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

APPROVED OMB NO. 3150-0104
EXPIRES: 04/30/98
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)
Oconee Nuclear Station, Unit 1	05000 269	1 of 9

TITLE (4)
Inadequate Analysis of ECCS Sump Inventory due to Inadequate Design Analysis

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER(S)
									Unit 2	05000 270
12	10	97	97	- 010 -	00	1	8	98	Unit 3	05000 278

OPERATING MODE (9): N
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)	X OTHER (*Specify in Abstract below and in Text, NRC Form 366A*)
20.405(a)(1)(iii)	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.E. Burchfield, Regulatory Compliance Manager	(864)	885-3292

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*)	X NO				

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

On November 20, 1997, a Self-Initiated Technical Audit documented potentially non-conservative assumptions used in the calculation for the Reactor Building (RB) water level following a large break LOCA. The most significant issue was the modeling of water trapped in areas inside of the RB. After preliminary evaluation, a Systems Engineer concluded, on December 10, 1997, that inadequate NPSH might exist during long term Emergency Core Cooling System (ECCS) operation in the recirculation mode. At 1411 hours, Oconee made a 1 hour notification to the NRC per 10 CFR 50.72(b)(1)(ii)(B), to report that Unit 3 was potentially outside its design basis. At this time, Unit 1 was at cold shutdown and Unit 2 was at 100% but was considered less susceptible to the event. A flow restricting flange and 2 basket strainers were removed from the Reactor Vessel Cavity and Fuel Transfer Canal drain lines on each unit to provide additional NPSH margin to assure current operability. Subsequently, the completed evaluation concluded that the system had been operable on all Oconee units. Therefore, this is a Voluntary Report. The root cause of this event is Inadequate Design Analysis. A contributing deficiency was Inadequate Change Management.

EVALUATION:

Background

The Emergency Core Cooling Systems (ECCS) [EIIS:BP] at Oconee include the High Pressure Injection [EIIS:BG] , Low Pressure Injection (LPI) [EIIS:BP], and Core Flood [EIIS:BP] Systems. In LOCA scenarios, short term core cooling is accomplished by injection of water from the Core Flood tanks and/or the Borated Water Storage Tank (BWST) [EIIS:TK]. As water is lost from the Reactor Coolant System (RCS)[EIIS:AB] due to the LOCA, it is collected inside containment where it fills the Reactor Building (RB)[EIIS:NH] basement. The basement floor is configured such that this inventory of water will flow into the Reactor Building Emergency Sump (RBES). When the BWST is depleted, long term core cooling is accomplished by establishing recirculation flow from the RBES using LPI system piping. Also, the RB Spray (BS) [EIIS:BE] System provides long term containment cooling by establishing recirculation flow from the RBES using suction piping shared with the LPI system.

The RB interior structures include the reactor cavity, and a Fuel Transfer Canal (FTC) which is located between the steam generator compartments and above the reactor cavity. See Attachment A. The reactor cavity houses the reactor vessel and serves as a biological shield. The reactor cavity provides accessibility for maintenance and inspection of incore instrumentation, piping, and nozzles. It also provides a sump and drainage line for leak detection.

The FTC has two sections, the shallow end immediately over the reactor vessel and a deep end. In LOCA scenarios, some of the BS inventory and some condensate from the leaking RCS inventory could collect in the FTC. There are two 4-inch drain lines off of the deep end of the FTC which are used during refueling operations to drain the canal. There is a branch line from this drain to the RB Normal Sump (RBNS) which contains two manual isolation valves in the Spent Fuel Cooling (SF) [EIIS:DA] System. These valves must be kept open during normal plant operation so that the canal will gravity drain to the normal sump. In the event of a LOCA, the RBNS will overflow to the RBES as part of the supply of water for post-accident sump recirculation for ECCS systems.

