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FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269

AUTH.NAME	AUTHOR AFFILIATION
WILKIE,L.V.	Duke Power Co.
HAMPTON,J.W.	Duke Power Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 93-006-01:on 930602,design deficiency resulted in condition outside design basis of containment for MS line break.Reanalyzed accident scenarios in FSAR.W/950829 ltr.

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Duke Power Company
Oconee Nuclear Site
P.O. Box 1439
Seneca, SC 29679

J. W. HAMPTON
Vice President
(803)885-3499 Office
(803)885-3564 Fax

DUKE POWER

August 29, 1995

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 269/93-06, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Revision 1 to Licensee Event Report (LER) 269/93-06, concerning a condition outside the design basis for a main steam line break accident. This revision includes information obtained from supplements to IE Bulletin 80-04.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(ii)(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
Vice President

/fts

Attachment

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, Georgia 30323

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

Mr. L. A. Wiens
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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

NRC FORM 366 (5-92) NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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

LICENSEE EVENT REPORT (LER)

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

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

TITLE (4) Design Deficiency Results In A Condition Outside The Design Basis Of Containment For A Main Steam Line Break

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Oconee, Unit 2	05000270
06	02	93	93	-- 06 --	01	08	29	95	Oconee, Unit 3	05000287

OPERATING MODE (9): N
POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iv)	X	50.73(a)(2)(ii) (B)		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 (Include Area Code)
Lanny V. Wilkie, Safety Review Manager	(803) 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

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

On June 2, 1993, at 1630 hours, Oconee Unit 1 and 3 were at 100% full power and Unit 2 was shutdown for refueling. A reanalysis, for extended fuel cycle designs and reload design optimization, resulted in the discovery of a condition outside the design basis of the plant. The main steam line break analysis in the Oconee Final Safety Analysis Report (FSAR), based on the original analysis by Babcock and Wilcox, had not taken into account the passive structural metal of the Reactor Coolant System acting as a heat source or the worst case scenarios regarding feedwater isolation. The reanalysis indicates that these are a significant heat source and shows potential for containment design pressure to be exceeded. Evaluations have concluded that existing procedures and training for operator action are adequate to limit containment pressure to below the design pressure for the most probable scenarios. The root cause is Design Deficiency; Functional design deficiency; Mechanical. Corrective actions are to continue the reanalyses of the accident scenarios in the FSAR, supplement the response to IE Bulletin 80-04, and modify the feedwater system.

BACKGROUND

The Main Feedwater (MFW) system [EIIS:BJ] takes a suction from the Condensate system [EIIS:SD], delivers the MFW to the Steam Generators (SGs) via the two steam driven MFW pumps. Each SG has a control valve and a block valve. MFW flow is controlled by the Integrated Control System (ICS). ICS controls the amount of flow by throttling the main or startup control valves.

The Final Safety Analysis Report (FSAR) Section 15.13 describes a main steam line break (MSLB) accident. The key acceptance criterion of the original analysis is based on the calculated offsite doses remaining within 10 CFR 100 limits. The original MSLB dose analysis does not differentiate between a rupture occurring inside containment and one occurring outside containment, thus no credit is taken for containment integrity. The design basis for the containment building is that it be capable of withstanding the internal pressure resulting from a Loss Of Coolant Accident, as defined in FSAR Section 15 with no loss of integrity. The FSAR acceptance criteria for a MSLB does not state containment design pressure as an acceptance criterion.

EVENT DESCRIPTION

In February 1980, IE Bulletin 80-04 required the review of the steam line break accident to determine if the potential for containment overpressure existed as a result of run out flow from the Emergency Feedwater system or due to other energy sources, such as continuation of Main Feedwater (MFW) or Condensate flow. Further, if the potential for containment overpressure existed, proposed corrective actions and schedules were to be provided. Duke Power Company responded to the Bulletin by letters dated May 7, 1980 and July 23, 1980. These responses were based on the existing containment pressure response analysis. On October 14, 1982 the NRC provided their safety and technical evaluation reports which concluded that the Oconee analysis was acceptable and no further action was required.

