pump discharge to the RWST suction line where SJ1 and SJ2 are located. The licensee indicated that once this increased bonnet pressure was established, there was no mechanism for the pressure to be relieved (assumption used in response to GL 95-07). At degraded voltage conditions, the licensee could not guarantee sufficient thrust would be generated by the motor operator to overcome the bonnet pressure. This would result in a loss of high head injection, though the charging pumps would be available for high pressure recirculation.

Valves 1SJ113 and 2SJ113 were determined to be subject to both the "pressure-trapping" effect and "bonnet heatup." The maximum bonnet pressure was estimated to be 225 psig. Again, once this increased bonnet pressure was established, there was no mechanism for the pressure to be relieved. The "bonnet heatup" occurs in the first two minutes following the initiation of the hot leg recirculation phase of a LOCA. The "pressure-trapping" is the result of surveillance testing.

Valves 1CS2 and 2CS2 were determined to be subject to the "pressure-trapping" effect. A maximum bonnet pressure was estimated at 250 psig as a result of surveillance testing of the containment spray pumps immediately upstream of the valves. Pump start on a containment high pressure may relieve the high pressure on the upstream side of the disc and allow the valves to open.

Valves PR6 and PR7 were determined to be subject to thermal binding. These valves are normally open at power unless they are cycled for surveillance testing or there is a fault on the PORV.

Additional Event-Related Information

The charging system (CVC) consists of two centrifugal charging pumps and one positive displacement pump. On a safety injection (SI) signal, the centrifugal charging pumps provide for high head safety injection. If valves SJ1 and SJ2 fail closed, the safety injection function of the charging pumps is defeated. However, assuming the charging pumps were throttled back following the failure of SJ1 and SJ2 prior to damage from a loss of suction, they would still be capable of providing service in the (piggyback) recirculation mode. The two safety injection pumps provide for intermediate head safety injection. Failure of valves SJ1 and SJ2 does not impact this mode of injection into the reactor coolant system.

Piggyback recirculation to the charging system and the SI system is provided separately by the individual RHR pumps. The A RHR pump provides for piggyback recirculation to the SI pumps' suction header. The B RHR pump provides for piggyback recirculation to the CVC suction header. Valves 1SJ113 and 2SJ113 are in parallel and connect the CVC and SI suction headers together. This provides an alternate path for recirculation to either the CVC or SI system should the primary path fail. Therefore, failure of both SJ113 valves does not fail piggyback recirculation without an additional failure occurring.

The containment spray system takes a suction from the RWST and delivers spray flow to the containment via valves 1CS2 and 2CS2. A failure of these valves to open would preclude containment spray using the containment spray pumps. Downstream of the CS2 valves, a connection from the discharge of the RHR pumps exists to provide containment spray in the recirculation phase of a LOCA. This path would be unaffected by a failure of the CS2 valves. Additionally, there are five containment cooler units that will limit the pressure increase in containment, assuming all five units operate without failure.

Modeling Assumptions

Valves SJ1 and SJ2 were considered to be unavailable due to pressure locking following a small-break LOCA (SLOCA). The charging pumps or the SI pumps are required to protect the core during a SLOCA. Since the pressure locking mechanism was assumed to be from "pressure-trapping," the condition was assumed to have existed for a one-year period following surveillance testing that recirculated higher pressure water back to the RWST. Basic event CVC-MOV-CC-SUCT represents the combination of SJ1 and SJ2 failing closed, so this event was set to "TRUE" (failed). The common cause failure of SJ1 and SJ2 was removed by setting basic event CVC-MOV-CF-SUCT to "FALSE" (not possible).

The NRC's simplified, plant-specific models used in ASP analyses currently do not include models for l large-break and medium-break LOCAs. These larger LOCAs are predicted to remove all decay heat out of the break location. Therefore, accumulator response and the progression to the recirculation mode are the key elements in a large-break or medium-break LOCA event tree. Those responses are not significantly impacted by the valve failures reported in the LER. Thus, no effort was made to model these accident conditions.

