Docket File

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

50-373

July 3, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF OPERATIONAL EVENT AT LASALLE COUNTY STATION, UNITS 1 AND 2

Dear Ms. Johnson:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at LaSalle County Station, Units 1 and 2, on June 28, 1996 (Enclosure 1), and was reported in Licensee Event Report (LERs) Nos. 373/96-007 and 373/96-008. 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 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 to incorporate ComEd's comments, perform any required reanalysis and prepare the final report of NRC's analysis of this event in a timely manner, ComEd is requested to complete its review and to provide any comments within 30 days of receipt of this letter. NRC has 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 the final analysis of the event has been completed, NRC will provide (for ComEd's information) the final precursor analysis of the event and the resolution of ComEd's 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.

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Also enclosed are several items to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, identifies the

criteria which NRC 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 ComEd should provide to support such a claim. Enclosure 3 is a copy of LER Nos. 373/96-007 and 373/96-008, that documented the event.

Please contact me at (301) 415-1322 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. ComEd's response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

ORIGINAL SIGNED BY:

Donna M. Skay, Project Manager
Project Directorate III-2
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Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures: As stated

cc w/encls: See next page

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LER No. 373/96-007, -008

Event Description: Concrete sealant fouls cooling water systems

Date of Event: June 28, 1996

Plant: LaSalle 1 and 2

Event Summary

A foam sealant was inadvertently injected into the service water tunnel for Units 1 and 2 resulting in fouling of nonessential service water, and potential fouling of essential (safety-related) cooling water systems. The conditional core damage probability (CCDP) estimated for this event is 2.3×10^{-5} at LaSalle 2. Because of the location of the sealant and the location of the emergency service water system (ESW) pump intakes, the significance of this event would be slightly lower for LaSalle 1 than for LaSalle 2.

Event Description

The Licensee Event Report (LER) for this event [Ref. 1, 2] reports that between May 21 and June 21, 1996, contractors, as requested by the station's Consolidated Facility Maintenance group, began sealing cracks in the walls and floors of the raw water intake building. The repair process involved drilling holes into the side of each crack along its length and injecting an expandable foam sealant ("Furmanite") into these holes to seal the crack. While performing the repairs, workers started fixing cracks on the top of the service water tunnel. This tunnel supplies cooling water to both the nonessential and essential service water systems. Because the workers believed that they were working on a concrete floor laid over soil, they proceeded to drill five holes through the ceiling of the service water tunnel. Consequently, instead of injecting sealant into a void under the building floor, the material was injected into the service water tunnel. In all, personnel injected between 80 and 120 cubic feet of sealant into the tunnel.

On June 19, 1996, a high differential pressure developed across the non-essential service water (NESW) strainers for both units. An automatic backwash for the strainers for both units failed and differential pressure across the strainers exceeded the normal backwash setpoint by as much as 8 psid. Reactor power was reduced to about 77% on both units to reduce loading of the service water system and the strainers were isolated one at a time for repairs. After the repairs were completed, operators were able to manually backwash each strainer successfully. Initially, it was believed that the strainers had been fouled by "corn cob" material being used in sandblasting the exterior of the raw water intake building.

On June 24, 1996, normal surveillance tests were attempted on the station's diesel fire water pumps (DFPs). While DFP 0A performed satisfactorily, DFP 0B experienced a high cooling water outlet temperature, indicative of flow blockage, after about 5 min of operation. DFP 0B was shut down and both pumps were declared inoperable. Later on the same day, high differential pressures were experienced again across the

non-essential service water strainers, which again failed to successfully automatically backwash. Operators reduced plant service water heat loads and manually backwashed the strainers.

On June 25 and June 26, test runs of the five emergency diesel generators and the four residual heat removal systems were made, and no operability problems were noted. Unit 1 scrammed due to instrument calibration problems late on June 26. On June 27, a Unit 2 service water strainer was inspected and was found to be operable. On June 28, divers performed an inspection of the service water tunnel and discovered that a significant amount of debris remained present (this was later determined to be the Furmanite sealant). Essential service water systems taking suction from the service water tunnel were declared inoperable. Unit 1 was scrammed from 1% power, and Unit 2 was reduced from 100% to 5% power and then scrammed.

Additional Event-Related Information

Both units were maintained in hot shutdown so that decay heat removal systems relying upon ESW would not be demanded and the service water tunnel was extensively cleaned. Between 80 and 120 cubic feet of sealant material were removed from the tunnel, including one large piece measuring about 8 ft wide by 15 ft long by up to 1 ft thick.

The Augmented Inspection Team report and other information provided by personnel at Commonwealth Edison [Ref. 3, 6] indicate that on July 4 and July 5, 1997, ESW pumps were run for several hours for testing and all performed adequately, except for the U1 A fuel pool emergency makeup (FPEM) pump and the U2 B FPEM pump, which were out of service. On July 5, the 2A RHRSW strainer was opened for inspection. A large number of sealant pieces of significant size were found and the strainer was described as being filled 50-60% with debris. Examination of several tubes with a borescope revealed that all tubes examined were blocked with sealant.[1] Vendor information indicated that debris particles greater than 2 in. in size would not be removed during strainer backwash because the internal diameter (ID) of the backwash arm was 2 in. Some pieces of sealant debris were observed to have at least one dimension greater than the ID of the backwash arm. Only small amounts of debris were found in the balance of the ESW system. On July 7, both units were placed in cold shutdown.

Subsequent analysis determined that, given the layout of the service water tunnel and the sealant injection points, the vulnerability of Unit 1 ESW systems to fouling was low. Potentially vulnerable systems taking suction from the Unit 2 end of the service water tunnel include the Unit 1 and Unit 2 NESW systems, 2A and 2B RHRSW systems, the 2A diesel generator cooling water system, 2A and 2B fuel pool emergency makeup pumps, and 0A and 0B diesel fire pumps.