Description of Event

As one corrective action related to events with the High Pressure Injections System (HPI) in May and June of 1997, Duke Management requested that a Self-Initiated Technical Audit (SITA) be performed to review the HPI and Low Pressure Injection (LPI) systems. Within Duke's Nuclear Generation Department, a SITA is an independent, in-depth evaluation of the technical adequacy of a system. This SITA conducted its evaluation from mid-November to late December, 1997. During this period, Oconee Unit 1 was shutdown in a refueling outage and Units 2 and 3 were operating at 100% full power.

On November 20, 1997, the SITA team initiated a Problem Investigation Process (PIP) entry to document assumptions which the SITA questioned as potentially non-conservative. These assumptions were used in Calculation OSC-1925, the approved calculation for the Reactor Building (RB) water level following a large break LOCA. The most significant issue was the modeling of water trapped in areas inside of the RB. There were two significant areas identified where water could be trapped such that it could not flow to the Reactor Building Emergency Sump (RBES).

One was the Reactor Vessel Cavity area. The Cavity is the area formed by the primary shield immediately surrounding the reactor vessel. It is accessible from the shallow end of the Fuel Transfer Canal (FTC) and from a crawl space below the Vessel. However, the crawl space is sealed during plant operation by metal doors. The Cavity is penetrated by piping going to the vessel (hot legs, cold legs, core flood/decay heat removal lines), a line from the FTC deep end, and incore nuclear instrumentation guide tubes (which pass through a sealed, tight fitting bulkhead). The bottom of the vessel cavity area includes a 4 inch line which drains to the RB Normal Sump (RBNS). However, the RBNS end of the drain line was covered by a flange with only a 3/4 inch pipe nipple and the SITA questioned the effect on the recirculation path to the RBES. See Attachment A.

The other area for water to be trapped was the deep end of the FTC. The drain lines contained "basket strainers" rather than perforated drain covers. Basket strainers on the drain line from the canal were assumed to have potential for blockage by debris. If blocked, they would prevent some of the water which reaches the deep end of the FTC (e.g. from Reactor Building Spray) from flowing to the RBES. A second line, located 1 foot above the floor of the deep end, would permit draining to the Reactor Vessel Cavity, where it would be subject to the 3/4 inch restrictions discussed above.

An Operability Evaluation was completed on November 23, 1997 by Systems Engineering and concluded that the net effect of the issues raised by the SITA was to lower the available water level from 5.3 feet to 3.07 feet, impacting the available NPSH for Emergency Core Cooling System (ECCS) pumps. The Operability Evaluation further concluded that this NPSH was acceptable and the RBES and associated ECCS were operable.

Subsequently, the Systems Engineer realized that the Operability Evaluation performed on November 23, 1997 had not considered that the reduced water level would increase the flow velocities toward the RBES. This velocity increase could affect the transportability of debris to the RBES. If sufficient debris reached the RBES and partially obstructed the flow path, the available NPSH potentially could be further reduced, potentially making the RBES (and, therefore the ECCS) inoperable. The Systems Engineer initiated another PIP and operability on December 8, 1997. On December 10, 1997, the assessment of operability progressed to the point where Management made a decision to make a 1 hour notification to the NRC per 10CFR 50.72(b)(1) (ii)(B), to report that Unit 3 was potentially outside its design basis. This notification was made at 1411 hours.

At that time, the ECCS on Unit 3 was considered potentially inoperable because this condition could possibly prevent proper operation in the sump recirculation mode following a LOCA. System Engineering recommended immediate compensatory actions which would provide additional margin for the available NPSH and would assure current operability of the ECCS on Unit 3. On December 10, 1997, power was reduced from 100% to approximately 30% to reduce dose while personnel entered Unit 3 containment to remove the flange from the end of the reactor cavity drain pipe at the RBNS and the strainers from the FTC drains. However, these personnel found that no flange was installed on the cavity drain pipe. A PIP was written to address this configuration control discrepancy. Some fibrous insulation, which was qualified for use inside containment based on lower flow velocities, was also removed at this time from a Feedwater line above the RBES. These actions were completed approximately 0100 hours on December 11, 1997 and power was increased back to 100%.