Similar to valves SJ1 and SJ2, 1SJ113, and 2SJ113 were considered to be unavailable due to pressure locking following a SLOCA. These valves are not specifically included in the NRC's simplified models; therefore, a basic event representing the probability of failure of 1SJ113 and 2SJ113 was added (HPR-MOV-CC-HPI) (base failure probability of 9.0 E-06) to the high pressure recirculation (HPR) and the HPR-L (LOOP) fault trees. This basic event was added via an OR gate with a new basic event (HPR-HPI-FM-CVC or HPR-CVC-FM-HPI) representing the success of the recirculation flow path elements in the opposite RHR train. Subsequently, basic event HPR-MOV-CC-HPI was set to "TRUE" (failed).

The PORV block valves (PR6 and PR7) were not considered in the analysis. These valves are subject to thermal binding, which can be mitigated over time. Additionally, these valves would need to be closed at the initiation of an accident to impact the ability of the unit to conduct feed and bleed operations. The Salem IPE indicates the probability of PR6 or PR7 being closed could range as high as 3.2×10^{-5}. When combined with the probability of an accident condition requiring feed and bleed, consideration of a PORV block valve failure becomes insignificant for analysis purposes. Additionally, the Salem FSAR does not take credit for the PORV valves mitigating the severity of any accident.

It was assumed that the failure of the containment spray valves would not impact the probability of core damage. The licensee considered that the containment spray pump start following a SI signal would likely relieve pressure on 1CS2 and 2CS2, allowing these valves to open. Furthermore, there appear to be several alternatives to reduce containment pressure if required.

The plant-specific model of the plant's response to a SGTR was modified. Previously, a loss of the high pressure injection function led directly to core damage. For this analysis, the possibility of lowering RCS pressure below the steam generator safety valve set point within 30 min was considered following the loss of high pressure injection capability by adding a basic event, PCS-XHE-DEPR-30. Based on the operator burden under a short time constraint, a failure probability of 0.1 was assigned to PCS-XHE-DEPR-30.

The probability associated with the basic event for the failure of the operator to switch the AFW system water supply to a backup source (AFW-XHE-XA-CST) was reduced from 4.0×10^{-2} to 1.0×10^{-3}. This change was based on the relatively large size of the normal AFW water supply (200,000 gallons) and the added time which the operator would have to switch to a backup source of water.

Analysis Results

This event is most sensitive to a SLOCA sequence which accounts for 81% of the increase in the CDP for the 1-year period analyzed. An overall increase of 5.8×10^{-6} in the CDP was calculated. This is above a base probability for core damage (the CDP) for the same period of 3.0×10^{-5}. The dominant core damage sequence, highlighted as sequence number 06 on the event tree in Fig. 2, involves:

- a SLOCA,
- the successful trip of the reactor,
- the successful operation of the auxiliary feedwater (AFW) system, and
- the failure of the HPI system (SI pumps) combined with the initial injection phase failure of the CVC system.

The next most significant sequence involves a SGTR and contributes 13% of the total increase in the CDP. This sequence also leads to core damage based on a failure of the HPI system and a failure to depressurize the RCS in a timely manner. Loss of HPI is the primary failure mechanism involved in all of the most dominant core damage sequences.

The first sequence that involves a failure of HPR or the failure of valves 1SJ113 and 2SJ113 is LOOP sequence number 10. This sequence contributes less than 1% to the total increase in the CDP. Therefore, the only significant valve failure related to this analysis from the LER involves the pressure locking of valves SJ1 and SJ2.

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

Acronyms

CCDP	conditional core damage probability
CDP	core damage probability
CVC	charging system
GL	generic letter
HPI	high pressure injection
HPR	high pressure recirculation
IPE	individual plant examination
IRRAS	Integrated Reliability and Risk Analysis System
LER	licensee event report
LOCA	loss of coolant accident
LOOP	loss of offsite power
MOV	motor operated valve
PORV	power operated relief valve
PWR	pressurized water reactor
RCS	reactor coolant system
RHR	residual heat removal
RWST	refueling water storage tank
SGTR	steam generator tube rupture
SLOCA	small-break loss of coolant accident
SI	safety injection

References

1. LER 272/96-002, Rev. 1, "Motor Operated Gate valves Susceptible to Pressure Locking and Thermal Binding," February 9, 1996.

2. *Salem Generating Station Individual Plant Examination*, July, 1993.

3. *Salem Generating Station Updated Final Safety Analysis Report*, Volume 3.

