50-370

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

August 14, 1997

Mr. H. B. Barron
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF OPERATIONAL CONDITION AT MCGUIRE NUCLEAR STATION, UNIT 2

Dear Mr. Barron:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational condition, which was discovered at McGuire Nuclear Station, Unit 2, on March 6, 1996 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 370/96-002. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORNL). The results of this preliminary analysis indicate that this 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 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.

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

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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. 370/96-002, which documented the event.

Please contact me at (301) 415-1484 if you have any questions regarding this request. This request is covered by the existing OMB clearance number (3150-0104) for NRC staff followup review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

Herbert N. Berkow

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-370

Enclosures: As stated (3)

cc w/encls: See next page

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ORIGINAL SIGNED BY H. BERKOW FOR:

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Office of Nuclear Reactor Regulation

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Units 1 and 2

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LER No. 370/96-002

Event Description: 2B Emergency Diesel Generator Inoperable due to Slow Instrumentation Response

Date of Event: March 6, 1996

Plant: McGuire Unit 2

Event Summary

McGuire Unit 2 was at 100% power when the 2B Emergency Diesel Generator (EDG), which was undergoing a scheduled operating test, tripped on a low lube oil pressure signal shortly after starting (Ref. 1). The test failure was the result of air entrainment into the instrument line for the lube oil piping combined with low room temperature. Personnel determined that these conditions (air ingress and cold room temperature), which were deemed sufficient to cause the 2B EDG to trip, existed for a combined total of 540 h. (The 540 h total was distributed over four separate occasions where the 72 h single EDG outage allowed by Technical Specifications was exceeded.) This long-term unavailability of the 2B EDG could have impacted the units' response to a loss-of-offsite power (LOOP). The estimated increase in the core damage probability (CDP) over the 540-hour period for this event (i.e., the importance) is 4.3×10^{-6}. The base probability of core damage (the CDP) for the same period is 1.7×10^{-6}.

Event Description

Unit 2 was at 100% power on February 6, 1996. The 2B EDG was scheduled for a non-prelubricated start test. The 2B EDG reached 95% of rated speed in 9 s (Ref. 2). The 2B EDG tripped on a "Low Lube Oil Pressure" signal 30 s later (39 s after starting the EDG). Indicated pressure was 15-20 psig and decreasing; normal operating pressure is 40 psig. However, personnel determined that the low lube oil pressure indication was false. The low pressure indication resulted from a slow instrument response due to air entrainment into the instrument line for the lube oil piping, coupled with the low EDG room temperature. (An inadequate design of the instrument lines allowed for air to be introduced into the system. The lube oil pressure switch impulse line for the 2B EDG is approximately 70 ft long. The licensee indicated that this length is considered excessive.) The cool EDG room temperature added to the slow instrument response by increasing the viscosity of the oil in the instrument line. Since the "Low Lube Oil Pressure" trip signal is not bypassed on an emergency start of the EDGs, the failure was classified as a valid test failure.

The lowest recorded EDG room temperature in the 7 d preceding the EDG failure to start was 62°F. EDG room temperature was 68°F just prior to the test. On March 6, 1996, the licensee determined that the 2B EDG should be considered to be inoperable with the current instrument line configuration when the EDG room temperature is less than 71°F and the "Before and After" (B&A) lube oil pump is not running. Based on this criteria, all other station EDGs were determined to be operable at the time the 2B EDG failed its operating test. Based on a review of the log books containing the EDG room temperature readings, the licensee calculated that the 2B EDG was susceptible to this type of failure for a total of 666 h. Because the

Enclosure 1

B&A lube oil pump runs for 15 min during each hour, the licensee estimated that the 2B EDG was susceptible to this type of failure only 75% of the time–a total of 499.5 h. NRC inspectors, in NRC Inspection Report 50-370/96-02 (Ref. 2), noted that previous EDG trips occurred while the B&A lube oil pump was running. Therefore, the NRC inspectors discounted the assumption that running a B&A lube oil pump at the time of a start demand with the EDG room temperature below 71°F would have prevented this type of failure of the EDG to start. The NRC inspection report tallied the amount of time above 72 h, per occurrence, that the room temperature dropped below 71°F and determined that the four susceptibility periods totaled 540 h.

