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July 6, 2004

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269,-270, -287
Licensee Event Report 269/2004-02, Revision 0
Problem Investigation Process No.: O-04-2808

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 269/2004-02, Revision 0, regarding a Main Steam Line Break mitigation design/analysis deficiency which could result in the main and startup feedwater control valves being technically inoperable for mitigation of some steam line break scenarios.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(i)(B) as a condition prohibited by Technical Specifications, 50.73(a)(2)(ii)(B) as an Unanalyzed Condition, and 50.73(a)(2)(V)(D) as a potential loss of safety function for Accident Mitigation. This event is considered to be of no significance with respect to the health and safety of the public.

Portions of this report are incomplete. The root cause investigation and an analysis of the consequences of potentially exceeding the Environment Qualification (EQ) envelope curve are still in progress. At this time we anticipate completing these tasks and providing a supplemented report by approximately August 31, 2004.

IE22

Very truly yours,

Bruce Hamilton/for

R. A. Jones

Attachment: Licensee Event Report 269/2004-02, Revision 0

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bxc: ONS Site:

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(Revised 4-5-2004)

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Oconee Nuclear Station, Unit 1	050- 0269	1 OF 7

4. TITLE

Main Steam Line Break Mitigation Design/Analysis Deficiency

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME: Unit 2	DOCKET NUMBER: 270
05	04	2004	2004	– 02	– 0	07	06	2004	FACILITY NAME: Unit 3	DOCKET NUMBER: 287

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)	X	50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
B.G. Davenport, Regulatory Compliance Manager	(864) 885-3044

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		08	31	2004

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

The Automatic Feedwater Isolation System (AFIS) Circuitry actuates various components including the main and startup feedwater control valves (FCVs) in order to mitigate a Main Steam Line Break (MSLB) with or without a Loss of Offsite Power (LOOP). Tech Spec 3.7.3 requires the FCVs to be operable. The FCVs fail-as-is and require Instrument Air (IA) to close.

On April 29, 2004, Units 1 and 3 were operating in Mode 1 at 100% power; Unit 2 was in No Mode during a refueling outage. During discussion between Site Engineering (SE) and General Office-based Safety Analysis (SA) personnel, it was recognized that, for smaller breaks, actuation signals/alarms may be delayed such that the IA header may depressurize before AFIS and/or operator actions initiate FCV closure. For breaks inside containment, this could lead to pressurization of the Reactor Building (RB) above the RB design pressure (but below RB failure pressure, 144 psig). Immediate action was taken to maintain a diesel air compressor operating at all times pending a more permanent resolution. On May 4, 2004 the event was determined to be reportable. The FCVs are considered to have been inoperable longer than allowed by TS. The root cause investigation is still in progress, this report will be supplemented after it is complete. This event is considered to have no significance with respect to the health and safety of the public.

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

EVALUATION:

BACKGROUND

Oconee Nuclear Station (ONS) Technical Specifications (TS) 3.7.3 requires the Main and Startup Feedwater Control Valves (FCVs) to be operable to close to isolate Main Feedwater (MFW) during a Main Steam Line Break (MSLB) event. This report involves the recognition that some MSLB event scenarios require Instrument Air (IA) to be available to close the FCVs at a time in the scenario after IA is no longer available. As a result the FCVs are considered to have been inoperable longer than allowed by TS. This event is reportable per 10CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS, 10CFR 50.73(a)(2)(ii)(B) as an Unanalyzed Condition and 50.73(a)(2)(V)(D) as a potential loss of safety function for Accident Mitigation. An ENS notification was made May 4, 2004 (NRC Event # 40724) which reported this event under 10CFR 50.72(b)(3)(ii)(B) Unanalyzed Condition and 50.72(b)(3)(v)(D) Accident Mitigation.