4. Phone conversation with Dennis Hassler and Bob Lewis, Salem Generating Station, February 13, 1997.

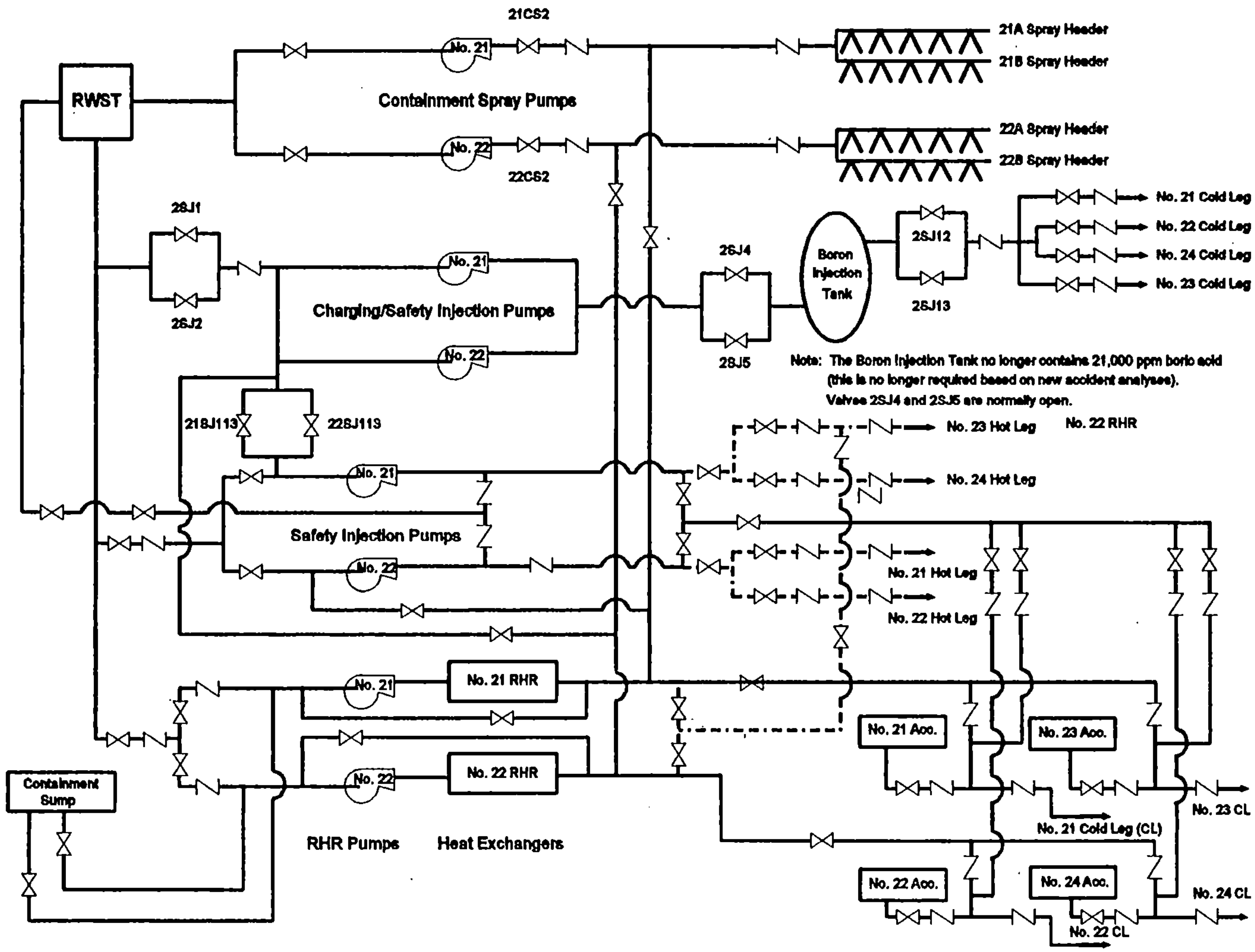

Fig. 1 Composite drawing of the Emergency Core Cooling System at Salem 2.

