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DUKE POWER

March 12, 1996

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 269/96-03

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report, 269/96-03, concerning the technical inoperability of the Reactor Coolant Makeup System for an Appendix R scenario.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(v)(B). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J W Hampton

J. W. Hampton, Vice President
Oconee Nuclear Site

/fts

Attachment

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Document Control Desk
March 12, 1996

xc: Mr. L. A. Wiens, Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. S. D. Ebneter, Regional Administrator
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

Mr. P. E. Harmon
NRC Resident Inspector
Oconee Nuclear Station

INPO Records Center
700 Galleria Parkway
Atlanta, GA 30339-5957

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit One	05000 269	1 OF 7

TITLE (4) Reactor Coolant Makeup System Technically Inoperable For Appendix R Scenario Due To Design Analysis

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
										05000
02	14	96	96	-- 03 --	00	03	12	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
		20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	100	20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	X 50.73(a)(2)(v) (B)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
L. V. Wilkie, Safety Review Manager	(864) 885-3518

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	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

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

On February 5, 1996, Oconee Unit 1 was at 100% full power. Engineering personnel were reviewing 10 CFR 50 Appendix R correspondence files and noticed that an evaluation completed in 1987 had assumptions on Reactor Coolant (RC) pump seal leakage that did not agree with the current assumptions. A Problem Investigation Process Report was initiated to evaluate the condition. On February 14, 1996, engineering concluded that, when the current RC pump seal leakage limits are applied to an Appendix R scenario, the RC System leakage could have exceeded the Reactor Coolant Makeup system design limits. The current maximum RC pump seal leakage limits have been in effect since 1993. The root cause of this event is Design Analysis; System functional design deficiency (application). Corrective actions included taking compensatory actions to limit leakage. Also, the Design Basis Document and operating procedures will be revised.

BACKGROUND

The Reactor Coolant Makeup (RCMU) system is provided to supply makeup to the Reactor Coolant (RC) [EIIS:AB] system in the event normal systems are inoperable due to any Standby Shutdown Facility [EIIS:NB] event (fire, flood, sabotage, or station blackout). The RCMU pump is capable of delivering 29 gpm to the RC system by taking suction from the Spent Fuel Pool and discharging to the RC pump seals.

The 10CFR50 Appendix R design requirements for a fire states that one train of equipment necessary to achieve and maintain Hot Shutdown shall remain free of damage by a single fire.

Oconee Nuclear Station has three Pressurized Water Reactor units with each unit having four RC pumps. Unit 1 has Westinghouse RC pumps and units 2 and 3 have Bingham RC pumps with different seal packages than the Westinghouse RC pumps.

Another difference between the units is that the unit 1 High Pressure Injection [EIIS:BG] system contains a normally closed, electric motor operated valve (1HP-276) that controls fill water flow to the RC pump standpipe. The valve is in a branch line off the RC pump number 1 seal leakoff main line. The valve is normally not opened with the RC system temperature above 250 F.

EVENT DESCRIPTION

The Reactor Coolant Makeup (RCMU) system was originally designed in 1980-1981 as part of the Standby Shutdown Facility (SSF). The SSF was placed in operation in 1984-1985.

An NRC Appendix R fire inspection was performed at Oconee from January 26-30, 1987. The inspection revealed two valves which could spuriously open during an Appendix R fire. A Problem Investigation Report was initiated and LER 269/87-02 (Appendix R

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

Review With Respect To Valve Operability) was submitted to the NRC. During the course of the investigation by Duke Power Company, it was noted that other valves could spuriously open during an Appendix R fire including valve 1HP-276. The detailed calculation of the maximum Reactor Coolant pump seal leakage indicated that there was enough capacity available from the RCMU pump to accommodate these spuriously opened valves.

On May 25, 1992, Unit 1 experienced a Technical Specification required shutdown to correct excessive RC pump seal leakage caused by the installation and premature degradation of obsolete seal parts (LER 269/92-09). As a result, an evaluation of the parameters that affect the RCMU system operability was performed. Specifically, the evaluation was to determine the adequacy of the RCMU system to supply the RC pump seals during a SSF event. Engineering calculated RC pump seal leakage rates which were higher than previously noted. Engineering was working on the Design Basis Document (DBD) for the SSF RCMU system at that time.

On July 1, 1993, engineering determined that the unit 1 RC pump seal leakage rates had occasionally exceeded the newly established maximum allowed seal leakage rates (LER 269/93-07).

On October 31, 1994, the DBD for the RCMU system was issued. The latest revision was issued September 6, 1995.

On February 5, 1996, an engineer was reviewing Appendix R correspondence files including the evaluation of the problem identified in 1987. The engineer questioned the assumptions on RC system leakage as compared to the current requirements. Due to the possibility of a problem, operations was notified, valve 1HP-276 was verified closed and its breaker opened.

On February 14, 1996, engineering completed an evaluation using the current RC pump seal leakage data. They concluded that if an Appendix R fire caused valve 1HP-276 to spuriously open, the back pressure downstream of the RC pump number 1 seals could decrease below the vapor pressure of the liquid passing through the seal. This would result in two phase flow across the RC pump number 1 seals. The manufacturer is unable to predict whether or not two phase flow across the RC pump number 1 seals will result in seal degradation or failure. If seal degradation or failure occurred, the capacity of the systems used to makeup for RC system inventory lost during this Appendix R scenario could be less than the leakage rate from the RC system. RC system inventory could be reduced and eventually natural circulation flow could be interrupted. Therefore, the spurious opening of valve 1HP-276 during an Appendix R event resulted in the technical inoperability of the RCMU system. Engineering did not notify Oconee Safety Assurance personnel immediately due to a misunderstanding of the reportability requirements.