In 1993, Safety Analysis (an engineering group located in the Duke Power general office) performed a reanalysis of the MSLB scenario. Safety Analysis determined that previous calculations, based on a vendor methodology, were non-conservative. Using improved methodology, calculations indicated the containment pressure design limit could be exceeded without prompt operator action to isolate MFW. This was reported to the NRC, reference LER 269/93-06 dated July 1, 1993. Long term corrective actions resulted in a series of modifications to install automatic control circuitry now known as AFIS. AFIS circuitry is safety-grade, but the FCVs, which are actuated by the circuitry, remained non-safety-grade.

A MSLB is defined in UFSAR Section 15.13 as a double-ended guillotine rupture of 34 inch diameter piping in the Main Steam System. Other sections of the UFSAR, e.g. Section 15.17, address smaller breaks. In the event of a MSLB, the AFIS modification was designed to automatically isolate MFW, prevent operation of the turbine-driven emergency feedwater (EFW) (TDEFW) pump, and inhibit motor-driven EFW flow to the faulted steam generator. These functions are credited in both the MSLB containment pressurization analysis of UFSAR Section 6.2.1.4 and the MSLB tube stress analysis of UFSAR Section 5.2.3.4. For the Section 6.2.1.4 analyses, "It is assumed that failure of a feedwater control valve to close on a feedwater isolation signal is beyond the licensing basis." However, Section 15.13 specifically does not credit closure of the FCVs (because the NRC acknowledged that they were not safety grade and subject to single failure). Section 15.13 concludes that dose consequences of a break inside containment are bounded by those of a break outside containment.

The IA system at ONS is non-safety. A loss of offsite power (LOOP) causes the loss of electrical power to the IA compressors, after which the available air is limited to the volume in air receiver tanks and the system piping. The IA system provides the motive force to operate the FCVs. These valves are designed to fail "as-is" to minimize a transient following a loss of IA during plant operation at power. For a MSLB/LOOP the valves must close. Therefore the FCVs must be closed before the IA system inventory becomes inadequate to operate them.

At the time of discovery of this event Units 1 and 3 were operating in Mode 1 at 100% power with no safety systems or components out of service that would have contributed to this event. Unit 2 was at No Mode during a refueling outage. However, this event is a historical issue and all three units have operated in this condition.

EVENT DESCRIPTION

In March 2000, a Problem Investigation Process (PIP) report was initiated due to unresolved items identified during a comprehensive review of event mitigation calculations. These items appeared to be assumptions which did not have supporting calculations. The PIP was to provide documentation of the issues and to track completion of the necessary supporting calculations. One corrective action was to validate the statements that the FCVs could actually close during the MSLB event if there is a coincident LOOP and/or loss of IA. This led to the creation of calculation OSC-8222 to quantify the amount of time that sufficient IA pressure would be available following a LOOP.

Calculation OSC-8222 was approved 1/30/2003 and showed that the FCVs would not be able to close after 2.1 minutes following a LOOP. With allowance for valve stroke time, this limit required that the signal to close the FCVs in a MSLB/LOOP event must be generated

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within approximately 1.6 minutes of the break. During the 1993 event, Operations and Training personnel had performed a number of validations on the Oconee simulator and had verified that, following the worst case (large) break, the Operators could close the FCVs within times which met this limit. The MSLB/AFIS modifications were installed to automate this action for large breaks. Because a double-ended guillotine MSLB would generate an automatic AFIS actuation within a few seconds of a break, site engineering concluded that the OSC-8222 calculation result was acceptable. No consideration was given to smaller MSLBs, which the ONS licensing basis states are mitigated by manual operator action within ten minutes. Also, the results of this calculation were not communicated to Safety Analysis.

In January 2004 additional PIP corrective actions were initiated to revise the IA and FDW Design Basis Documents (DBDs) to include documentation of the requirements for the FCVs to close in the MSLB event and the requirement for IA to support those closures. When preparing 50.59 documentation for these revisions, site engineering personnel recognized an apparent discrepancy between the OSC-8222 results and the licensing basis documents related to AFIS. A meeting was held between Safety Analysis personnel from the Duke general office and site engineering. As a result of that meeting, site personnel learned that, in order to limit smaller breaks scenarios, operator actions were credited later in the event than they had previously understood. Safety Analysis personnel learned that earlier statements as to adequacy of IA had been based on the large break scenario expectation that operator actions were performed early in the event before IA reservoirs were depleted.