Additional Event-Related Information

McGuire Nuclear Station maintains a Safe Shutdown Facility (SSF) designed to provide an alternate and independent means to achieve and maintain hot standby conditions (Ref. 3). The facility includes an EDG that can be utilized to operate a positive displacement pump to supply seal injection water to the reactor coolant pump (RCP) seals, preventing a RCP seal loss-of-coolant accident (LOCA). Credit for the SSF is included in the ASP models via the LOOP initiating event frequency, operator nonrecovery probabilities, and the RCP seal failure probability.

The most important recovery action with respect to this condition assessment is the possibility of restoring ac power to Unit 2 from Unit 1 via a cross-tie, given a station blackout at Unit 2. Because procedures exist detailing this operation, it is considered to be a viable option. Recovery via the cross-tie is included in the LOOP recovery probabilities discussed below.

There was a brief period (5.3 h) when both EDGs were technically out of service due to maintenance activities on Motor Control Center 1EMXH-1, which affected ventilation. The 2A EDG was functionally available and would have been able to perform its design function. Technical Specifications allow both EDGs to be out of service for up to 8 h.

Modeling Assumptions

Similar to the licensee's analysis of this event (Ref. 1), the failure probability of the 2B EDG was set to 1.0 (TRUE) for this condition assessment.

The licensee indicated that if an actual failure to start occurred under circumstances similar to the conditions that existed since February 6, then a second start attempt would likely be successful (Ref. 1). Therefore, the emergency power non-recovery probability (EPS-XHE-NOREC) was adjusted from 0.8 to 0.34, as shown in Table 1, to reflect the fact that the equipment appeared recoverable and was accessible (Recovery Class 2).

The 2B EDG failure appears to be a failure mode unique to the physical setup of the lube oil pressure instrumentation lines on the 2B EDG. A similar failure of the 2A EDG was documented by special report 25 months earlier (Ref. 4). The length of time between events and consequently, the number of successful surveillance tests between events, indicates that the two failures were random rather than because of any common-cause effects. Consequently, the common-cause failure probability for the EDGs was not adjusted from the nominal value of 1.1×10^{-3} shown in Table 1.

During the 5-h period that both EDGs were declared unavailable, the 2A EDG was functionally available and would have been able to perform its design function. This 5-h period was not considered separately when calculating the increase in the CDP over the entire 540-h period because the importance (i.e., the increase in the CDP) is less than the ASP cut-off value of 1.0×10^{-6}

The possibility of preventing a seal LOCA using the SSF and the possibility of providing ac power via the cross-tie are factored into the following LOOP parameters:

IE-LOOP	LOOP Initiating Event Frequency, which includes short-term recovery actions, including cross-tieing between units
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 h
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6h
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Battery Depletion
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power During Seal LOCA
RCS-MDP-LK-SEALS	RCP Seals Fail.

The probability of SSF failure is 0.36 based on information in the plants' *Individual Plant Examination* (Ref. 5). It was assumed that at least 30 min would be required to staff the SSF. It was also assumed that personnel could cross-tie the power buses at Unit 1 with the buses at Unit 2 in under 1 h 5% of the time, and within 2 h 95% of the time. The LOOP parameters were then calculated using a lognormal distribution for the SSF failure probability; and a Weibull distribution for the LOOP initiating event frequency, the operator non-recovery probabilities, and the RCP seal failure probability [per ORNL/NRC/LTR-89/11 (Ref. 6)]. [The LOOP initiating event frequency accounts for the failure to staff the SSF within 30 min, as well as the failure of the SSF itself. Hence, the failure of the SSF is not specifically indicated in the results.]

Analysis Results

The increase in the CDP (i.e., the importance) over a 540-h period for this event is 4.3×10^{-6}. This is over the nominal CDP of 1.7×10^{-6}. The dominant core damage sequence for this event (sequence 28 on Fig. 1) involves

- a postulated LOOP,
- a successful reactor trip,
- a failure of emergency power,
- success of the auxiliary feedwater (AFW) system,
- no challenge to the power operated relief valves (PORVs),
- a RCP seal LOCA, and
- a failure of the operators to restore offsite power.