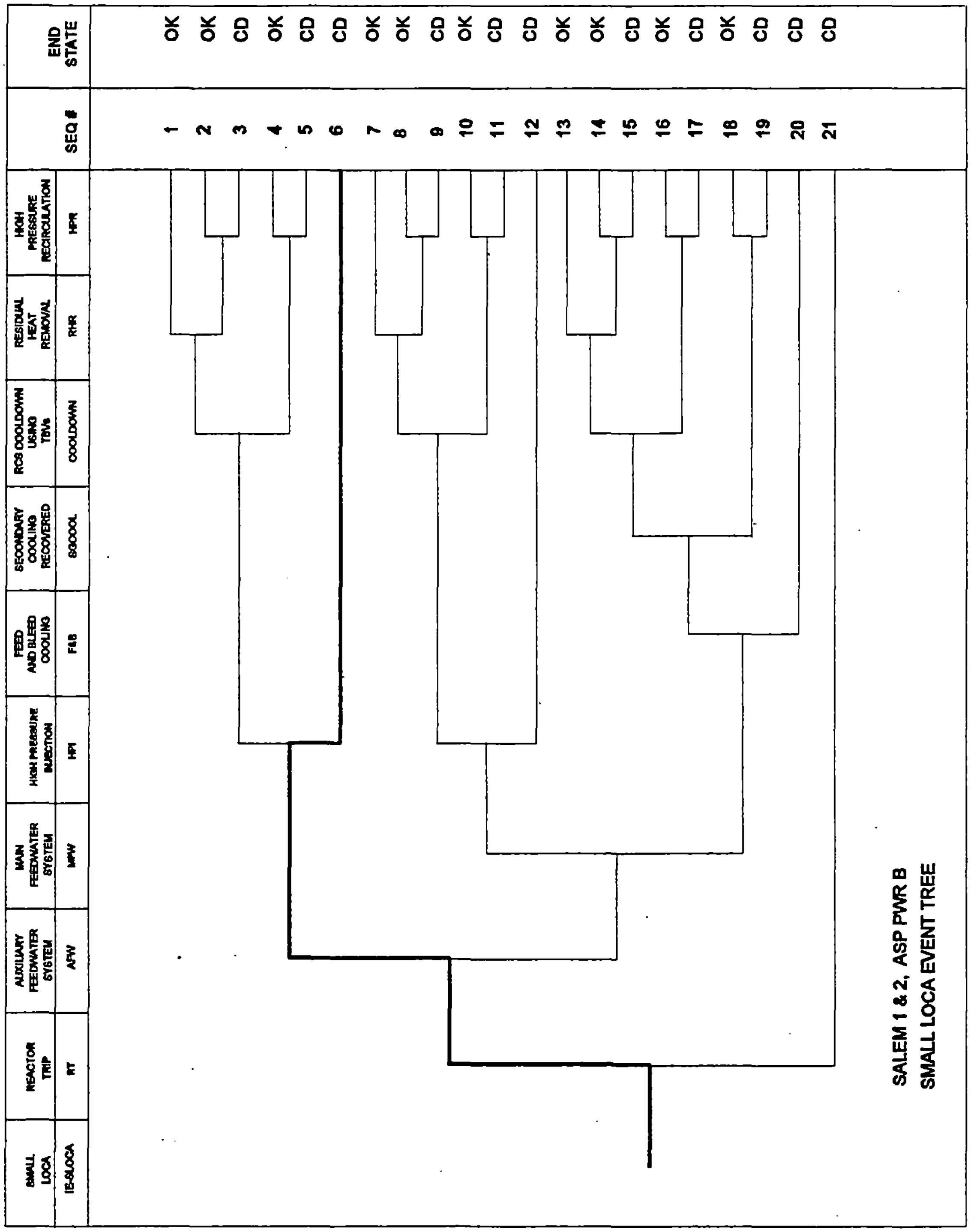

Fig. 2. Dominant core damage sequence for LER No. 272/96-002.