Modeling Assumptions

A probabilistic risk analysis of the event prepared by the licensee [Ref. 4] indicates that, based on the judgment of personnel responding to the events, the combined probability that either the June 19 or the June

[1]Not all tubes could be examined using the borescope.

24 strainer failures could have resulted in a total loss of normal service water and subsequent scram was about 0.5. This assumption appears reasonable, and the event was analyzed as an at-power scram with non-safety-related service water unavailable. Unavailability of the NESW renders dependent systems unavailable, including the condensate system (CDS), main feedwater (MFW), power conversion systems (PCS), the control rod drive hydraulic supply system (CRD), and systems required for containment venting (CVS).

No test data could be identified to show that the potentially affected systems were fully operable after the June 19 event. Some testing was performed around June 25, 1996, that indicated that ESW was operable after the second fouling event. An exhaustive system test was performed on July 4, 1996, after a cleanup of the sealant material. Results from this testing also indicated that ESW was operable.

The 2A RHRSW system strainer was reported to be 50-60% fouled with sealant debris subsequent to the aforementioned testing. This fouling occurred despite the fact that the system had been tested after the two reported strainer fouling events and much of the original quantity of debris possibly removed from the tunnel. In addition, the 0B diesel fire pump was found by surveillance testing on June 24 to be inoperable, due to sealant fouling of its cooling water supply.

It is difficult to assess the likelihood that additional safety-related systems dependent on the service water tunnel could have been rendered inoperable by sealant fouling, but it is clear that the sealant material did present a potential concern. In its probabilistic risk assessment of the event, Commonwealth Edison assumed a probability of failure of 0.01 for division 2 core standby coolant supply systems dependent upon ESW due to sealant fouling. Independent failure of the ESW trains was incorporated into the ASP model fault trees with events DIV1FOUL and DIV2FOUL. The primary impact on plant safety systems from the loss of a train of ESW (either by DIV1FOUL or DIV2FOUL) is to fail the service water supply to an RHR heat exchanger. An independent failure probability of 0.01 was used for each basic event. Consistent with the common-cause strainer failure data presented in INEL-94/0064 (Ref. 5), a common-cause failure probability of 0.1 for the second train of ESW, given that the first train has failed, was also added to the ASP model (basic event ESW-CCF).

In addition, the B fuel pool emergency makeup pump was assumed to be failed based on additional information obtained about the event [Ref. 6].

Analysis Results

The CCDP estimated for this event is 2.3×10^{-5}. The dominant sequence highlighted on the event tree in Figure 1 involves

- the postulated scram with unavailability of PCS and main feedwater,
- failure of RHR, and
- unavailability of containment venting.

As stated above, the loss of NESW fails CDS, MFW, PCS, CRD, and CVS.

Definitions and probabilities for 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

ASP	accident sequence precursor
CCDP	conditional core damage probability
CDS	condensate storage system
CRD	control rod drive system
CVS	containment venting system
DFPs	diesel fire water pumps
ESW	essential service water
HPCS	high pressure core spray
ID	internal diameter
LER	licensee event report
MFW	main feedwater
NESW	non-essential service water
PCS	power conversion system
RCIC	reactor core isolation cooling
RHR	residual heat removal
RHRSW	residual heat removal system service water
SRVs	safety relief valves
TRANS	transient

References

1. Licensee Event Report (373/96-008) from Commonwealth Edison to the U.S. Nuclear Regulatory Commission, "Foreign Material Injected Into Service Water Tunnel Causes Dual Unit Shutdown Due to Inadequate Work Control," November 25, 1996.

2. Licensee Event Report (373/96-007) from Commonwealth Edison to the U.S. Nuclear Regulatory Commission, "Unit 1 Reactor Scram on Main Steam Flow High Trip Isolation During Surveillance," July 24, 1996.

3. "NRC Region III Augmented Inspection Team Review of the Potential Loss of the Ultimate Heat Sink Due to Foreign Material in the Safety-Related Service Water Intake Tunnel Inspection Report," U.S. Nuclear Regulatory Commission, August 2, 1996.

4. "PRA Report of the Impact of Foam Sealant Injection in the LaSalle County Nuclear Station Service Water Tunnel," Commonwealth Edison, September 23, 1996.

5. Marshall and Rasmuson, *Common-Cause Failure Data Collection and Analysis System Volume 6 – Common-Cause Failure Parameter Estimates*, INEL-94/0064, December 1995.

6. Teleconference involving personnel from the U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, and Commonwealth Edison, April 11, 1997.