This sequence accounts for 29% of the total contribution to the increase in the CDP. Sequence 37 is similar to LOOP sequence 28, except LOOP sequence 37 involves a PORV lift and successful re-closure. Combined, these two sequences account for 46% of the total contribution to the increase in the CDP (Table 2). Core damage in these two sequences is the result of a RCP seal LOCA. Core damage results from a failure of

AFW in one other sequence (27% of the increase in the CDP) and results from battery depletion in two additional sequences (20% of the increase in the CDP).

Acronyms

AFW	auxiliary feedwater system
B&A	before and after lube oil pump
CCDP	conditional core damage probability
CDP	core damage probability
EDG	emergency diesel generator
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
NRC	Nuclear Regulatory Commission
PORV	power-operated relief valve
PWR	pressurized water reactor
RCP	reactor coolant pump
SGTR	steam generator tube rupture
SLOCA	small-break LOCA
SSF	safe shutdown facility
TRANS	transient

References

1. LER 370/96-002, Rev. 0, "Past Inoperability of Emergency Diesel Generator 2B Due to Low Lube Oil Pressure Caused by Unanticipated Interaction of Systems and Components" March 29, 1996.

2. NRC Inspection Report No. 50-370/96-02, Inspection Conducted: March 11, 1996 - April 1, 1996.

3. *Final Safety Analysis Report*, McGuire Nuclear Station.

4. Special Report 94-01, "Diesel Generator Special Report," Duke Power Company, McGuire Nuclear Station, PIP 2-M94-0242, March 15, 1994.

