CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9705290213 DOC.DATE: 97/05/21 NOTARIZED: NO DOCKET #
FACIL:50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270

AUTH.NAME	AUTHOR AFFILIATION
BOND,R.	Duke Power Co.
HAMPTON,J.W.	Duke Power Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-001-00:on 970421,discovered unisolable leak in Reactor Coolant Sys.Caused by failure to implement effective surveillance program.Repaired nozzle components & will establish effective program to insp nozzles.W/970521 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 15
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 PD	1	1	LABARGE,D	1	1
INTERNAL:	ACRS	1	1	AEOD/SPD/RAB	2	2
	AEOD/SPD/RRAB	1	1	FILE CENTER	1	1
	NRR/DE/ECGB	1	1	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DRCH/HHFB	1	1
	NRR/DRCH/HICB	1	1	NRR/DRCH/HOLB	1	1
	NRR/DRCH/HQMB	1	1	NRR/DRPM/PECB	1	1
	NRR/DSSA/SPLB	1	1	NRR/DSSA/SRXB	1	1
	RES/DET/EIB	1	1	RGN2 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	LITCO BRYCE,J H	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 25 ENCL 25

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

J. W. Hampton
Vice President
(864)885-3499 Office
(864)885-3564 Fax

DUKE POWER

May 21, 1997

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

Subject: Oconee Nuclear Station Unit
Docket Nos. 50-270, -287
Licensee Event Report 270/97-01, Revision 0
Problem Investigation Process No.: 2-097-1324

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 270/97-01, concerning the completion of a plant shutdown required by the plant's Technical Specifications. This shutdown was due to an unisolable leak in the Reactor Coolant System, which was a condition resulting in a principal safety barrier being degraded.

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(A) and 50.73(a)(2)(ii). 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

/fts

Attachment

IE22

030015

Document Control Desk
Date: May 21, 1997

cc: Mr. Luis A. Reyes
Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S. W., Suite 23T85
Atlanta, GA 30303

Mr. D. E. LaBarge
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

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

Mr. M. A. Scott
NRC Resident Inspector
Oconee Nuclear Station

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit Two	05000 270	1 of 13

TITLE (4)
Unisolable Reactor Coolant Leak Due to Inadequate Surveillance Program

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(S)
									Unit Three	05000 287
04	21	97	97	- 001 -	00	05	21	97		05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

OPERATING MODE (9)	N							
POWER LEVEL (10)	100%							
			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 (*Specify in Abstract below and in Text, NRC Form 366A*)
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)A		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)	X	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	
Rick Bond, Safety Review Manager	(864)	885-3043

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
E	AB	PSP	X000	NO					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (*f yes, complete EXPECTED SUBMISSION DATE*)	X	NO				

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

On April 21, 1997, Unit 2 was at 100% Full Power(FP). At 2245 hours, Operators noted indications of a 2.5 gpm Reactor Coolant System (RCS) leak. The source could not be determined, so at 0352 hours on April 22, power reduction began. At 20%FP Operators could not identify the leak as isolable, so the decision was made to go to cold shutdown. At 1448 hours, the reactor was tripped by a planned test. At 1500 hours, a NOUE was declared when the leak exceeded 10 gpm. The NOUE was terminated at 2032 hours after the leak reduced below 10 gpm. The leak was found to be a crack at the safe end to pipe weld on the High Pressure Injection to RCS cold leg nozzle near Reactor Coolant Pump 2A1. The safe end and pipe were found to be cracked internally and the thermal sleeve was found to be loose and damaged. The failures were caused by thermal cycling fatigue. The root causes were determined to be failure to implement an effective HPI nozzle inspection program based on available industry recommendations and failure to effectively evaluate known problems and implement appropriate corrective actions. Corrective actions include repair of the nozzle components and establishing an effective program to inspect and support nozzles. Evaluation shows that the HPI line still had a factor of safety greater than 2 under design basis event loads. Prompt shutdown prevented the development of an unsafe condition.

<u>Background</u>

The High Pressure Injection (HPI) System [EIIS:BQ] controls the Reactor Coolant System (RCS) [EIIS:AB] inventory, provides the seal water for the Reactor Coolant Pumps [EIIS:P], and recirculates RCS letdown for water quality maintenance and reactor coolant boric acid concentration control.