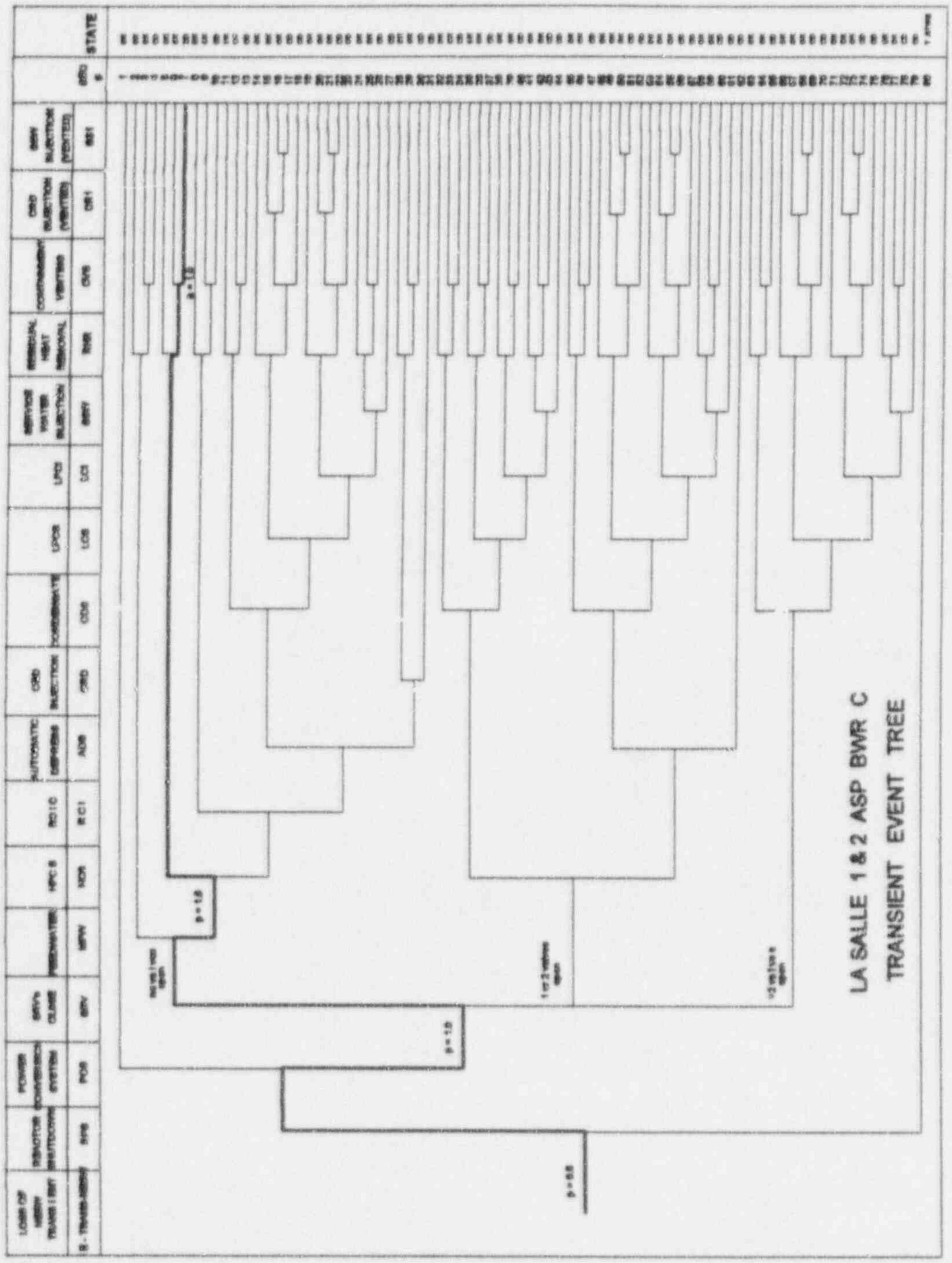

Figure 1. Dominant core damage sequence for LER Nos. 373/96-007, -008. (The loss of NESW fails the following systems: CDS, MFW, PCS, CRD, and CVS.)

Table 1. Definitions and Probabilities for Selected Basic Events for LER No. 373/96-007, -008

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss-of-Offsite Power Initiating Event	8.5 E-006	0.0 E+000		Yes
IE-SLOCA	Small Loss-of-Coolant Initiating Event	1.7 E-006	0.0 E+000		Yes
IE-TRANS-NESW	Transient (TRANS) Initiating Event–Loss-of-NESW	5.0 E-001	5.0 E-001	NEW	Yes
ADS-XHE-XE-ERROR	Operator Fails to Depressurize Using the Automatic Depressurization System	1.0 E-002	1.0 E-002		No
CDS-TNK-HW-CST	Condensate Storage Tank Fails	1.0 E-005	1.0 E-005		No
CSS-XHE-XE-NOREC	Operator Fails to Recover Containment Sprays	1.0 E+000	1.0 E+000		No
DIV1FOUL	Failure of Division 1 Safety Systems due to the Loss of ESW	1.0 E-002	1.0 E-002	NEW	Yes
DIV2FOUL	Failure of Division 2 Safety Systems due to the Loss of ESW	1.0 E-002	1.0 E-002	NEW	Yes
ESW-CCF	Common-Cause Failure of the ESW	1.0 E-001	1.0 E-001	NEW	Yes
HCS-MDP-FC-TRAIN	High Pressure Core Spray (HPCS) Train Level Failures	1.3 E-002	1.3 E-002		No
HCS-XHE-XE-NOREC	Operator Fails to Recover HPCS	7.0 E-001	7.0 E-001		No
PPR-SRV-OO-2VLVS	One or Two Safety Relief Valves (SRVs) Fail to Close	3.2 E-002	3.2 E-002		No
RCI-TDP-FC-TRAIN	Reactor Core Isolation Cooling (RCIC) Train Components Failures	3.8 E-002	3.8 E-002		No
RCI-XHE-XE-NOREC	Operator Fails to Recover RCIC	7.0 E-001	7.0 E-001		No
RH-PC-LT	Operator Fails to Recover the Residual Heat Removal (RHR) System and the Power Conversion System (PCS) Over a 12 h Period	2.8 E-002	2.8 E-002		No
RHR-MDP-CF-MDPS	Common-Cause Failure of RHR Pumps	1.0 E-004	1.0 E-004		No

Table 1. Definitions and Probabilities for Selected Basic Events for LER No. 373/96-007, -008

Event name	Description	Base probability	Current probability	Type	Modified for this event
RHR-MDP-FC-TRNA	RHR Train A Components Fail	3.8 E-003	3.8 E-003		No
RHR-MDP-FC-TRNB	RHR Train B Components Fail	3.8 E-003	3.8 E-003		No
RHR-MOV-OO-BYPSA	RHR Heat Exchanger Bypass Valve Fails to Close	3.0 E-003	3.0 E-003		No
RHR-MOV-OO-BYPSB	RHR Heat Exchanger B Bypass Valve Fails to Close	3.0 E-003	3.0 E-003		No
SDC-XHE-XE-NOREC	Operator Fails to Recover RHR	1.0 E+000	1.0 E+000		No
SPC-XHE-XE-NOREC	Operator Fails to Recover RHR	1.0 E+000	1.0 E+000		No
SRV	One or More of the SRVs Fail to Close	3.2 E-002	3.2 E-002		No
SSW-MDP-FC-FPEMU	Failure of the Fuel Pool Emergency Makeup Pump Train	3.3 E-003	1.0 E+000		Yes