5. McGuire Nuclear Station, *Individual Plant Examination*.

6. ORNL/NRC/LTR-89/11, "Revised LOOP Recovery and PWR Seal LOCA Models", August 1989.

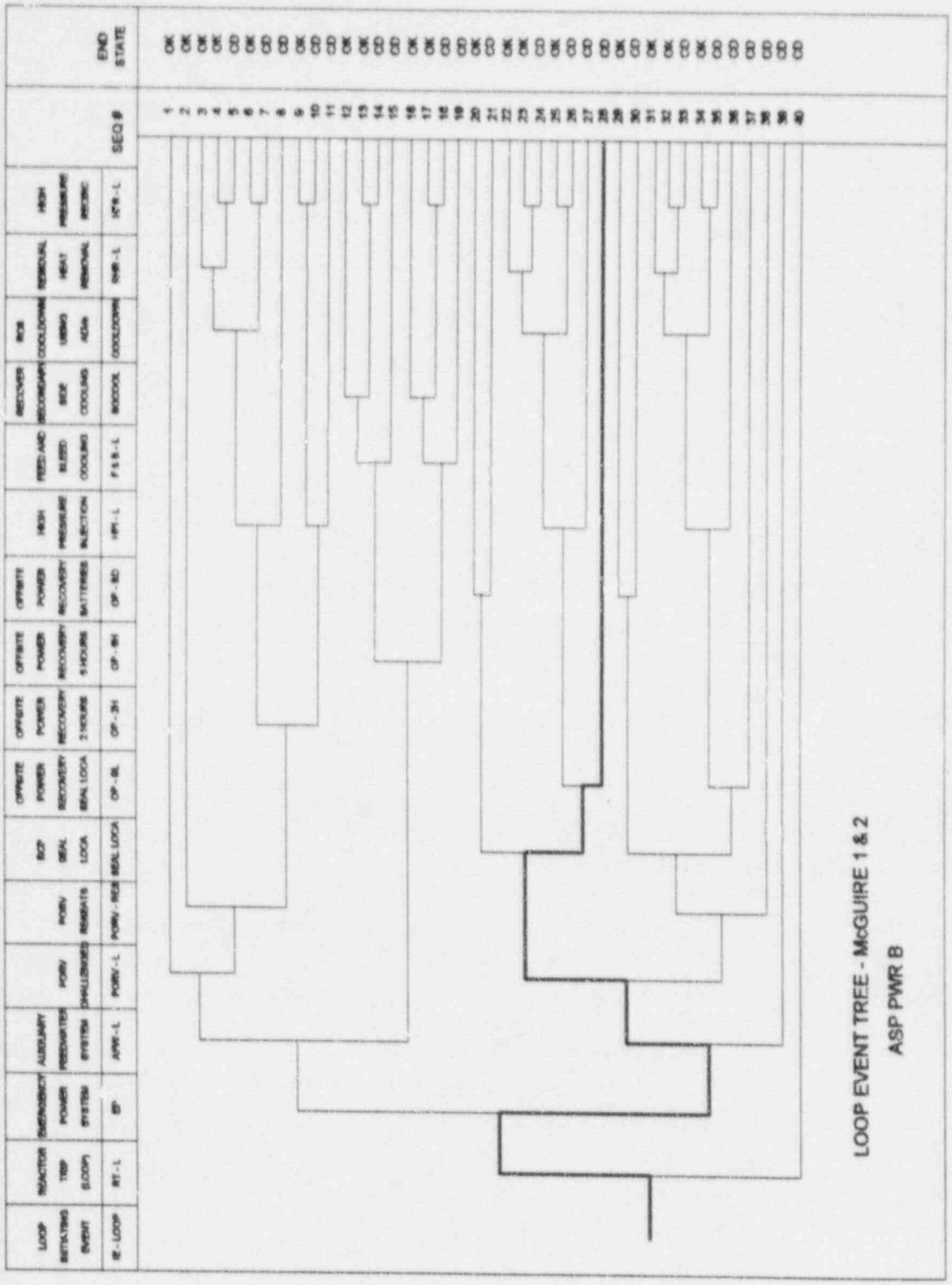

Fig. 2 Dominant core damage sequence for LER No. 370/96-002.

Table 1. Definitions and Probabilities for Selected Basic Events for LER No. 370/96-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–Loss-of-Offsite Power (LOOP)	1.6 E-005	1.6 E-005		No
IE-SGTR	Initiating Event–Steam Generator Tube Rupture	1.6 E-006	1.6 E-006		No
IE-SLOCA	Initiating Event–SLOCA	1.0 E-006	1.0 E-006		No
IE-TRANS	Initiating Event–Transient (TRANS)	5.3 E-004	5.3 E-004		No
AFW-TDP-FC-1A	AFW Turbine-Driven Pump Fails	3.2 E-002	3.2 E-002		No
AFW-XHE-NOREC-EP	Operator Fails to Recover AFW During a Station Blackout	3.4 E-001	3.4 E-001		No
EPS-DGN-CF-ALL	Common-Cause Failure of EDGs	1.1 E-003	1.1 E-003		No
EPS-DGN-FC-1A	Diesel Generator A Fails	4.2 E-002	4.2 E-002		No
EPS-DGN-FC-1B	Diesel Generator B Fails	4.2 E-002	1.0 E+000	TRUE	Yes
EPS-XHE-NOREC	Operator Fails to Recover Emergency Power	1.0 E+000	3.4 E-001		Yes
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 h	7.6 E-002	7.6 E-002		No
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 h	3.6 E-002	3.6 E-002		No
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Battery Depletion	8.5 E-003	8.5 E-003		No
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power During a Seal LOCA	5.4 E-001	5.4 E-001		No
PPR-SRV-CO-SBO	PORVs Open During a Station Blackout	3.7 E-001	3.7 E-001		No
PPR-SRV-OO-PRV1	PORV 1 Fails to Reclose	2.0 E-003	2.0 E-003		No
PPR-SRV-OO-PRV2	PORV 2 Fails to Reclose	2.0 E-003	2.0 E-003		No
PPR-SRV-OO-PRV3	PORV 3 Fails to Reclose	2.0 E-003	2.0 E-003		No
RCS-MDP-LK-SEALS	RCP Seals Fail Without Cooling and Injection Water	3.4 E-002	3.4 E-002		No

Table 2. Sequence Conditional Probabilities for LER No. 370/96-002

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP-CDP)	Percent contribution[a]
LOOP	28	1.5 E-006	3.0 E-007	1.2 E-006	29.2
LOOP	39	1.4 E-006	2.8 E-007	1.1 E-006	27.1
LOOP	37	8.9 E-007	1.7 E-007	7.1 E-007	17.0
LOOP	2[illegible]	6.6 E-007	1.3 E-007	5.3 E-007	12.8
LOOP	30	3.9 E-007	7.7 E-008	3.1 E-007	7.4
LOOP	38	2.8 E-007	5.6 E-008	2.3 E-007	5.5
Total (all sequences)		**6.0 E-006**	**1.7 E-006**	**4.3 E-006**	

[a]Percent contribution to the total importance.