The HPI System is also a part of the Emergency Core Cooling System (ECCS) which mitigates the consequences of loss of coolant accidents (LOCA). The HPI System prevents uncovering of the core for smaller break sizes, where high RCS pressure is maintained, and delays the uncovering of the core for intermediate break sizes. The HPI System, during emergency operation, supplies borated water to the RCS from the Borated Water Storage Tank (BWST). The HPI System has three parallel HPI pumps that have the capability to take suction from the BWST. The HPI pumps have the capability to discharge through two redundant flow headers into the RCS, utilizing four injection lines (two per header). The HPI headers are cross-connected by piping and associated valves to provide for remote manual alignment to ensure flow to the core through both HPI trains should a single failure of an HPI pump or HPI injection valve prevent automatic injection through one train.

The HPI System, during normal makeup operation, supplies borated water to the RCS from the Letdown Storage Tank (LDST). The "A" HPI header supplies normal RCS makeup flow, typically 15 to 20 gpm through each of the two lines. These lines are each equipped with a "bypass" or "warming" line that provides a minimum flow preset by procedure to 3 gpm. The "B" HPI header is for emergency use only, and has no bypass lines.

The HPI injection lines terminate at injection nozzle [EIIS:NZL] assemblies located on each of the reactor inlet pipes downstream of the Reactor Coolant Pumps. Each nozzle assembly consists of a carbon steel nozzle (inconel clad on inside), to which a stainless steel safe end is welded. The HPI piping is welded to the other end of the safe end. Inside the safe end is a stainless steel thermal sleeve, which extends into the main RCS flow path. The function of the thermal sleeve is to minimize thermal shock and stresses on the nozzle by transporting the relatively cold HPI water (approximately 100 to 120F) into the main flow path. There it will mix with the 555F RCS cold leg water. Without the sleeve, the HPI water would have a direct impact on the nozzle itself, producing unacceptable thermal stress on the nozzle material.

Technical Specification 3.1.6.1 states: "If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shut down within 24 hours of detection."

Technical Specification 3.1.6.2 states: "If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shut down within 24 hours of detection."

Technical Specification 3.1.6.3 states: "If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shut down, and cooldown to the cold shut down condition shall be initiated within 24 hours of detection."

Technical Specification 3.3.1 requires three HPI pumps and two HPI flow paths to be operable when RCS temperature is greater than 350 degrees with fuel in the core. Additionally, valves HP-409 and HP-410 in the cross-connect must be operable. This is based on considerations of potential small breaks at the Reactor Coolant Pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling. Included in the Technical Specification definition of operable is the requirement that essential auxiliary equipment, such as instrumentation and controls, be capable of performing its related support function.

Event Description

On April 21, 1997, at approximately 2245 hours, while Oconee Unit 2 was operating at 100% Full Power (FP), the Reactor Operator noticed a change in the rate of decrease of the Letdown Storage Tank (LDST) and an increase in the Reactor Building (RB) Normal Sump (RBNS) rate, followed by Reactor Building radiation monitor system alarms. At 2300, Operations entered the Abnormal Procedure (AP) on excessive Reactor Coolant System (RCS) leakage. RCS leakage indicated 2.36 gpm leak. At 2337 hours, it was determined that the RCS leak was greater than the Technical Specification (TS) 3.1.6.2 limits on unidentified leakage.

At 0215 hours on April 22, a RB entry was made which determined that a leak existed at the second grating level in the "A" Steam Generator Cavity, near Reactor Coolant Pump 2A1. However, a positive identification of the leak could not be made.

Unit shutdown was commenced at 0352 hours after a meeting with Operations shift personnel, Radiation Protection, the Shift Work Manager, and the Station Manager. The original intent was to reduce power to 15%FP, where radiation levels would be reduced to the point that personnel could enter the area to better identify (and, if possible, isolate) the leak, while still keeping the main turbine on line.

At 0426 hours, an Emergency Notification System call was made to the NRC to report a shutdown due to RCS leakage in excess of TS limits.

The power reduction was stopped at 20%FP, at approximately 0900 hours. With power stable, a more accurate leak rate calculation could be performed. A calculation completed at 0940 hours indicated that the leak rate had increased to 6.25 gpm. By 1048 hours it had increased slightly above 8 gpm.

