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AUTH.NAME AUTHOR AFFILIATION
LOWERY,H.R. Duke Power Co.
BARRON,H.L. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-013-00:on 911104,B&W notified util that boron precipitation inside core would occur sooner than previously analyzed after certain LOCA scenarios.Caused by design defect.Inoperability procedures revised.W/911126 ltr.

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AO

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DUKE POWER

November 26, 1991

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 269/91-13

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 269/91-13 concerning technical inoperability of the post LOCA Decay Heat Removal System.

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

Very truly yours,

H. B. Barron
Station Manager

/ftr

Attachment

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Oconee Nuclear Station, Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 2 6 9

PAGE (3): 1 OF 0 8

TITLE (4): Post LOCA Decay Heat Removal System Declared Technically Inoperable Due to Design Deficiency

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 NAMES	DOCKET NUMBER(S)
1 1	0 4	9 1	9 1	– 0 1 3 –	0 0	1 1	2 6	9 1	Oconee, Unit 2	0 5 0 0 0 2 7 0
									Oconee, Unit 3	0 5 0 0 0 2 8 7

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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)	[X] 50.73(a)(2)(ii)(c)	[ ] 50.73(a)(2)(viii)(B)	50.72(b)(1)(ii)(c)
[ ] 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	
Henry R. Lowery, Chairman Oconee Safety Review Group	8 0 3	8 8 5 – 3 0 3 4

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On November 4, 1991, at 1645 hours, Babcock and Wilcox (B&W) notified utilities that boron precipitation inside the core would occur sooner than previously analyzed after certain LOCA scenarios. As a result, the Post LOCA Boron Dilution System (BDS) must be in service significantly earlier than previously required to avoid boron crystallization on fuel assemblies and the resulting degraded heat transfer. Operators were instructed to place BDS in service within 90 minutes of the LOCA in accordance with preliminary guidance from B&W. On November 5, 1991, Duke Power calculated that BDS must be in service within 9 (rather than 24) hours after the LOCA. Corrective action was to revise procedures to initiate system operation within the new time limit. The Low Pressure Injection system was declared technically inoperable from installation of the BDS (in 1976) until November 5, 1991, because the BDS may not have been placed in service when needed. All three Oconee units were at 100 % full power when notified. The root cause is Design Deficiency.

BACKGROUND

On August 4, 1974, Babcock and Wilcox (B&W) submitted Topical Report BAW-10091 to describe the B&W Emergency Core Cooling System evaluation model with respect to Appendix K of 10CFR50. During subsequent correspondence an issue was raised concerning control of system chemistry following a LOCA and the corresponding effects on core cooling. If a large break occurred at the bottom of a cold leg (Reactor Coolant [EIIS:AB] Pump discharge pipe), a condition could exist where water from the Low Pressure Injection (LPI) [EIIS:BP] System would flow into the downcomer region of the reactor vessel [EIIS:VSL], then out the break, rather than into and through the core region. (See Attachment 1.) Some steaming would occur in the vessel, and steam would pass through the internal vent valves in the upper plenum. Over time, this steaming action would result in the increase in boron concentration in the core region. Eventually, the boron concentration would reach the limit of solubility, and boron would begin to precipitate out of solution. If allowed to continue, the boron precipitate could form crystal deposits on fuel assemblies and internals, which could lead to core damage by reducing heat transfer and/or blocking coolant flow paths.

Duke Power responded to this concern by installing a Boron Dilution System as a sub-system of the LPI system on all three units in 1976 to mitigate the possible consequences of boron buildup. This system modified the Reactor Coolant and LPI systems to provide two flow paths, each of which would allow flow to exit the core through a hot leg (Reactor Vessel outlet) and pass to the Reactor Building [EIIS:NX] sump for recirculation. This would allow adequate mixing flow in order to maintain the core region boron concentration below the solubility limit. The Duke Power response was based upon the evaluation in Topical Report BAW 10091, especially with respect to the time requirements for the system.

EVENT DESCRIPTION

On June 26, 1990, Duke Power Design Engineering (DE) discovered that the Post LOCA Boron Dilution System (BDS) on all three Oconee units did not meet the single failure criterion stated in the Final Safety Analysis Report. Valves on the two redundant trains were powered from the same motor control center, resulting in the potential that a failure of that power source could render both trains inoperable. This condition was reported as LER 269/90-011 dated July 26, 1990.