The small MSLB with LOOP design deficiency was identified on April 29, 2004, and a PIP was initiated to address the problem.

Operations shift personnel were notified and took action to assure continued operability by starting a back-up diesel air compressor. This would maintain an air source for the FDW control valves in the event of a LOOP.

Operations initially considered that FCV closure was not credited in UFSAR 15.13 and concluded that the event did not meet reportability requirements per 10CFR 50.72.

On May 4, 2004 the operating backup diesel compressor experienced an oil leak and no diesel was in operation for a period of time while a second backup diesel was placed in service. During review of this additional event, ONS concluded that the issue was reportable and an ENS notification was made at 1908 hours on May 4, 2004 (NRC Event # 40724).

Subsequently, additional diesel air compressors were connected to the IA header as spares to improve reliability. Operations procedures were revised to require one diesel air compressor in operation at all times pending a more permanent resolution to this issue.

A root cause team was formed in June 2004 to establish the root cause for the event.

CAUSAL FACTORS

The root cause investigation results have not yet been finalized. This report will be supplemented within thirty days of the date the root cause report is approved.

CORRECTIVE ACTIONS

Immediate:

1. Operations took action to assure continued operability by starting a back-up diesel air compressor.

Subsequent:

1. Additional diesel air compressors were connected to the instrument air header as spares to improve reliability.

2. Operations procedures were revised to require one diesel air compressor in operation at all times pending a more permanent resolution to this issue.

Planned:

1. An Engineering project team is currently evaluating proposals for both interim and permanent resolutions of this issue. Appropriate corrective actions will be identified and implemented.

2. Additional corrective actions are being developed in association with the root cause investigation to address the root causes of this event. This report will be supplemented following completion of the root cause investigation and appropriate corrective actions will be included.

None of the corrective actions identified to date are considered an NRC Commitment items. There are no NRC Commitment items contained in this LER.

SAFETY ANALYSIS

There were no actual safety system functional failures associated with this event. However, this event scenario represents a potential failure on each of the three ONS units; therefore this event will count as three (3) safety system functional failures for the NRC/INPO Performance Indicator (PI) program.

MFW isolation is credited for some aspects of a MSLB inside containment event but not for other aspects. Specifically it is credited for control of steam generator tube stresses but is not credited for offsite dose, since the limiting scenario for offsite dose is a break outside containment.

Safety Analysis performed an analysis of the small MSLB with LOOP. The peak pressure for the largest break that does not actuate AFIS within 2 minutes is 106.2 psig (0.6 ft2 break). The containment pressure and temperature exceeds the Environment Qualification (EQ) envelope curve. The consequences of this condition are still under evaluation. This report will be supplemented after this evaluation of EQ consequences is complete.

The risk impact of the AFIS design deficiency is very low. The deficiency is judged to have no material impact on the core damage frequency. The frequency of a main steam line break leading to core damage is reported in the Oconee PRA Revision 2 at less than

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1E-08. Even if this entire CDF were conservatively considered to lead to a large early release, the resulting impact would fall well below the risk significant LERF threshold 1E-07.

When additional factors are considered such as the specific break size and location, the actual impact is expected to be considerably less. In particular, the Oconee containment has been shown to be very robust under overpressure conditions (Reference: Oconee IPE Submittal, Volume III, Appendix G, "Containment Capacity Assessment"). Up to a pressure of approximately 107 psig, the estimated probability of containment failure is less than 1 percent. The mean containment failure pressure is estimated to be 144 psig. Any contribution to LERF would be expected to be at least 2 orders of magnitude below the CDF contribution.

Therefore, there was no actual impact on the health and safety of the public due to this event.

ADDITIONAL INFORMATION

The evaluation for recurring problems depends on the root cause classification. This section will be revised when the report is supplemented.

There were no releases of radioactive materials, radiation exposures or personnel injuries associated with this event.

This event is not considered reportable under the Equipment Performance and Information Exchange (EPIX) program.