Another RB entry was made to better determine the exact location of the leak. At 1217 hours, personnel in the RB informed the control room that it appeared to be on 2HP-127, the High Pressure Injection (HPI) block valve closest to the HPI injection nozzle on the 2A1 Reactor Coolant Pump cold leg. The decision was made to continue to cold shutdown.

At 1250 hours, Unit 2 turbine-generator was taken off line. At 1448 hours, the reactor was tripped, by a planned test, to meet commitments on control rod trip time testing.

At 1600 hours, a Notice of Unusual Event (NOUE) was declared when the leak increased above 10 gpm. The leak rate peaked at about 12 gpm at 1750 hours, then began decreasing as RCS system pressure was reduced while shutting down. The NOUE was terminated at 2032 hours after two consecutive leak measurements showed the leak had reduced below 10gpm. The RCS cooldown continued until the unit was at cold shut down. Subsequent entries into the RB, on the morning of April 23, identified the leak as being at the safe end to pipe weld at the 2 ½ inch OD, schedule 160 Stainless Steel HPI pipe to RCS cold leg nozzle near Reactor Coolant Pump 2A1.

A Failure Investigation Process (FIP) Team was created to investigate the root cause and a Recovery Team was created to address the necessary repair activities.

Over the night of April 27-28, the pipe was cut from the cracked portion of the safe end out to valve 2HP-127, including the tie to the minimum flow line connection. This section of pipe was sent to the Babcock and Wilcox (B&W) Lynchburg Research Center for failure analysis. Radiographic Tests (RT) and visual inspection of the 2A1 HPI nozzle thermal sleeve determined

that the thermal sleeve was loose. Also, the nozzle safe end had multiple internal cracks discovered during Dye Penetrant Test (PT). A contingency plan was implemented to weld a temporary cap on the safe end and refill the RCS above reduced inventory conditions, while developing plans and an implementation package for the thermal sleeve an safe end repairs.

Duke Power reviewed the potential impact of this problem on Units 1 and 3 and generated a Justification for Continued Operation, which indicated that it was appropriate, based on knowledge available at the time, to continue to operate both units.

Ultrasonic Tests (UT) on the 2A1 safe end also showed the internal cracking revealed by PT. UTs were performed on the safe ends of the other three nozzles (2A2, 2B1, 2B2). UT was also performed on the pipe to safe end welds and pipe back to the first isolation valve. No other lines on Unit 2 showed any rejectable indications.

On April 29, an internal video inspection was performed on the 2A1 thermal sleeve. It was found to be axially cracked through wall and to have holes where two pieces were missing. The 2A2, 2B1 and 2B2 HPI thermal sleeves were inspected by inserting video equipment through disassembled HPI check valves. No problems were found.

On May 1, the 2A1 thermal sleeve and safe end were removed and sent to the B&W laboratory for analysis. PT was performed on the nozzle inner radius (knuckle area) for indication of cracks on the cladding. None were found. Then the safe end and thermal sleeve were replaced.

One portion of the FIP activity was to review the maintenance and operational history of the HPI nozzles. This led to the review of documentation associated with similar problems with thermal sleeves and safe ends in 1982 at Crystal River 3 and Oconee, and in 1988 at Farley 2 and Davis Besse. These events were described in a series of NRC documents such as IE Information Notice 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants", Generic Letter 85-20, "High Pressure Injection/Make-up Nozzle Cracking in B&W plants", and NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and an associated set of reports by the Babcock and Wilcox (B&W) Owners' Group (BWOG). Reports related to Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," were also reviewed relative to this event.

Generic Letter 85-20 adopted recommended corrective actions listed in a BWOG report on the Crystal River and Oconee problems in 1982. An Augmented Inservice Inspection Plan was to RT the safe ends periodically to check for loosening of the thermal sleeves, as indicated by the existence (or lack) of a gap or separation in the rolled fit joint between the thermal sleeve and the safe end. Also, "adjacent piping" (i.e. pipe at/near the pipe to safe end weld) was to be UT inspected on the normal makeup lines (for Units 2 and 3). By Bulletin 88-08, the pipe to safe end weld (plus others) on the emergency HPI lines were to be periodically UT inspected.

A review of inspection schedules indicated that these inspections had been performed on Unit 2 in May of 1996. It was learned that the actual scope of the UT exams was not as extensive as described in the BWOG recommendations from the 1982 event. Specifically, the requirement to UT adjacent piping on the normal makeup lines was not included in the program. Therefore, the weld which was cracked and leaking had not been inspected since 1982.