Table 1. Definitions and Probabilities for Selected Basic Events for LER No. 272/96-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event - Loss-of-Offsite Power	8.5 E-006	8.5 E-006		No
IE-SGTR	Initiating Event - Steam Generator Tube Rupture	1.6 E-006	1.6 E-006		No
IE-SLOCA	Initiating Event - Small-Break Loss-of-Coolant Accident	1.0 E-006	1.0 E-006		No
IE-TRANS	Initiating Event - Transient	5.3 E-004	5.3 E-004		No
AFW-PMP-CF-ALL	Common-Cause Failure of AFW Pumps	2.8 E-004	2.8 E-004		No
AFW-XHE-NOREC	Operator Fails to Recover the AFW System	2.6 E-001	2.6 E-001		No
AFW-XHE-XA-CST	Operator Fails to Initiate Backup Water Supply	4.0 E-002	1.0 E-003		Yes
CVC-MOV-CC-SUCT	Failure of CVC RWST Suction MOVs to Open (SJ1 and SJ2)	1.1 E-004	1.0 E+000	TRUE	Yes
CVC-MOV-CF-SUCT	CVC-HPI RWST Suction fails to Open (SJ1 and SJ2) CCF	2.6 E-004	2.6 E-004	FALSE	Yes
HPI-MDP-CF-ALL	Common-Cause Failure of HPI Motor-Driven Pumps	7.8 E-004	7.8 E-004		No
HPI-MDP-FC-1A	HPI Train A Fails	3.9 E-003	3.9 E-003		No
HPI-MDP-FC-1B	HPI Train B Fails	3.9 E-003	3.9 E-003		No
HPI-MOV-OC-SUCT	HPI Suction Valves Fail (SJ30)	1.4 E-004	1.4 E-004		No
HPI-XHE-NOREC	Operator Fails to Recover the HPI System	8.4 E-001	8.4 E-001		No
HPI-XHE-XM-FB	Operator Fails to Initiate Feed-and-Bleed Cooling	1.0 E-002	1.0 E-002		No
HPR-CVC-FM-HPI	HPR path to CVC from HPI Fails (excludes failure of SJ113 valves)	7.0 E-003	7.0 E-003	NEW	Yes
HPR-CVC-FM-HPI	HPR path to HPI from CVC Fails (excludes failure of SJ113 valves)	7.0 E-003	7.0 E-003	NEW	Yes

Table 1. Definitions and Probabilities for Selected Basic Events for LER No. 272/96-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPR-MOV-CC-HPI	Failure of SJ113 Suction Cross-Connect Valves	9.0 E-006	1.0 E+000	NEW/ TRUE	Yes
MFW-SYS-TRIP	MFW System Trips	2.0 E-001	2.0 E-001		No
MFW-XHE-NOREC	Operator Fails to Recover MFW	3.4 E-001	3.4 E-001		No
PCS-XHE-DEPR-30	Operator Fails to Depressurize RCS Within 30 Minutes (SGTR-Loss of HPI)	1.0 E-001	1.0 E-001	NEW	Yes
PPR-MOV-OO-BLK1	PORV 1 Block Valve Fails to Close	3.0 E-003	3.0 E-003		No
PPR-MOV-OO-BLK2	PORV 2 Block Valve Fails to Close	3.0 E-003	3.0 E-003		No
PPR-SRV-CC-1	PORV 1 Fails to Open on Demand	3.0 E-002	3.0 E-002		No
PPR-SRV-CC-2	PORV 2 Fails to Open on Demand	3.0 E-002	3.0 E-002		No
PPR-SRV-CO-TRAN	PORVs Open During Transient	4.0 E-002	4.0 E-002		No
PPR-SRV-OO-1	PORV 1 Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No
PPR-SRV-OO-2	PORV 2 Fails to Reclose After Opening	3.0 E-002	3.0 E-002		No
PPR-XHE-NOREC	Operator Fails to Close PORVs or Block Valves	1.1 E-002	1.1 E-002		No

Table 2. Sequence Conditional Probabilities for LER No. 272/96-002

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP-CDP)	Percent contribution[a]
SLOCA	06	4.8 E-006	4.2 E-008	4.8 E-006	82.1
SGTR	08	7.8 E-007	6.9 E-009	7.8 E-007	13.3
TRANS	08	8.6 E-008	7.5 E-010	8.5 E-008	1.4
TRANS	20	1.7 E-006	1.7 E-006	6.0 E-008	1.0
Total (all sequences)		**3.6 E-005**	**3.0 E-005**	**5.8 E-006**	

[a] Percent contribution to the total importance.