Table 3. Sequence Logic for Dominant Sequences for LER No. 370/96-002

Event tree name	Sequence number	Logic
LOOP	28	/RT-L, EP, /AFW-L-EP, /PORV-SBO, SEALLOCA, OP-SL
LOOP	39	/RT-L, EP, AFW-L-EP
LOOP	37	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, SEALLOCA, OP-SL
LOOP	21	/RT-L, EP, /AFW-L-EP, /PORV-SBO, /SEALLOCA, OP-BD
LOOP	30	/RT-L, EP, /AFW-L-EP, PORV-SBO, /PORV-EP, /SEALLOCA, OP-BD
LOOP	38	/RT-L, EP, /AFW-L-EP, PORV-SBO, PORV-EP

Table 4. System Names for LER No. 370/96-002

System name	Logic
AFW-L-EP	No or Insufficient AFW Flow During a Station Blackout
EP	Failure of Both Trains of Emergency Power
OP-BD	Operator Fails to Recover Offsite Power Before Battery Depletion
OP-SL	Operator Fails to Recover Offsite Power During a Seal LOCA
PORV-EP	PORVs Fail to Reclose (No Electric Power)
PORV-SBO	PORVs Open During a Station Blackout
RT-L	Reactor Fails to Trip During a LOOP
SEALLOCA	RCP Seals Fail During a LOOP

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER No. 370/96-002

Cut set number	Percent contribution	CCDP[a]	Cut sets[b]
LOOP Sequence 28		1.5 E-006	
1	97.4	1.5 E-006	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, /PPR-SRV-CO-SBO, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 39		1.4 E-006	
1	96.8	1.3 E-006	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, AFW-TDP-FC-1A, AFW-XHE-NOREC-EP
LOOP Sequence 37		8.9 E-007	
1	97.4	8.4 E-007	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, /PPR-SRV-OO-PRV1, /PPR-SRV-OO-PRV2, /PPR-SRV-OO-PRV3, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 21		6.6 E-007	
1	97.4	6.4 E-007	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, /PPR-SRV-CO-SBO, /RCS-MDP-LK-SEALS, OEP-XHE-NOREC-BD
LOOP Sequence 30		3.9 E-007	
1	97.4	3.7 E-007	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, /PPR-SRV-OO-PRV1, /PPR-SRV-OO-PRV2, /PPR-SRV-OO-PRV3, /RCS-MDP-LK-SEALS, OEP-XHE-NOREC-BD
LOOP Sequence 38		2.8 E-007	
1	32.5	9.1 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-PRV1
2	32.5	9.1 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-PRV2
3	32.5	9.1 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PPR-SRV-CO-SBO, PPR-SRV-OO-PRV3
Total (all sequences)		**6.1 E-006**	

[a]The CCDP 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 (540 h). This approximation is conservative for precursors made visible by the initiating event. The frequencies of interest for this event are:

$\lambda_{TRANS} = 5.3 \times 10^{-4}/h$, $\lambda_{LOOP} = 1.6 \times 10^{-5}/h$, $\lambda_{SLOCA} = 1.0 \times 10^{-6}/h$, and $\lambda_{SGTR} = 1.6 \times 10^{-6}/h$. The importance is determined by subtracting the CDP for the same period but with plant equipment assumed to be operating nominally.

[b]Basic event EPS-DGN-FC-1B 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.

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DUKE POWER

DATE: April 2, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 2
Docket No. 50-370
Licensee Event Report 370/96-02, Revision 0
Problem Investigation Process No.: 2-M96-0331

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/96-02, Revision 0, concerning past inoperability of Emergency Diesel Generator 2B. This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (i) (B). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T.C. McMeekin

JWP/gts

Attachment

cc: Mr. S.D. Ebneter
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Mr. George Maxwell
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McGuire Nuclear Station

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Enclosure 3

bxc: B. L. Walsh (EC11C)
Z. L. Taylor (CNS)
G. A. Copp (EC050)
J. I. Glenn (MG02ME)
P. R. Herran (MG01VP)
C. B. Davis (MG01CP)
J. E. Burchfield (ONS Reg Compliance)
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P. M. Abraham (EC08I)
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K. L. Crane (MG01RC)
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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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)
McGuire Nuclear Station, Unit 2	05000 370	1 OF 8

TITLE (4)
Past Inoperability Of Emergency Diesel Generator 2B Due To Low Lube Oil Pressure Caused By Unanticipated Interaction Of Systems And Components.