Also, the criteria for reviewing the RTs of the safe ends were not well defined and the personnel currently performing the reviews did not observe indications that the condition of the 2A1 thermal sleeve was degraded in 1996.

A Level III RT inspector performed a reassessment of all the RTs of thermal sleeves performed since 1983 in order to evaluate the impact on Units 1 and 3. The reassessment found there had been no RT taken on Unit 1 since 1989 but the RTs taken between 1983 and 1989 indicated no degradation of any Unit 1 thermal sleeves. The reassessment of Unit 3 RT results indicated that the 3A1 thermal sleeve was potentially degraded. On May 1, at 1430 hours, Management made the decision to shut down Unit 3 for inspection.

The 3A1 thermal sleeve was found to be damaged. Indications on the inside of the 3A1 nozzle were found by PT. Additional UTs with increased scope were performed on the nozzle and indicated that the observed cracks were limited to the cladding and did not penetrate to the carbon steel base metal. Framatome Technology Inc. (formerly B&W Nuclear Technology) performed an evaluation of the indications and determined them to be acceptable with no repairs necessary. The UTs of the other Unit 3 nozzles found no rejectable indications.

Both the 2A1 and 3A1 nozzles were restored by installation of new safe ends, thermal sleeves, and associated piping.

Throughout this period, communications continued between Duke Power and the NRC relative to the Justification for Continued Operation for Unit 1. Duke concluded that continued operation was justified, but has agreed to shut down Unit 1, on or before June 14, 1997, for inspection of the HPI nozzle components.

Conclusion

There were no radiological overexposures, radioactive releases, or personnel injuries associated with this event. This event did involve a weld leak, however the weld leak does not meet criteria to be NPRDS reportable as an equipment failure.

Laboratory results indicate that the leaking weld crack was circumferential with a 360 degree inside diameter crack, of which approximately 77 degrees was through-wall to the outside diameter. The face of the crack was a brown color, indicating that the crack had propagated over a long period of time, believed to be longer than two years. The primary initiator of the crack was high cycle/low amplitude stresses consistent with thermal cycling in the weld region. Any contributory role of vibration was limited to final failure after the crack was virtually through wall.

The pipe from the weld to the warming line connection was found to have cracks, averaging 20% through wall, consistent with thermal cycling.

The 2A1 and 3A1 thermal sleeves' damage may have been initiated and propagated by localized thermal fatigue, with some contribution from flow induced vibration after the sleeves was loosened.

The FIP team concluded that the failure scenario was:

- High cycle thermal fatigue, resulting from thermal mixing of the warming line, makeup and Reactor Coolant System (RCS) flow, caused cracking in the pipe, pipe to safe end weld, and safe end and contributed to the thermal sleeve failure.

- Thermal mixing occurred in the thermal sleeve, safe end and piping due to varying operational conditions, including low makeup flow through the thermal sleeve.

- Thermal hot shocking (sudden significant increase in temperature) initiated thermal sleeve loosening.

- RCS flow induced vibration contributed to thermal sleeve failure after loosening.

- It is improbable that the weld, piping, or safe end damage would have occurred without concurrent damage to the thermal sleeve.

- The through wall crack propagated over a long period of time.

The root causes of the event were:

1. Failure to implement an effective HPI nozzle inspection program based on available industry recommendations.

   This event is similar to events (in 1982) which involved cracking of welds, pipe, and thermal sleeves at Oconee and Crystal River. The 1982 Oconee event did not result in through-wall leakage of RCS. However, the corrective actions from that event were intended to detect a similar crack prior to degradation to the point of leakage. Therefore, this event is considered recurring. The corrective actions from the 1982 event did not prevent this event because the actual corrective actions taken did not meet the letter or the intent of the recommended corrective actions.

   The Augmented Inservice Inspection (ISI) program set up after the 1982 event:

a) added periodic UT of the safe end of all normal make up nozzles, but did not define this to cover the base metal in addition to the safe end to nozzle weld.

b) did not include periodic UT of "some associated piping" for normal make up nozzles, which should have included the pipe to safe end

weld.

c) did not include periodic UT of the "cold leg ID nozzle knuckle transition" for the three nozzles where thermal sleeves had been replaced in 1982.

d) added periodic Non-Code RT of the safe end and thermal sleeves but did not provide adequate procedural guidance or acceptance criteria.

e) designated the refueling outage numbers when the inspections were to be performed but did not adequately document that these inspections had a fixed frequency and did NOT have the flexibility contained in the ASME Code.