Table 3. Sequence Logic for Dominant Sequences for LER No. 272/96-002

Event tree name	Sequence name	Logic
SLOCA	06	/RT, /AFW, HPI
SGTR	08	/RT, /AFW, HPI, RCS-HPI
TRANS	08	/RT, /AFW, PORV, PORV-RES, HPI
TRANS	20	/RT, AFW, MFW, F&B

Table 4. System Names for LER No. 272/96-002

System name	Logic
AFW	No or Insufficient AFW Flow
F&B	Failure to Provide Feed-and-Bleed Cooling
HPI	No or Insufficient HPI Flow
MFW	Failure of the MFW System
PORV	PORVs Open During Transient
PORV-RES	PORVs Fail to Reseat
RCS-HPI	Failure to Depressurize RCS<SG Relief Within 30 Minutes (HPI Fails)
RT	Reactor Fails to Trip During Transient

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER No. 272/96-002

Cut set number	Percent Contribution[a]	Change in CCDP (Importance)[b]	Cut sets[c]
SLOCA Sequence 06		4.8 E-006	
1	83.3	4.0 E-006	CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
2	15.0	7.2 E-007	CVC-MOV-CC-SUCT, HPI-MOV-OC-SUCT, HPI-XHE-NOREC
3	1.6	7.8 E-008	CVC-MOV-CC-SUCT, HPI-MDP-FC-1A, HPI-MDP-FC-1A, HPI-XHE-NOREC
SGTR Sequence 08		7.8 E-007	
1	82.1	6.4 E-007	CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC, PCS-XHE-DEPR-30
2	14.1	1.1 E-007	CVC-MOV-CC-SUCT, HPI-MOV-OC-SUCT, HPI-XHE-NOREC, PCS-XHE-DEPR-30
3	1.5	1.2 E-008	CVC-MOV-CC-SUCT, HPI-MDP-FC-1A, HPI-MDP-FC-1A, HPI-XHE-NOREC, PCS-XHE-DEPR-30
TRANS Sequence 08		8.5 E-008	
1	32.9	2.8 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-XHE-NOREC, CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
2	32.9	2.8 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-XHE-NOREC, CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
3	9.1	7.7 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-MOV-OO-BLK2, CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
4	9.1	7.7 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-MOV-OO-BLK1, CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
5	5.9	5.0 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-XHE-NOREC, CVC-MOV-CC-SUCT, HPI-MOV-OC-SUCT, HPI-XHE-NOREC
6	5.9	5.0 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-XHE-NOREC, CVC-MOV-CC-SUCT, HPI-MOV-OC-SUCT, HPI-XHE-NOREC
7	1.6	1.4 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, PPR-MOV-OO-BLK2, CVC-MOV-CC-SUCT,HPI-MOV-OC-SUCT, HPI-XHE-NOREC
8	1.6	1.4 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, PPR-MOV-OO-BLK1, CVC-MOV-CC-SUCT,HPI-MOV-OC-SUCT, HPI-XHE-NOREC

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER No. 272/96-002

Cut set number	Percent Contribution[a]	Change in CCDP (Importance)[b]	Cut sets[c]
Trans Sequence 20		6.0 E-008	
1	60.0	3.6 E-008	AFW-XHE-XA-CST, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC , CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
2	16.7	1.0 E-008	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC , CVC-MOV-CC-SUCT, HPI-MDP-CF-ALL, HPI-XHE-NOREC
3	11.3	6.8 E-009	AFW-XHE-XA-CST, AFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC , CVC-MOV-CC-SUCT, HPI-MOV-OC-SUCT, HPI-XHE-NOREC
Total (all sequences)		**5.8 E-006**	

[a]Percent contribution to the sequence total importance

[b]The change in conditional probability (importance) is determined by calculating the conditional probability for the period in which the condition existed, 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 λt, where λ 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, $\lambda_{LOOP} = 8.5 \times 10^{-6}$/h, $\lambda_{SLOCA} = 1.0 \times 10^{-6}$/h, and $\lambda_{SGTR} = 1.6 \times 10^{-6}$/h.

[c]Basic event CVC-MOV-CC-SUCT is a type TRUE event. This type of event is not normally included in the output of the fault tree reduction process. This event has been added to aid in understanding the sequences to potential core damage associated with the event.

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.