Table 2. Sequence Conditional Probabilities for LER 373/96-007, -008

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Percent contribution
TRANS-NESW	07	1.9E-005	83.9
TRANS-NESW	62	1.5E-006	6.4
TRANS-NESW	31	1.2E-006	5.1
TRANS-NESW	37	6.5E-007	2.7
Total (all sequences)		2.3E-005	

Table 3. Sequence Logic for Dominant Sequences for LER 373/96-007, -008

Event tree name	Sequence number	Logic
TRANS-NESW	07	/RPS, PCS, /SRV, MFW, /HCS, RHR, CVS
TRANS-NESW	62	/RPS, PCS, P2, HCS, ADS
TRANS-NESW	31	/RPS, PCS, /SRV, MFW, HCS, RCI, ADS, CRD
TRANS-NESW	37	/RPS, PCS, P2, /HCS, CDS, /LCS, RHR, CVS

Table 4. System names for LER 373/06-007, -008

System name	Logic
ADS	Automatic Depressurization Fails
CDS	Failure of the Condensate System
CRD	Insufficient CRD Flow to the RCS
CVS	Containment (Suppression Pool) Venting
HCS	HPCS Fails to Provide Sufficient Flow to the Reactor Vessel
LCS	Low Pressure Core Spray
MFW	Main Feedwater System
P2	One or Two SRVs Fail to Close
PCS	Power Conversion System Fails
RCI	RCIC Fails to Provide Sufficient Flow to the RCS
RHR	Residual Heat Removal Fails
RPS	Reactor Shutdown Fails
SRV	None of the SRVs Fail to Close

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER No. 373/96-007, -008

Cut set number	Percent Contribution	CCDP[a]	Cut sets[b]
TRANS-NESW Sequence 07		1.9 E-005	
1	68.8	1.3 E-005	IE-TRANS-NESW, /SRV, DIV1FOUL, ESW-CCF, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
2	6.8	1.3 E-006	IE-TRANS-NESW, /SRV, RHR-MDP-CF-MDPS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
3	6.8	1.3 E-006	IE-TRANS-NESW, /SRV, DIV1FOUL, DIV2FOUL, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
4	2.6	5.1 E-007	IE-TRANS-NESW, /SRV, RHR-MDP-FC-TRNA, DIV2FOUL, /HCS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
5	2.6	5.1 E-007	IE-TRANS-NESW, /SRV, DIV1FOUL, RHR-MDP-FC-TRNB, /HCS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
6	2.0	4.0 E-007	IE-TRANS-NESW, /SRV, DIV1FOUL, RHR-MOV-OO-BYPSB, /HCS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
7	2.0	4.0 E-007	IE-TRANS-NESW, /SRV, RHR-MOV-OO-BYPSA, DIV2FOUL, /HCS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
TRANS-NESW Sequence 62		1.5 E-006	
1	97.0	1.4 E-006	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, HCS-MDP-FC-TRAIN, HCS-XHE-XE-NOREC, ADS-XHE-XE-ERROR
TRANS-NESW Sequence 31		1.2 E-006	
1	96.6	1.1 E-006	IE-TRANS-NESW, /SRV, HCS-MDP-FC-TRAIN, HCS-XHE-XE-NOREC, RCI-TDP-FC-TRAIN, RCI-XHE-XE-NOREC, ADS-XHE-XE-ERROR

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER No. 373/96-007, -008

Cut set number	Percent Contribution	CCDP[a]	Cut sets[b]
2	1.9	2.3 E-008	IE-TRANS-NESW, /SRV, CDS-TNK-HW-CST, HCS-XHE-XE-NOREC, ADS-XHE-XE-ERROR
TRANS-NESW Sequence 37		6.5 E-007	
1	68.8	4.4 E-007	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, DIV1FOUL, ESW-CCF, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
2	6.8	4.4 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, RHR-MDP-CF-MDPS, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
3	6.8	4.4 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, DIV1FOUL, DIV2FOUL, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
4	2.6	1.7 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, RHR-MDP-FC-TRNA, DIV2FOUL, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
5	2.6	1.7 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, DIV1FOUL, RHR-MDP-FC-TRNB, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
6	2.0	1.3 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, DIV1FOUL, RHR-MOV-OO-BYPSB, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
7	2.0	1.3 E-008	IE-TRANS-NESW, PPR-SRV-OO-2VLVS, RHR-MOV-OO-BYPSA, DIV2FOUL, SDC-XHE-XE-NOREC, SPC-XHE-XE-NOREC, RH-PC-LT, CSS-XHE-XE-NOREC
Total (all sequences)		**2.3 E-005**	

[a]The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probabilities for the initiating events and the basic events are given in Table 1.

[b]Initiating events, such as IE-TRANS-NESW, are 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. Unavailability of the NESW initiates the transient and renders numerous systems that are dependent on service water unavailable, including the CDS, MFW, PCS, CRD, and CVS.

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

---

* 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.

Commonwealth Edison Company
LaSalle Generating Station
2601 North 21st Road
Marseilles IL 61341-9757
Tel 815-357-6761

ComEd

July 24, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #96-007-00, Docket #050-373 is being submitted to your office in accordance with 10 CFR 50.73(a)(2)(iv).