The short term solution to that problem was to provide instructions and procedures to align at least one of the BDS trains shortly after the accident if the appropriate accident scenario occurred. Additional procedural guidance was provided to allow compensatory measures, if the single failure occurred, to connect one valve to a different power supply within 24 hours, which would allow one train to operate. The long term solution is to change the normal power source for that valve to a different power source. Implementation of this long term solution is currently planned for the next refueling outage on each unit, which should occur in 1992.

As part of the evaluation of the electrical supply problem, DE performed calculations which led them to question the Babcock and Wilcox (B&W) analyses. The DE calculation indicated that only 45 hours may be available before reaching the boric acid solubility limit, rather than the 30 days indicated by B&W. DE forwarded these concerns to B&W on August 6, 1990 and B&W initiated a review of their calculations and the engineering assumptions used. Because the DE calculation was still greater than the procedural requirement of 24 hours, system operability was not in question.

On November 1, 1991, B&W issued Preliminary Safety Concern 2-91 as a letter to affected utilities describing the problem. This letter stated that it was possible that the solubility limit might be reached in a substantially shorter time than 30 days.

The principle difference in the new analyses compared to the previous analyses is the amount of internal mixing due to assumed flow through the internal vent valves in the upper plenum assembly above the core. The original B&W analysis used a correlation which indicated that the level of the boiling water/steam mixture would be elevated enough to produce some water flow through the vent valves to the vessel downcomer region. This would provide some mixing flow and significantly slow down the increase in boron concentration in the core region. In other words, the more concentrated borated water in the core would "boil over" through the vent valves and be diluted by the water in the downcomer region. The new analysis assumes that only steam would reach and pass through the vent valves. This steam would have virtually no boron content. This assumption means that virtually no mixing occurs, therefore the boron concentration in the core region would increase rapidly.

At 1645 hours on November 4, 1991, B&W notified utilities that the time to reach the solubility limit could be as short as 90 minutes. All three Oconee units were operating at 100 % full power at the time notification was received. The immediate response to the B&W notification was to provide interim administrative instructions to the operators on shift as to appropriate compensatory actions in case such a scenario occurred. The instructions given were that the BDS was to be lined up when initiating recirculation flow (i.e. when the Borated Water Storage Tank nears its minimum level and the LPI pump suction is re-aligned to the Reactor Building Sump). This should occur approximately twenty to thirty minutes into the event for a large break. DE personnel used the new B&W assumptions and conservatively calculated that the time to reach the solubility limit should be approximately 9 hours.

On November 5, 1991, DE completed required verification reviews of their calculations and confirmed the 9 hour time limit. B&W stated that their time of 90 minutes was based on a preliminary calculation which simply verified that the boron concentration at 90 minutes was less than the limit, and was not intended to represent the time at which the limit would be reached.

At approximately 1420 hours, upon notification from DE that the new time limit was less than the procedural guidance, Operations declared the LPI systems inoperable on all three units until changes in the Emergency Operating Procedures could be made. This placed all three units in a Limiting Condition for Operation (LCO) in accordance with Technical Specification 3.0 which requires that the condition be corrected or the unit be at hot shutdown within 12 hours. Due to the nature of the problem and the fact that procedure changes were expected to be processed rapidly, all three units remained at 100 % Full Power throughout the event.

Appropriate permanent procedure changes were made and approved to replace the interim administrative instructions. At 1600 hours the changes were in place and the units exited the LCO. If the assumed single failure of the power supply actually occurs during or after the accident, then the new instructions specify that the actions to connect to an operable power source must be complete and the flow path established within 9 hours. DE also completed an Operability Evaluation which concluded that the BDS is considered operable now, with the new procedural guidance, but that it must be considered to have been technically inoperable in the past because the applicable procedures had allowed 24 hours for placing the system in service.

CONCLUSIONS

The root cause of this event is Design Deficiency, (Functional Design Deficiency). During initial design of the Nuclear Steam Supply System, Babcock and Wilcox (B&W), the vendor, did not adequately design for potential boron precipitation. When questions arose in the 1975 to 1976 time frame, B&W used assumptions, which are now considered inappropriate, in specifying the design parameters for a mitigation system. In their November 1, 1991 letter, B&W concluded that the upper plenum geometric model used in computer calculations was not appropriate for this application.

This design error occurred prior to system installation in 1976. The design of an Oconee specific Boron Dilution System by Duke Power Design Engineering was based directly on the system requirements specified by B&W. The Duke Power design process and the documentation which it requires have been significantly upgraded since that time. Therefore, Duke Power does not consider it necessary to make any additional corrective changes to its design process as a result of this event.