Duke Power Nuclear Services began reanalyzing the Final Safety Analysis Report (FSAR) Chapter 15 main steam line break (MSLB) accident in order to determine the limits that this accident might impose on plans for extended fuel cycle designs and reload design optimization. The impact of this reanalysis on the specifications for MFW block valves, which were being evaluated for replacement, was also being investigated. The focus of these reanalyses was the core response following a MSLB. The results of the core response analyses were acceptable.

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	05000 269	93	- 06 -	01	3 OF 7

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

Upon the completion of the core response, analyses were initiated in January 1993 to evaluate the containment response to a steam line break. Preliminary results of these reanalyses identified unexpectedly high containment pressure results relative to the existing FSAR containment pressure results. These results prompted evaluations into the causes of the differences. As part of the evaluation, modeling details of the original FSAR analyses were discussed with Babcock and Wilcox Company. The first cause was that the passive structural metal of the Reactor Coolant System (RCS) [EIIS:AB] was not modeled as a heat source in the FSAR analysis. The reanalyses indicated that it was a significant heat source. The second cause was that the FSAR calculation of containment response did not include worst case scenarios regarding MFW isolation. Only the impact on the core response was analyzed for all scenarios. When the reanalyses calculated the containment response for these cases, the results were more severe than the cases which were presented in the FSAR. Containment pressure could exceed 59 psig for these scenarios, without operator action.

Once these preliminary reanalyses were completed and determined to be valid, a meeting was convened at Oconee on March 11, 1993 to evaluate their impact.

It was then decided to perform additional analyses with the intent of quantifying the performance requirements on the operator and on the MFW control and block valves. The NRC Resident Inspector and Oconee Management were briefed on the reanalysis results. A second meeting was held on May 19, 1993 to discuss the results of these additional reanalyses. The following scenarios were considered at the second meeting:

1. Automatic Main Feedwater Scenario (Normal Operating mode)

   Operator action is required to initiate closure of the main feedwater control valves within 170 seconds to maintain reactor building pressure below 59 psig. Operator action would also be required to isolate Emergency Feedwater from the affected steam generator.

2. Manual Main Feedwater Control (less likely operating mode)

   Operator action is required to initiate closure of the main feedwater control valves within 25 to 30 seconds (followed by 20 second stroke time) to maintain reactor building pressure below 59 psig. Operator action would also be required to isolate Emergency Feedwater from the affected steam generator.

3. Manual Main Feedwater Control with stuck open control valve (probability of 4.0 E-10 events per reactor year)

   Containment design pressure (59 psig) exceeded. Operator action is required to initiate closure of the main feedwater block within 120 seconds (followed by 120 seconds stroke time) to maintain RB pressure approximately 140 psig. Operator action would also be required to isolate Emergency Feedwater from the affected steam generator.

4. Loss of Offsite Power with main steam line break

   Feedwater and Condensate System pumps trip due to loss of power. Operator must initiate closure of the main feedwater control valves within 170 seconds (followed by approximately 20 seconds valve stroke time) to maintain RB pressure below 59 psig. Operator action would also be required to isolate Emergency Feedwater from the affected steam generator.

The NRC was notified by telecon of the results of the second meeting. The results, consequences and proposed actions, summarized in this report, were submitted to the NRC in a letter dated May 27, 1993. It was determined that the MSLB accident analysis was not reportable based on the fact that the dose calculation assumes a MSLB outside containment, which was considered to be the worst case. On June 2, 1993 at 1630 hours, following discussions with the NRC and after consideration that the resultant containment pressure from this accident placed the plant in a condition outside the containment design basis, as described in the response to IE Bulletin 80-04, the postulated event was determined to be reportable.