Respectfully,

for D. J. Ray
Station Manager
LaSalle County Station

Enclosure

cc: H. J. Miller, NRC Region III Administrator
M. P. Huber, NRC Senior Resident Inspector - LaSalle
C. H. Mathews, IDNS Resident Inspector - LaSalle
F. Niziolek, IDNS Senior Reactor Analyst
INPO - Records Center
DCD - Licensing (Hardcopy: Electronic:)

IE22/1

Enclosure 3

(5-92)

EXPIRES 05/31/95

LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1):	DOCKET NUMBER (2)	PAGE (3)
LaSalle County Station Unit One	05000373	1 of 5

TITLE (4)
Unit 1 Reactor Scram on Main Steam Flow High Trip Isolation during Surveillance

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	26	96	96	007	00	07	24	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100 %

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(iii)	☐ 73.71(b)
☐ 20.2203(a)(1)	☐ 20.2003(a)(3)(ii)	☒ 50.73(a)(2)(iv)	☐ 73.71(c)
☐ 20.2203(a)(2)(i)	☐ 20.2003(a)(4)	☐ 50.73(a)(2)(v)	☐ OTHER
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(viii)(A)	(Specify in Abstract below
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)(B)	and in Text, NRC Form 366A)
☐ 20.2003(a)(2)(v)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
William Kirchhoff, Site Engineering	(815) 357-6761 Extension 2927

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines 16)

At 20:56 hours on June 26, 1996, a full Main Steam Isolation Valve (MSIV) isolation was received on Unit 1 during the Instrument Maintenance Department (IMD) instrument mechanics (IM's) performance of a surveillance for calibrating the Main Steamline High Flow Isolation switches. The MSIV isolation trip resulted in an automatic Reactor Scram of Unit 1 due to the Reactor Protection System (RPS) trip signal from the MSIV not full open trip logic. All Safety systems functioned as expected and the reactor safely shut down.

A Primary Containment Isolation System (PCIS) Group 1 half isolation trip on the A2 channel was in place due to the calibration of the 1B21-N010C switch and had not been reset. The IM had just completed actions for prepressurizing the high flow differential pressure switch to near reactor pressure, and was in the process of throttling open the flow switch high side isolation valve when a trip was received from the PCIS Group 1 channel B2 for main steam flow high trip. The A2 instrument and B2 instrument lines are shared. The combination of the A2 channel and B2 Channel trips resulted in the full PCIS Group 1 isolation of the MSIVs.

The root cause of the event was an Instrument Maintenance Department work practice deficiency in the proper technique for prepressurizing instruments. A contributing factor was a procedural weakness. The procedure did not include steps to reset the main steam high flow isolation trip channel prior to returning the instrument to service. This action would have reduced the probability of receiving a full PCIS Group 1 isolation from a pressure spike induced while valving in the instrument.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): One	Event Date: 06/26/96	Event Time: 20:56 Hours
Reactor Mode(s): 1	Mode(s) Name: Run	Power Level(s): 100%

B. DESCRIPTION OF EVENT

At 20:56 hours on June 26 1996, the Instrument Maintenance Department (IMD) was performing Instrument Surveillance LIS-MS-102," Unit 1 Main Steam Line (MSL) High Flow MSIV Isolation Calibration." Ten of the sixteen Main Steam Isolation Valve (MSIV,MS)[SB] high flow differential pressure switches had already been successfully calibrated and returned to service. The calibration of the 1B21-N010C flow switch had just been completed and the instrument mechanics were in the process of returning the switch to service.

There are four instrument racks, and each instrument rack contains one Primary Containment Isolation System (PCIS, PC) [NH] subchannel. Each PCIS subchannel consists of four flow switches; one from each main steam line. These subchannels are A1 (A switches), B1 (B switches), A2 (C switches) and B2 (D switches). The switches are configured in a one-out-of-two twice logic. To trip a PCIS subchannel from the high flow switches, at least one switch in a channel must trip on high flow. To receive a Group 1 isolation, one of the A or C channels must trip along with one of the B or D channels.

A Primary Containment Isolation System (PCIS) Group 1 half isolation trip on the A2 channel was in place during the calibration of the 1B21-N010C switch and had not been reset. An Instrument Mechanic had just completed actions for prepressurizing the hi flow differential pressure switch to near reactor pressure, and was in the process of throttling open the flow switch high side isolation valve when a trip was received from the PCIS Group 1 channel B2 for Main steam flow hi trip. The combination of the A2 channel and B2 Channel trips resulted in the full PCIS Group 1 isolation of the MSIVs. The MSIV isolation trip resulted in an automatic Reactor Scram of Unit 1. All control rods fully inserted and Operations established control of Reactor Vessel water level and pressure using the Motor Driven Feed Pump (MDRFP) and Safety Relief Valves (SRVs). Both loops of suppression pool cooling were started to remove the added heat to the suppression pool from the SRVs and the Reactor Core Isolation System (RCIC) was manually started to assist in pressure control of the reactor vessel.

This event is being reported in accordance with the requirements of 10 CFR 50.73(a)(2)(iv) due to an automatic actuation of an Engineered Safety Feature (ESF).

C. CAUSE OF EVENT

The cause of the Scram was a Group 1 Isolation signal initiated by the spurious trip of Main Steam Line (MSL) High Flow switch 1B21-N010D caused by a pressure spike on the instrument sensing line during the return to service of the 1B21-N010C switch. The instrument sensing lines of the 1B21-N010C switch (PCIS Group 1 A2 channel) are shared with the 1B21-N010D switch (PCIS Group 1 B2 channel) and with a Feedwater-Reactor level control system main steam line flow transmitter (1C34-N004C). The pressure spike occurred because the main steam high flow switch did not get properly prepressurized prior to returning the instrument to service. The PCIS Group 1 Isolation trip of the A2 channel had not been reset and the subsequent B2 Channel trip of the PCIS Group 1 logic resulted in a full Group 1 isolation.

The manifold and vent valves of the main steam hi flow switch, 1B21-N010C, which was being returned to service were subsequently replaced and tested. It was suspected that one of the valves may have been leaking or sticking in a manner to result in an abrupt valve position change. It was determined by testing the manifold and vent valves as they responded to maintaining pressure, that none of the valve seats were leaking. However, the hi side vent valve required a greater amount of rotation than normal before valve seat contact was broken as evidenced by dropping of pressure being held. The high side vent had been the valve used by the IM to prepressurize the flow switch.