On February 19, 1996, Oconee Safety Assurance was notified. The NRC was notified at 1430 hours of the past inoperability. A Problem Investigation Process Report was initiated because the notification to the NRC was not made within four hours as required by 10 CFR 50.72. Enhancements to the notification process will be made as a result of this incident.

CONCLUSIONS

The root cause of this event is a Design Analysis; system functional design deficiency (application), which occurred during the original design of the Reactor Coolant Makeup (RCMU) system in 1980-1981.

The original design calculations had not properly taken into account the effect that seal injection water temperature versus back pressure downstream of the RC pump number 1 seal could have on RC pump seal leakage. This condition was not identified as part of the postulated Appendix R event reported in 1987. Spurious 1HP-276 valve openings during an Appendix R event could have been considered when the new RC pump seal leakages were calculated in 1993. The engineers who were familiar with the newly calculated RC pump leakage rates were not originally involved with the problem in 1987. They were not aware of the assumptions made until the old correspondence was found.

The Design Basis Document (DBD) for the Standby Shutdown Facility (SSF) RCMU system was completed in 1994. The DBD has identified various deficiencies in the original design assumptions of the SSF RCMU system. It also addresses the Appendix R design criteria. However, the identification of the particular problems associated with the spurious opening of valves, as related to the RC pump seal leakage, is not addressed. The scope of the DBD did not require inclusion of equipment outside of the SSF RCMU system. Spurious valve actuation was a separate issue addressed in the Appendix R regulation.

This event is considered not recurring. The problem identified in this report was recognized during a review of correspondence associated with previous LER 269/87-02. There have not been any 10 CFR 50 Appendix R events associated with the SSF RCMU system during the past two years. There have been events with a root cause incorporating deficient Design Analysis. These Design Analysis deficiencies relate to the original designs of systems or equipment and were not associated with the SSF RCMU system. There have been Design Analysis deficiencies associated with the SSF RCMU system as referred to in this report. However, they were originally identified in 1992 and 1993. Also, the 10 CFR 50 Appendix R event, referred to in LER 269/87-02, was an original design deficiency that identified a similar scenario. Other original design deficiencies identified in 1992 and 1993 were opportunities to discover this problem but would have only shortened the duration, not prevented this event.

There were no personnel injuries, releases of radioactive materials, or NPRDS reportable equipment failures associated with this event.

CORRECTIVE ACTIONS

Immediate

1. The Operations Shift Manager was notified, valve 1HP-276 was verified closed, and the breaker for the valve was opened.

Subsequent

None

Planned

1. Revise affected operating procedures to comply with the Appendix R requirements as required by the engineering evaluation.

2. Revise the High Pressure Injection Design Basis Document to describe valve and breaker position requirements needed to insure Appendix R compliance.

SAFETY ANALYSIS

The High Pressure Injection (HPI) and Component Cooling (CC) [EIIS:CC] systems provide cooling to the Reactor Coolant (RC) pump seals during normal plant operation. If these systems are unable to provide seal cooling, the Reactor Coolant Makeup (RCMU) system can be used to provide RC pump seal cooling, in addition to replenishing the RC system to offset seal leakage and RC system shrinkage during cooldown to hot shutdown.

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

The Standby Shutdown Facility (SSF) Reactor Coolant Makeup (RCMU) system would perform as designed/expected. The Appendix R mitigation strategy is supposed to "bottle-up" the miscellaneous Reactor Coolant (RC) System leakage paths to minimize RC system inventory loss. This allows the capability to achieve and maintain RC system natural circulation flow. However, if valve 1HP-276 spuriously actuates, RC system expected leakage could potentially be greater than the flow rate that the SSF RCMU system is capable of delivering. This could have resulted in the inability to maintain RC system natural circulation flow.

The Oconee Final Safety Analysis Report (FSAR) analyzes Loss Of Coolant Accident (LOCA) events for a spectrum of break sizes that envelope RC pump seal LOCA's. The FSAR analyses demonstrate that the core will remain covered and radiological releases will remain within 10 CFR 100 limits for seal LOCA's with HPI safety injection. RC pump seal LOCA events without HPI safety injection are not analyzed in the FSAR because no plausible single failure would fail the HPI and CC systems. However, this type of design event has been analyzed in support of safety evaluations for a station blackout (SBO). For a SBO with a postulated seal leakage of greater than 25 gpm per RC pump, eventually the core would be uncovered unless HPI could be placed back in service. Returning HPI to service would keep the core covered. With the core covered, the radiological consequences of a RC pump seal LOCA are expected to be bounded by the FSAR Chapter 15 LOCA analyses.

A probabilistic risk assessment analysis has been performed on this event. This analysis indicates that a possible seal LOCA resulting from a spurious opening of valve 1HP-276 during an Appendix R fire is not risk significant.

The health and safety of the public were not compromised by this event. Also, this event did not result in the release of any radioactive materials, radiation exposures or personnel injuries.