This event is considered recurring. LER 269/91-10, "High Pressure Injection System Technically Inoperable For Some Single Failure LOCA Scenarios Due to Design Deficiency," documented a similar design deficiency which potentially could cause loss of an Emergency Core Cooling System during certain LOCA scenarios. Because the deficiency in this event has existed since 1976, no corrective action from these previously discovered events could have prevented it.

The Design Basis Document project is in the process of documenting the Oconee design basis, and is reviewing many calculations with a questioning attitude. It is anticipated that similar deficient calculations and/or assumptions may be found and corrected through this process.

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

CORRECTIVE ACTIONS

Immediate

1. Interim administrative instructions were provided, on November 4, 1991, to the operators on shift to take immediate compensatory actions should the applicable accident scenario actually occur.

Subsequent

1. Duke Power Design Engineering performed calculations to document the time available for placing the Boron Dilution System in operation.

2. Appropriate permanent procedures were revised, on November 5, 1991, to specify the new time limit and required actions.

Planned

1. The power supply to LP-104 will be changed on all three units as specified in LER 269/90-011 (Nuclear Station Modifications 12867, 22867, and 32867).

SAFETY ANALYSIS

This report addresses a design assumption affecting long term decay heat cooling after a certain class of large break LOCAs. The scenario requires that the break must occur in the bottom of a cold leg (cold water return from the steam generator). If this unlikely combination of conditions were to occur, however, the existing procedures would not have assured that corrective actions would have been completed within the necessary time frame. The appropriate emergency procedures specified that the Boron Dilution System (BDS) be placed into service within 24 hours after the initial accident. Due to the location of the step within the procedures and the desire to assure all necessary actions are taken within a conservative time frame, it is expected that, had an event occurred prior to the discovery of this deficiency, the operators would have attempted to place the system in service before 9 hours had elapsed. However, it cannot be assured that this would have occurred. The response could be further complicated due to the current susceptibility of the system to a single failure of an electrical power supply which would make both trains inoperable. In the event of the failure of the power supply, the required action to connect LP-104 (Train A Post LOCA Boron Dilution Valve) to another power source would have been weighed against other activities and may not have been given enough priority to be completed within 9 hours when the requirement was 24 hours.

If the BDS was not placed in service within 9 hours, it must be assumed that boron precipitation would have occurred. If allowed to continue, the boron precipitate could have formed crystal deposits on fuel assemblies and internals, which could lead to core damage by reducing heat transfer and/or blocking coolant flow paths. It is assumed that localized fuel overheating would have occurred resulting in clad damage and fuel melt.

The additional time for boron precipitation to cause fuel damage after the solubility limit has been reached has not been calculated. While it is conservatively assumed that fuel damage could occur immediately if the solubility limit is calculated to be reached, i.e. 9 hours into the accident, the actual time to fuel damage is expected to be much longer, i.e. greater than 24 hours, due to the following conservatisms:

1. No credit is taken for recirculation mixing through the gap between the plenum assembly and the hot leg nozzle on the reactor vessel itself. B&W originally attempted to justify that this flow path was sufficient by itself to prevent excessive boron concentration. The approach was not accepted by the NRC due to lack of empirical data to prove that the flow was adequate. However, it is expected that enough flow will occur to increase the time until the limit of solubility is reached.

2. No credit is taken for any boron content in the steam which is being passed through the vent valves. In actuality the steam will have some moisture content, and therefore, some boron content.

3. The solubility limit used in the calculation includes a 4 % (by weight) reduction recommended by the NRC. It does not include any allowance for the increase in solubility due to elevated containment pressures (5 to 12 psig) expected 9 hours after a large break.

4. A conservatively low core mixing volume is used.

If loss of heat transfer due to boron precipitation occurred between 9 and 24 hours into such an event, prior to the discovery of this error, the control operator, reactor engineer, and others in the Technical Support Center would have received some information to help them diagnose the onset of boron precipitation. There are several core exit thermocouples inside the core which should be operating after the accident. These should detect localized overheating. Post accident core damage mitigation training has included techniques for comparing the temperature distribution from these thermocouples to the pre-accident core power distribution data to determine flow and/or core cooling anomalies and hot spots. However, the diagnosis of the cause of the loss of heat transfer would have been more difficult because all available documentation would have shown that boron precipitation should not be occurring so soon. Therefore, it is possible that appropriate corrective actions to establish boron dilution would have been delayed.