Unit 2 was also considered potentially affected but to a lesser extent due to the lesser amounts of fibrous insulation present. The preliminary engineering evaluation was that Unit 2 still had adequate NPSH, but Management conservatively decided to implement similar corrective actions on Unit 2. The flange on the end of the cavity drain pipe at the RBNS and the strainers on the FTC drains were removed at power.

Unit 1 was considered potentially affected, but its ECCS was not required to be operable because the unit was still at cold shutdown. The flange on the end of the vessel cavity drain pipe at the Normal Sump and the strainers on the FTC drains were subsequently removed prior to the end of the outage. Portions of qualified fibrous insulation located inside the secondary shield walls at the basement elevation were also removed.

As part of the operability evaluation initiated on December 8, System Engineering had initiated a search of documentation and calculations to determine the basis for the flange on the reactor vessel cavity drain. According to UFSAR Section 3.8.3.1, "Description of the Internal Structures," the reactor cavity was designed to structurally contain core flooding water up to the level of the reactor nozzle. Conversations with FTI confirmed that this was an original design feature that later was determined to be unnecessary. No credit is taken for this water external to the vessel in any accident analyses. While this water may be beneficial for some scenarios, it is beyond the design basis requirements for the station. Therefore, the removal of the flange is within the licensing basis for Oconee.

The strainers in the deep end of the FTC have been used for ten or more years for ALARA purposes. Investigation associated with this event indicated that the strainers were installed during an outage and were allowed to remain during operation without being evaluated and approved through the station modification process. They were removed.

The System Engineering ongoing evaluation concluded, on January 8, 1997, that the increase in transport velocity did not make the RBES inoperable. Therefore, the ECCS was operable and Oconee was within its design basis in the past.

Conclusion

The past operability has concluded that the Emergency Core Cooling Systems (ECCS) were NOT rendered inoperable by the conditions described in this report.

The root cause of this event is considered to be Design Deficiency, Inadequate Design Analysis because calculations did not adequately address areas within containment where water could be retained.

Two instances of inadequate configuration control contributed to this event. First, the Fuel Transfer Canal floor drains were apparently modified by replacing perforated drain covers with basket strainers without following the station modification process. Second, the Unit 3 Vessel cavity drain flange was not in place as shown on design drawings. The first deficiency apparently occurred in the 1980's and it is not known when the second deficiency occurred. These appear to be improperly approved or unapproved modifications. The cause of these deficiencies is classified as Inadequate Change Management (Risks and consequences not adequately reviewed or assessed).

A review of the PIP database found two reportable problems involving inadequate design analyses discovered within the previous two years. One was LER 269/97-02, Revision 1, related to water hammer in Low Pressure Service Water lines associated with the Reactor Building Cooling System. The other was LER 269/95-07, Revision 1, concerning the Low Pressure Injection System being technically inoperable due to flow induced vibration in the Low Pressure Service Water lines serving the decay heat coolers. Although the type of analyses involved were different, these events resulted in technically inoperable Engineered

Safeguards systems used for LOCA mitigation. Therefore, this event is considered recurring.

There were no personnel injuries, overexposures, or radioactive releases involved with this event.

CORRECTIVE ACTION:

Immediate:

1. Removed strainers and affected previously qualified insulation from Unit 3, and determined that the flange was not installed.

Subsequent:

1. Removed flange and strainers from Unit 2.

2. Removed flange, strainers, and affected previously qualified insulation from Unit 1 prior to start-up.

Planned:

1. System Engineering will complete formal revisions to the calculations affected by this issue.

2. System Engineering will evaluate options to increase the NPSH margin for operation in the ECCS recirculation mode.

3. The Oconee Site will complete its review and response to findings/issues from the Self Initiated Technical Audit.

Planned corrective action 1 is considered to be an NRC Commitment Item It is the only NRC Commitment Item contained in this LER.

SAFETY ANALYSIS:

Since the operability evaluation did find that adequate NPSH would exist, there would be NO expected failures and therefore no adverse impact on the ability of the Emergency Core Cooling System to fully perform its function. Therefore, this "event" had no impact on the health and safety of the public.

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

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 1	269	97	010	00	

ATTACHMENT A