During the prepressurization of the flow switch, the IM throttled the high side vent valve. This was done in order to minimize the effect on the final switch pressure from the closure of the vent valve which causes further pressurization of the switch due to the relative incompressibility of the water. Because the valve was being throttled near the fully closed position, the valve was not open off its seat. As a result, the IM was not actually pressurizing the instrument volume, but only the gage which was attached upstream of the high side vent. This method of prepressurizing instruments was verified to be used by other Instrument Mechanics. The root cause is a work practice deficiency in the proper technique for prepressurizing instruments.

A contributing factor was a procedural weakness. The procedure did not include steps to reset the main steam high flow isolation trip channel prior to returning the instrument to service. This action would have reduced the probability of receiving a full PCIS Group 1 isolation from a pressure spike induced while valving in the instrument.

LIS-MS-102, Unit 1 Main Steam Line High Flow MSIV Isolation Calibration, instructs the IM to have the Operator reset the PCIS isolation trip logic after the switch being tested has been valved back in to service. It is however possible to reset the isolation logic prior to valving the instrument into service because the flow switch is reset with the manifold equalization valve open. Had the isolation trip been reset prior to instrument being valved back in service, it is possible that only a B2 channel trip would have occurred as a result of the spike from the 1B21-N010D switch. The 1B21-N010C switch may not have tripped causing a channel A2 trip, because the sensed differential pressure across the instrument with the equalizing valve open would still have been lower than the spike on the adjoining switch.

D. ASSESSMENT OF SAFETY CONSEQUENCES

The safety significance of this event was minimal. The Primary Containment Isolation System functioned as designed when the high flow isolation signal was received. The positive reactivity due to pressure increase resulting from the closure of the MSIVs is reduced by the initiation of the reactor scram upon start of MSIV closure. A failure of the scram from MSIVs closing would result in a scram from either the Reactor Pressure High scram trip or from the Average Power Range Monitor high flux trip logic. However, the scram and Group 1 isolation do represent a significant challenge to safety related equipment which should be minimized. All PCIS and RPS actions were initiated and completed as designed.

E. CORRECTIVE ACTIONS

Instrument Maintenance Department personnel will receive additional training on proper techniques to be used in valving instruments back into service. The training will emphasize the risks involved when instruments are not properly prepressurized and appropriate precautions which must be taken. The Control System Technicians will be trained by October 1, 1996. The "A" IMs will be trained by February 1, 1997.

A procedure change to LIP-GM-909, "Opening Process Instrument Lines and Valve Manipulation," is being implemented to incorporate the information provided during the training sessions described above. The procedure will be revised by December 31, 1996.

A procedure change is being implemented to the Unit 1 and Unit 2 calibration procedures, LIS-MS-102 (202) to reset the isolation trip prior to valving the instrument flow switch back in service. Electrically defeating the trip from the instrument being returned to service is also being evaluated as a part of this procedure revision. Completion of this procedure change is planned prior to the performance of the procedure while at power. LIS-MS-102 and LIS-MS-202, will be revised before August 15, 1996.

F. PREVIOUS OCCURRENCES

LER NUMBER	TITLE
373/94-015-00	Unit 1 Primary Containment Isolation and SCRAM Due to Switch Failure

In the referenced LER, a PCIS Group 1 isolation of the main steam hi flow trip logic occurred. At the time of the investigation, the tripping of one of the high flow switches was believed to have spuriously occurred due to a contact resistance problem on the microswitch of the Static O Ring (SOR) switch. Foreign material found on the switch contact caused the flow switch to have erratic calibration settings. This was not believed to be the problem in the recent event because the calibration settings of the B2 channel switches were not erratic or abnormal. It is however possible that the cause of the previous event was due to inadequate prepressurization of the flow switch, and not from the erratic operation of the switch.

G. COMPONENT FAILURE DATA

Since no component failure occurred, this section is not applicable.

Commonwealth Edison Company
LaSalle Generating Station
2601 North 21st Road
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Tel 815-357-6761

ComEd

July 28, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #96-008-00, Docket #050-373 is being submitted to your office in accordance with 10 CFR 50.73(a)(2)(i).

Respectfully,

D. J. Ray
Station Manager
LaSalle County Station

Enclosure

cc: H. J. Miller, NRC Region III Administrator
M. P. Huber, NRC Senior Resident Inspector - LaSalle
C. H. Mathews, IDNS Resident Inspector - LaSalle
F. Niziolek, IDNS Senior Reactor Analyst
INPO - Records Center
DCD - Licensing (Hardcopy: Electronic:)

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 05/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
LaSalle County Station Unit One	05000373	1 of 7

TITLE (4) Foreign Material Injected Into ervice Water Tunnel Causes Dual Unit Shutdown Due t Inadequate Work Control

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									LaSalle County Station Unit Two	05000374
06	28	96	96	008	00	07	28	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 0

POWER LEVEL (10): 001

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(iii)	☐ 73.71(b)
☐ 20.2203(a)(1)	☐ 20.2003(a)(3)(ii)	☐ 50.73(a)(2)(iv)	☐ 73.71(c)
☐ 20.2203(a)(2)(i)	☐ 20.2003(a)(4)	☐ 50.73(a)(2)(v)	☐ OTHER
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(vii)	(Specify in Abstract below and in Text, NRC Form 366A)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.2203(a)(2)(iv)	☒ 50.73(a)	☐ 50.73(a)(2)(viii)(B)	
☐ 20.2003(a)(2)(v)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Dennis Tristave, System Engineering	(815) 357-6761 Extension 2081

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☒ YES (If yes, complete EXPECTED SUBMISSION DATE)	☐ NO		12	01	96

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 2315 hours on June 28, 1996, with Unit 1 at 1% power and Unit 2 at 100% power, the station declared all Core Standby Cooling Systems (CSCS), Emergency Core Cooling Systems (ECCS) and Diesel Generators (DG) inoperable due to foreign material identified on the floor of the service water tunnel. The tunnel is the source for the Essential Service Water System, Non-essential Service Water Systems and the Fire Protection systems. The foreign material had the potential to cause a common mode failure of the Essential Service Water System. Although the systems were declared inoperable, they were available. The foreign material was an injectable sealant foam substance which had been used since May, 1996, in the Lake Screen House (LSH) to seal water seepage cracks in a portion of the floor of the building (the ceiling of the service water tunnel). After the units were shutdown, sealant material was removed from the tunnel. Systems and components which would have been affected by the foreign material were inspected, cleaned and tested to verify operability prior to returning the units to service.