While it is conceivable that some fuel damage could possibly have occurred, boron dilution flow would have been established as required by the existing procedures. The FSAR analysis for the Maximum Hypothetical Accident assumes a significant amount of fuel damage and demonstrates that the health and safety of the public would still be protected.

The scenario described above has not occurred. Even if it had, the consequences are bounded by FSAR analysis. Therefore, the health and safety of the public have not and would not have been affected by this condition.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

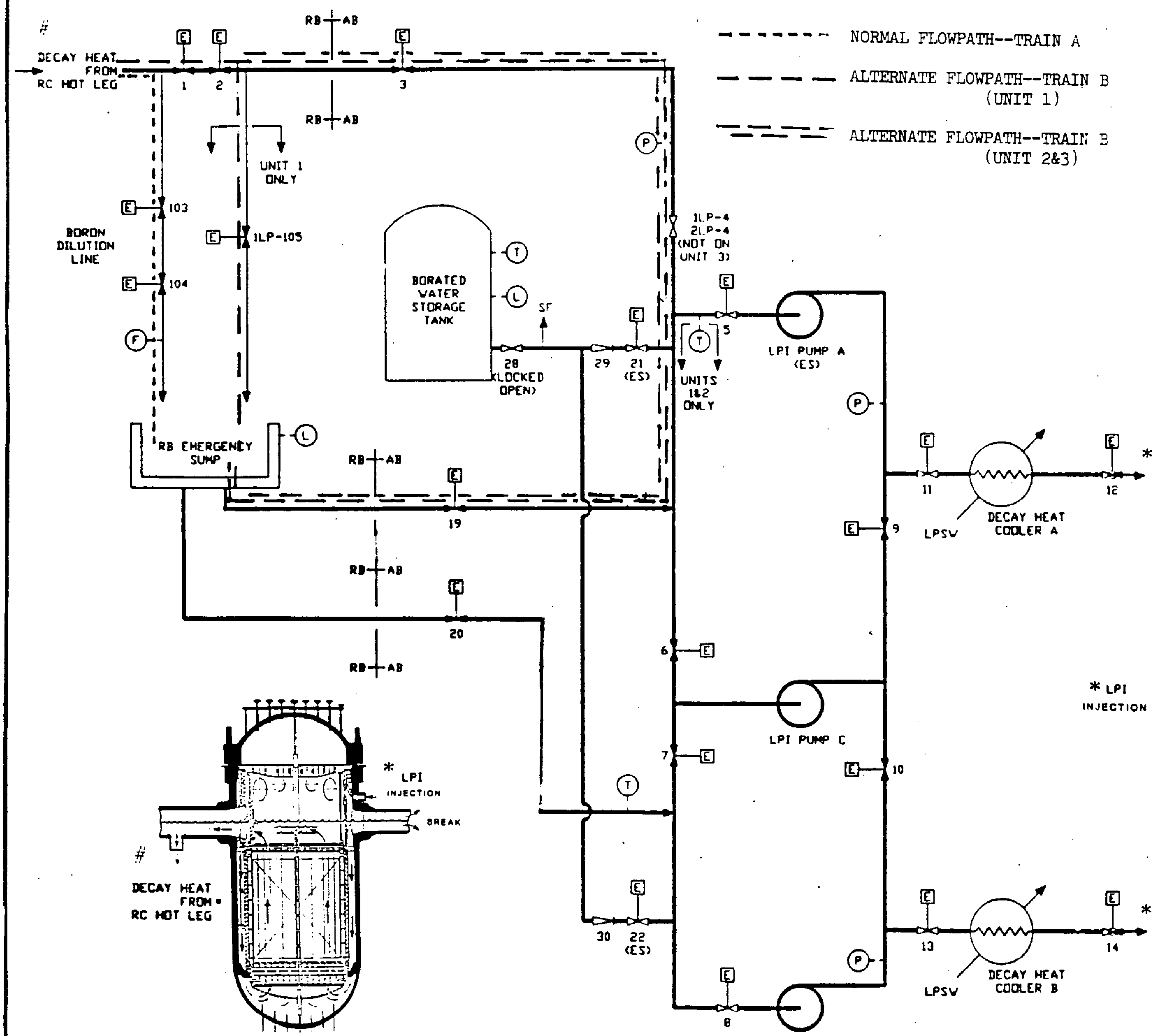
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 1	0 5 0 0 0 2 6 9	9 1	— 0 1 3	— 0 0	0 8 OF 0 8

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

ATTACHMENT 1

Boron Dilution Flowpaths



BORON DILUTION FLOW THRU THE REACTOR VESSEL