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(S)
										05000
03	06	96	96	02	0	03	29	96	N/A	05000

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

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

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)	
J. W. Pitesa	(704)	875-4788

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
B[illegible]	EEBLOA	IDISSW	C753	NO					
B6	EEBLOA	IDISSW	U075	NO					

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

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

Unit Status: Unit 2 was in Mode 1(Power Operation) at 100 percent power.

Event Description: On February 6, 1996, during the performance of a scheduled operability test involving a non-prelubed start of Emergency Diesel Generator (EDG) 2B, the EDG tripped after approximately 39 seconds due to a low lube oil pressure indication. Investigation of this event revealed that the indication was false, and had resulted from slow instrumentation response due to excessive air entrainment coupled with low EDG room temperature. Inadequate design for the associated impulse lines led to the air entrainment. On March 6, 1996, it was determined that periods existed when EDG 2B was past inoperable. For reporting purposes it was assumed the inoperable period exceeded the Technical Specifications limiting condition. In fact, the EDG was not continuously inoperable during these periods. A similar failure had occurred on EDG 2A on February 15, 1994. The impact of air entrainment coupled with low EDG room temperature on the speed of pressure indication response was not considered as significant when formulating corrective actions associated with the 1994 event.

Event Cause: This event is assigned a cause of Design Deficiency due to unanticipated interaction of systems and components.

Corrective Action: Corrective actions include increased frequency for periodic venting and implementation of modifications to the impulse lines for these instruments on all EDGs.

EVALUATION:

Description of Event

On February 6, 1996, Unit 2 was in Mode 1(Power Operation) at 100 percent power.

- Operations (OPS) personnel were running procedure PT/2/A/4350/02B, Diesel Generator 2B Operability Test, involving a non-prelubed start of Emergency Diesel Generator [EIIS:DG] (EDG) 2B. This test is required twice a year and is performed in August and February to test temperature extremes.

- After verification of prerequisites, the EDG 2B start sequence was initiated per enclosure 13.1 of the procedure.

- The EDG started, and the time to reach 95 percent speed, as indicated on the EDG control panel [EIIS:PL], was noted to be acceptable.

- At that time the Operator running the test heard the EDG slowing down and then observed that it had tripped. He also noted the annunciator [EIIS:ANN] for "Low Lube Oil Pressure" on the EDG control panel was lit.

- The EDG Engine Lube Oil (LD) system [EIIS:LA] pressure as indicated on control panel pressure gauge [EIIS:PI] 2LDPG 5130, was at 15-20 psig and decreasing (normal operating pressure is 40 psig).

- The Operator then notified Control Room [EIIS:NA] OPS personnel, and Engineering personnel of the EDG trip.

- Initial investigations found no problems with the associated circuitry, and it was concluded that the EDG had most likely tripped due to a low LD system pressure indication at approximately 39 seconds, as designed.

- This trip is considered a "valid failure" because the "Low Lube Oil Pressure" trip is not bypassed during an emergency start. Special Report 96-001 was submitted to the NRC on March 07, 1996, as required to document this failure.

- Further investigation by OPS, Engineering, and Maintenance personnel revealed that the low LD system pressure indication was false and had resulted from slow response by the associated instrument loop.

- The cause of the slow response was further determined to be primarily due to inadequate design of the associated instrument impulse lines which creates the possibility for air to be introduced into the system. This coupled with increased viscosity of the oil caused by low EDG room temperatures caused the oil pressure instrumentation response to be slowed significantly.

- Based on past operating experience at McGuire Nuclear Station and information obtained from other utilities, it was determined that long impulse lines in conjunction with low EDG room temperatures have caused similar problems in the past. The total length of the pressure switch impulse line for EDG 2B is approximately 70 feet. The lowest recorded room temperature in the seven days preceding this event was approximately 62 degrees F, but had risen to approximately 68 degrees F at the time of the test.