The cause of the event was a breakdown in the procedural and work control process. The LSH crack repair had been incorrectly classified as minor facility maintenance and had not been adequately reviewed as a nuclear work request. Weaknesses in the policies and procedures for control of work of minor maintenance actions were identified and corrective actions scheduled.

NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 05/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
LaSalle County Station Unit One	05000373	96	008	00	2 of 7

(If more space is required, use additional copies of NRC Form 366A)(17)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 1/2	Event Date: 06/28/96	Event Time: 23:15 Hours
Reactor Mode(s): 2/1	Mode(s) Name: Startup/Run	Power Level(s): 001/100%

B. DESCRIPTION OF EVENT

On May 21, 1996, workers began sealing cracks in the walls and floors of the Lake Screen House with an injectable sealant. This work was being done to stop ground water inleakage and prepare the building for painting. The work continued through June 21, 1996. The activity, evaluated to be minor maintenance work, was performed under an action request by the Station's Consolidated Facility Maintenance(CFM) group using contractors experienced with this type repair.

The process for performing the crack repair normally required drilling holes on each side of a crack along its length and injecting an expandable sealant into these holes to seal the crack. Cracks of 3 feet to 9 feet long were being repaired. Normally, this process would not require any drilling through the wall or floor. However, if a void was found, as indicated by excessive amounts of water, then the practice was to drill through near the crack and inject to fill the void. The sealant used would expand and block further water intrusion.

While doing floor repairs, the workers started fixing cracks on the top or ceiling of a service water tunnel which runs the length of the building approximately 20 feet below lake level (see sketch). This tunnel supplies cooling water to both the non-essential and essential(safety-related) cooling water pumps at the plant. As they repaired these cracks, the large amount of water at pressure indicated to the workers that a large void was present. The workers believed that they were working on a concrete floor laid over soil. They proceeded to drill five holes through the ceiling of the service water tunnel and inject sealant. Instead of being injected into a void under the building floor, the material was injected into the tunnel accumulating on the ceiling and floor or dispersing into the cooling water.

On June 19, 1996 with both units at approximately full power, high differential pressure occurred on the on-line non-essential service water strainers(WS)(KG). Operators also observed that service water header pressure had decreased below normal. Upon inspection, two of the three strainers were found in automatic backwash but failures on the backwash valve actuators and/or binding of the strainer basket diverters prevented proper flushing of accumulated material. Power reductions were done on both units to approximately 850 MWe to reduce the service water heat loads and isolate each strainer, one at a time, to repair the valve actuators and free the diverter. Following this, the operators were able to manually backwash each strainer successfully. The initial investigation into what caused the high differential pressures identified that "corn cob" material used for sandblasting the exterior of the Lake Screen House was the potential problem. A large amount of this material was along the outside walls of the building and it was postulated that the material could have gotten into the water and been pulled into the strainers. Operators w e stationed to periodically backwash the strainers and a contingency

established to trip the units if service water header pressure could not be maintained.

On June 24, high strainer differential pressure occurred again on the on-line non-essential service water strainers. Prior to this, Operating had been conducting normal surveillance tests on the 0A and 0B Diesel Fire pumps (FP) [KP]. These pumps take their suction from the service water tunnel. After the 0A pump was tested satisfactorily, the 0B pump test was unsuccessful when the pump had to be stopped five minutes into it's run due to high cooling water temperature, an indication of a possible flow blockage. The 0B and the 0A Diesel Fire Pumps were declared inoperable at 0927 hours and Unit 1 and 2 were reduced in power to 850 MWe to reduce cooling water demand.

Based upon the observation of sealant material in the trash basket which collects the strainer backwash water and through discussions with the workers performing crack repair in the Lake Screen House, it was now concluded that sealant material was the probable cause of the problems with the strainers and fire pumps, not the corn cob material. When the strainers were again backwashed, it became clear from the material collected in the flush water that sealant material was the predominant foreign material. Chemical analysis of samples collected from several locations including the 0B Diesel Fire Pump confirmed this. These samples did not reveal any "corn cob" material.

Along with this evaluation, further inspections of the tunnel and strainers were scheduled. In addition, the Residual Heat Removal (RHR) [BO] and Diesel Generator Service Water Systems were run to verify operability with satisfactory results. An Operability Evaluation was performed. Information was obtained from the vendor as to the expected behavior of the sealant when injected into the service water tunnel. This information indicated that the sealant would expand in the tunnel and that the resulting mass would float. Based on this, the evaluation concluded that there was no risk to the Essential Service Water System. This was determined by the fact that the material floated, that the suction points from the service water tunnel were relatively low in the tunnel, and given the velocity of water in the tunnel, it was not probable that floating material would be drawn into the pumps suction supply lines. A compensatory action taken was to bring two diesel fire pumper trucks on site to back up the non-essential service water pumps as the source of water for fire suppression. This equipment was manned on a round the clock basis. This action was taken because the diesel fire pumps had previously been declared inoperable.