On August 19, 1993, Duke Power Company submitted a revised response to IE Bulletin 80-04. Since the reanalysis identified that the potential for overpressurization exists without Integrated Control System and operator actions, corrective actions are necessary. In order to alleviate the reliance on operator action, feedwater isolation will be initiated by an automatic signal during a MSLB. A modification is planned for all three Oconee Units to accomplish the automatic feedwater isolation (including the block of the start of the Turbine Driven Emergency Feedwater Pump).

CONCLUSIONS

The root cause of this event is Design Deficiency; Functional design deficiency; Mechanical. During the initial design analysis of the Nuclear Steam Supply System, Babcock and Wilcox (B&W), the vendor, did not account for the passive structural metal in the primary system as a heat source. Also, the Final Safety Analysis Report (FSAR) calculation of containment response did not include worst case scenarios regarding Main Feedwater isolation.

This design error occurred prior to the operation of the first Oconee Unit in the early 1970's. The evaluation in 1980, based on IE Bulletin 80-04, only considered the results of continued feedwater flow for a main steam line break scenario relative to the existing FSAR analysis. Since the errors in the existing FSAR analyses were not identified at that time, the evaluation based on those analyses resulted in incorrect conclusions. The error was not discovered earlier because a complete reanalysis had not been undertaken until January 1993. While reanalyzing for longer fuel cycles and core design optimization the discrepancy was discovered. The B&W and Duke Power Company design processes and required documentation have been significantly upgraded since the original analyses. Therefore, Duke Power does not consider it necessary to make any corrective changes to its design process as a result of this event. Nuclear Services will continue the FSAR Chapter 15 reanalyses associated with the extended fuel cycles and reload design optimization. There is a potential that similar deficiencies may be found and corrected through this process.

This event is considered non-recurring.

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

CORRECTIVE ACTIONS

Immediate

None

Subsequent

None

Planned

1. A supplemental response to IE Bulletin 80-04 will be completed. (Completed August 19, 1993)

2. Revise/update the Final Safety Analysis Report as required.

3. Modify the Feedwater system to provide automatic isolation of feedwater and block the start of the Turbine Driven Emergency Feedwater Pump to the Steam Generators during a Main Steam Line Break.

SAFETY ANALYSIS

The main steam line break (MSLB) accident has been analyzed with several assumptions regarding Integrated Control System (ICS) and operator action to isolate feedwater. The worst case assumption for the steam line break accident, in addition to no operator action, is that the ICS does not perform its design function of controlling main feedwater (MFW) on steam generator level.

The results of the core response analysis show that the unit can successfully mitigate the transient without taking credit for ICS or operator action.

The containment design pressure is 59 psig. The results of the Oconee Individual Plant Evaluation, submitted in response to Generic Letter 88-20, show that the median ultimate strength of the Oconee containment building is 144 psig with a standard deviation of 18. No major failure of internal civil structures or components is predicted at this pressure. The equipment required to mitigate the consequences of the MSLB is environmentally qualified and would perform its safety function.

The results of the containment response analysis shows that the containment design pressure of 59 psig will be exceeded without operator action. In one scenario, operator action is required within 120 seconds to avoid exceeding the 144 psig median ultimate yield strength of the containment building. This scenario has a probability of 4.0E-10 events per reactor year.

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

The most probable scenario for a MSLB is when MFW is being controlled in automatic. Provided the operator manually isolates the MFW by closing the MFW control valves (20 second stroke time) beginning no later than 170 seconds, the peak containment pressure will remain below the design pressure of 59 psig. This operator action requirement is in current emergency operating procedures and is required to be committed to memory for all licensed operators. The estimated frequency of steam line break scenarios inside containment at Oconee, which exceed the design limit of 59 psig, is less than 1.0E-6 per year per unit. Furthermore, the accident scenarios do not involve core damage. Based on the above, offsite dose consequences of the MSLB inside containment are bounded by the current FSAR analysis and the event is not a significant risk to the public.