- Testing was performed to determine the impact of these findings on EDG operability. Based on the results of this testing, on March 6, 1996, EDG 2B was determined to be "Past Operable" for periods with the room temperature >/= 71 degrees F, and for periods when the Before and After Lube Oil Pump [EIIS:P] was running (runs for 15 min out of each hour). However, the EDG was determined to be "Past Inoperable" for periods with the room temperature < 71 degrees F when the Before and After Lube Oil Pump was not running. Using this criteria, all other EDGs were determined to have been past operable.

- A detailed review of the EDG 2B room temperature trend showed that EDG 2B had experienced some periods in excess of 72 hours when the room temperature was < 71 degrees F. As noted in the above paragraph, EDG 2B was operable for at least 15 minutes of each hour during these low temperature periods. However, for reporting purposes, the EDG was assumed to be potentially inoperable in excess of the 72 hour Limiting Condition For Operation (LCO) for one EDG inoperable.

- Review of the operability of EDG 2A during those times revealed that the only concurrent inoperability occurred on January 4, 1996, when motor control center [EIIS:MCC] 1EMXH-1 was removed from service for maintenance. Train A of the Nuclear Service Water (RN) system

[EIIS:BI] and EDG 2A were declared technically inoperable during that maintenance period. They were functionally available during this period and would have been able to perform their design functions. Duration of the technical inoperability was 5 hours and 17 minutes which is less than the 8 hour Technical Specification LCO time for two EDGs inoperable.

- On March 6, 1996, the NRC was notified of the violation of the Technical Specification in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements.

Conclusion

This event did not result in any uncontrolled releases of radioactive material, personnel injuries, or radiation overexposures. The event is not Nuclear Plant Reliability Data System (NPRDS) reportable.

This event is assigned an NRC cause code of Design Deficiency, resulting in an unanticipated interaction of components.

- The design of the impulse lines for the LD system pressure instrumentation associated with this event creates the possibility that air will be introduced into the system.

- Initial observation of EDG 2B indicated that little air was emitted from the impulse line for the pressure instrumentation when it was vented. However, during extensive testing of EDG 2B, conducted on March 5, 1996, a significant amount of air was found to be in the impulse line, as observed from the main control panel vent and the individual instrument vents. Following this extensive venting, the EDG 2B LD system pressure instrumentation response improved dramatically. This indicates that air entrainment is the primary contributor to the trip of EDG 2B on February 6, 1996.

- Low EDG room temperatures cause low oil temperature in the impulse line, which increases oil viscosity, and slows instrumentation response to pressure transients when coupled with air entrainment. This was evidenced by slow pressure increase as indicated on the EDG 2B Control Panel Gauge at reduced room temperatures, but improved response at higher room temperatures.

- The results from testing performed show that the combination of these factors (air entrainment and low room temperatures) causes pressure response of the instrumentation to be slowed more than had previously been concluded. Although similar failures had occurred in the past, the results of this combination of factors had not been previously understood.

- A review of the Operating Experience Program (OEP) and Problem Investigation Process (PIP) data bases for the past 24 months revealed that a similar failure had occurred on EDG 2A on February 15, 1994, as documented on Special Report 94-001 and PIP 2-M94-0242. During the evaluation of that event, the impact of low EDG room temperature on the speed of pressure indication response was considered, but determined to not be a significant factor. Therefore, the corrective actions taken at the time of that event did not fully address the problem as it is now known, and were not adequate to prevent recurrence. This event is considered to be recurring.

CORRECTIVE ACTION:

Immediate:

1. Troubleshooting was performed on the associated circuitry to determine the cause of the failure.
2. The Lube Oil Full Flow Filter Bypass Valves [EIIS:V] (Units 1 and 2LD-0108A, and Units 1 and 2LD-0113B) were manually opened to ensure maximum LD system header pressure response during an EDG start.
3. The EDG room and LD system keepwarm temperatures were increased for all EDGs.
4. The Lube Oil Full Flow Filter Bypass Valves were reclosed on all EDGs.
5. A Conditional Operability statement was issued for EDG 2B to require performance of a Semi-Daily Surveillance to ensure maintenance of EDG room temperature > 75 degrees F and LD system keepwarm temperature > 115 degrees F per OPS Special Order 96-02.