On June 25, inspections in the Service Water Tunnel were started using divers and, later, robotic inspection equipment. Due to diver safety considerations, the first inspection was restricted to the area around the bottom of the ladder in the Service Water Tunnel. The diver inspection was delayed one day because of a unit 1 SCRAM on June 26. The SCRAM was not related to the sealant material in the service water tunnel. Divers with robotic equipment reentered the Service Water Tunnel on June 27. They noticed sealant material attached at the top of the Service Water Tunnel but did not find sealant on the tunnel floor. On June 27, a Unit 2 service water strainer was inspected. This strainer is supplied by pumps close to where the sealant was injected. There were no signs of plugging and no signs of damage. Using this information together with the successful surveillance tests, the Operability Evaluation was completed on June 28, 1996 indicating that Essential Service Water systems were operable. Later that day divers encountered sealant material on the floor of the tunnel. The discovery that the sealant did not float invalidated the Operability Evaluation and as a result the Essential Service Water

systems Core Standb[illegible]ing System [VF], Emergency Core Cooling System, and Diesel Generator Cooling Water System were declared inoperable at 2315 hours on June 28, 1996. Unit 1 was at approximately 1% core thermal power in the startup mode at the time and Unit 2 was at 100% core thermal power. To comply with Technical Specification 3.0.3, Unit 1 was manually scrammed at 2338 and Unit 2 was reduced in power to 5% and manually scrammed at 0528 on June 29, 1996.

The station requested a Notice of Enforcement Discretion (NOED) to allow both Units to remain in Hot Shutdown until July 9, 1996, to maximize decay heat removal capability and minimize the probability of the material entering the essential service water system while cleanup and removal of the material were performed. This request was granted.

Extensive inspection and testing of plant equipment were performed to identify the presence of sealant material. Approximately eighty cubic feet of the material was removed, mostly from the tunnel. Some material was found in the essential service water equipment including large pieces in one of the residual heat removal service water strainers (Unit 2, Division I). Following successful cleanup and testing of affected equipment, the units were returned to service.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(i) because of entry into Technical Specifications 3.0.3.

C. **CAUSE OF EVENT**

The root cause of this event is that work affecting plant safety related structures was assigned and performed outside the controls of the Nuclear Work Request Process. This occurred when the work was approved without identifying a potential impact on the Seismic Category 1 Service Water Tunnel or the Service Water System from the sealant injection process used in concrete crack repairs and resulted in the work not being reviewed by Engineering. The sealant work being performed was considered "Materiel Condition" repair and incorrectly assumed to be non-intrusive. The workers were using "craft capability" to perform the sealant work, no Work Package was generated for the work. Two Action Requests (ARs) were used as the authorization for the repairs. Crack repair work was performed on walls as well as the floor at the Lake Screen House but also on the Seismic Category 1 Service Water Tunnel.

D. **ASSESSMENT OF SAFETY CONSEQUENCES**

This event resulted in degradation of station Non-Essential and Essential Service Water Systems and the Fire Protection Systems. The degradation occurred due to the presence of injectable sealant material which was free to move within the Service Water Tunnel. Since the material was free to move, in the event that Essential Service Water Pumps were required and had started, a loss of Essential Service Water could have occurred. In this case, the event is Safety Significant, and resulted in increased risks to the facility. The injected material created the potential for a common mode failure of Essential Service Water and Fire Protection Systems. However, the actual consequences of the event were minimal because the both units were maintained in a hot shutdown condition which did not require the use of essential service water pumps. By not operating these pumps, significant amounts of foreign material were not pulled into these systems. In addition, the residual heat removal and diesel generator service water systems were tested during the event with satisfactory results.

E. CORRECTIVE ACTIONS

1. Floor and wall repair work in the Lake Screen House was stopped.

2. The service water tunnel was cleaned and inspected.

3. Essential Service Water Systems and balance of plant equipment that could have been impacted by the intrusion of sealant were inspected, cleaned and tested to ensure that the systems and equipment would function as designed.

4. The structural integrity of the Seismic-Class I service water tunnel was evaluated to ensure that repair activities had not degraded the structure.

5. An Engineering Policy was published which outlined repair and controls associated with building structural repairs and use of sealants as a temporary or permanent repair requiring engineering review and approval.

6. Procedures and guidance documents were revised to ensure consistent direction is provided to personnel when assigning, preparing, and supervising work of this nature (sealing). The following documents were revised: LAP-1300-1, Work Request Procedure; LAP-240-6, Temporary Alteration Procedure; and Maintenance Memo 600.4, Use of Furmanite. Specifically, all "sealant" activities are now required to go through the Nuclear Work Request Process.

7. Appropriate Maintenance, Operations, and CFM personnel were coached on the expectations regarding sealant type work and the new engineering policy.

8. The Action Request (AR) screening review process has been changed to include multi-discipline involvement.

9. The Station has prepared and implemented a Consolidated Facilities Maintenance (CFM) Responsibilities Administrative Controls document to clearly delineate work scope boundaries and process limitations of the CFM organization. Guidance was provided to clearly outlines supervisory responsibilities for acceptance of work assignments within the area. This document al.o provided clarification of the expectation that Maintenance LAPs and Memo's application to the CFM organization.

10. Clear identification (by signs or other measures) of the operational safety significance of the Service Water Tunnel and Lake Screen House floor has been completed.

11. A Corporate investigation team has been established to independently review this event and the Station's response to the event including the investigation and evaluation of conservative decision making, unexpected conditions and the resolution of problems. Corrective actions will be issued to institutionalize the lessons learned and apply them to all ComEd sites. This action will be completed by October 1, 1996. The details of this investigation will be provided in a supplemental report.

12. An EPN assigned coding or equivalent system to accurately and easily determine structures and non-system specific components which may be safety, seismic, or regulatory related will be developed. This system will be part of the Q-List

(or other engineering controlled document) and EWCS data base. This action will enable simple and accurate decision making when performing screening of new work through the EWCS System. This will be completed by October 1, 1996.

F. PREVIOUS OCCURRENCES

LER NUMBER **TITLE**

None

G. COMPONENT FAILURE DATA

Since no component failure occurred, this section is not applicable.

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

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EXPIRES 05/31/95

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

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(If more space is required, use additional copies of NRC Form 366A)(17)

Lake Screen House Cross-section View