Subsequent:

1. A Conditional Operability statement was issued for Unit 2 EDGs to require performance of a Semi-Daily Surveillance to ensure maintenance of EDG room temperature > 75 degrees F.

2. The Unit 1 Operator Rounds Sheet was revised to ensure maintenance of EDG room temperature between 75-85 degrees F.

3. Engineering personnel conducted further investigations and testing to determine the exact cause of the failure.

4. The periodic maintenance tasks for flush and vent of the LD system pressure loops for all EDGs was changed from an annual to a quarterly frequency and improved guidance was added within the tasks to ensure proper venting.

5. Engineering personnel initiated periodic testing of the LD system impulse lines on all EDGs for degradation.

Planned:

1. Nuclear Station Modifications will be performed to shorten the impulse lines and move the pressure instrumentation for both Unit 1 and 2 EDGs.

2. Engineering personnel will evaluate the results of the modifications on LD system performance and adjust the frequency of periodic testing for degradation and periodic maintenance tasks for flush and vent accordingly.

SAFETY ANALYSIS:

Based on this analysis, this event is not considered to be significant. At no time were the health and safety of the public or plant personnel affected as a result of this event.

- The EDGs are required to start and run to support Engineered Safety Feature (ESF) loads to mitigate an accident involving a Loss of Offsite Power (LOOP). FSAR Chapter 15 contains the analysis of several accidents assuming the LOOP event. The primary event of interest is the LOOP event as an initiating event. The existing McGuire Probabilistic Risk Assessment (PRA) assumes the frequency of LOOP events to be 0.07 events per year.

- The particular failure associated with this event is a failure of the EDG 2B to start and run on demand due to a false low lube oil pressure indication. During all time periods that EDG 2B was susceptible to failures of this nature, the 2A EDG was functionally available. Should EDG 2A fail following a LOOP and the EDG 2B fail to start and run, a variety of means are available to mitigate the LOOP. Some examples of these are as follows:

  ◊ OPS personnel could try to restart EDG 2B since the failure only involves a slow impulse line response.

  ◊ At McGuire there are two shared 4160V auxiliary transformers which can power the 4160V buses from either Unit. These transformers would be available except for a LOOP involving both Units. Based on industry data, LOOP to more than one unit occurs in only 17 percent of all LOOP events.

  ◊ Assuming both EDGs failed following the occurrence of a LOOP and failures of power from the other unit, from power run back, and from recovery of offsite power, the Unit could still be maintained in a safe shutdown condition with the use of the Standby Shutdown Facility (SSF), which can supply the Reactor Coolant [EIIS:AB] Pump Seal Injection and provide Steam Generator [EIIS:SG] cooling by means of the Turbine [EIIS:TRB] Driven Auxiliary Feedwater [EIIS:BA] Pump.

- This event was analyzed using the McGuire PRA models by setting EDG 2B start failure to "True" (failure probability of 1) and using an exposure time of 499.5 hours. The exposure time is based on a period of 666 hours that EDG 2B was susceptible to the conditions necessary for the failure to occur minus 25 percent to account for the time that the Before And After Lube Oil Pump normally runs (the start failure would not have occurred while the Before And After lube oil pump was running). Also, the above mitigation means were applied. The result of this analysis is that the increase in core damage frequency due to this event is approximately 1E-08. This event has a negligible impact on normal plant risk.

- LER 370/96-01 documents a Unit 2 Refueling Water Storage Tank Level Instrumentation Inoperability During Cold Weather Conditions. The impact of the FWST level instrument inoperability while EDG 2B was considered inoperable has been examined with respect to the LOCA-LOOP scenario risk. Using an exposure time of 4 days, a bounding

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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)
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assumption of a LOOP following a LOCA (1E-03), and the probabilistic failure mode of EDG 2A being operable, the associated accident sequence is estimated to have a probability of occurrence of less than 1.8E-08. Therefore, this scenario is not probabilistically significant.

- During the periods of reduced reliability of EDG 2B, no event requiring the use of the EDGs occurred at the McGuire site.