Additions to the Augmented Inservice Inspection (ISI) program, required by Bulletin 88-08 after the Farley 2 event, were properly implemented but addressed the HPI emergency header nozzle lines only.

2. Failure to effectively evaluate known problems (industry and in-house experience) and implement appropriate corrective actions (during a period of 1982 through the present).

   Operating experience from Davis Besse, and Oconee data from 1990 collected for Bulletin 88-08, indicated that the normal makeup lines were subject to periods of thermal cycling due to unexpected phenomena. These problems were documented in B&W reports issued in 1990 and 1992. Recommendations in these reports were not acted upon. These recommendations, if acted upon, could have reduced the thermal cycling experienced by the HPI makeup lines.

NOTE: The Failure Investigation Process Team will issue a final written report which will provide more technical detail of the failure and the investigation. The above conclusions are based on data received to date.

Corrective Action:

Immediate:

1. Unit 2 was brought to cold shutdown to reduce the leak.

2. A Failure Investigation Process (FIP) team was initiated to investigate the cause of the leak.

Subsequent:

1. The 2A1 thermal sleeve, safe end, and cracked piping between 2HP-127 and the safe end were replaced.

2. The potential impacts on Units 1 and 3 were evaluated and a Justification for Continued Operation (JCO) was developed. Compensatory actions supporting the JCO were implemented under procedural controls, including increased testing and surveillance of potential Reactor Coolant System (RCS) leakage.

3. The need to closely monitor indicators of potential RCS leakage was stressed to the Operators on Units 1 and 3. They were also instructed to treat all leaks inside the Reactor Building as unisolable for immediate response.

4. Operators on Units 1 and 3 were instructed to be more aware of the need to maintain stable High Pressure Injection (HPI) make-up flow. Maintenance and testing that could upset HPI make-up flow was minimized.

5. Upon re-examination of the Radiographic Test (RT) indication on 3A1 thermal sleeve, Unit 3 was shut down and the sleeve and safe end replaced.

6. The JCO for Unit 1 was revised following discovery of the indication on Unit 3.

7. Additional inspections were performed on the other three nozzles on both Unit 2 and Unit 3.

8. Other B&W Owners' Group members were notified. A list of questions and answers related to HPI operation and thermal sleeve experience was shared among the members.

9. The current Inservice Inspection (ISI) program and associated commitments were evaluated. All other Augmented ISI commitments were found to be fully implemented.

Planned:

1. Complete the laboratory analysis on the sleeves, safe ends, and associated piping removed from Units 2 and 3 and issue a formal report on the findings.

2. Complete a comparison of the field ultrasonic test (UT) results with the metallurgical lab results of the 3A1 nozzle components. The intent of this comparison will be to make a judgment of the capability of UT examinations to locate safe end to piping, safe end, and safe end to nozzle flaws.

3. Unit 1 will commence shutdown on or before June 14, 1997 to perform RT and UT examinations on the HPI nozzle components.

4. Duke will submit a new HPI System nozzle component augmented inspection plan for all three units to the NRC no later than
   a) 30 days prior to the scheduled start of the next Unit 1 refueling outage, currently scheduled in September, 1997, or
   b) September 1,1997, whichever comes first.

5. Establish a more effective Engineering Support Program for nozzle and thermal sleeve inspection and assessment. This program should ensure the inclusion of up to date industry experience, operating conditions and component reliability.
6. During implementation of NSM-12975 on Unit 1, replace all the piping from the safe ends to valves 1HP-126, -127, -152, and -153 and perform concurrent video inspections of all safe ends and thermal sleeves.

7. Install temporary temperature instrumentation to monitor the makeup lines. Periodically evaluate the data.

Planned corrective actions 1 through 7 are considered to be NRC Commitment Items. These are the only NRC Commitment items contained in this report.

Safety Analysis:

This leak started small and the growth rate was slow enough that an orderly shut down could be performed without exceeding the capability of the normal make-up system. However, the leak was a non-isolable fault in a Reactor Coolant System (RCS) strength boundary and the leak rate did exceed Technical Specification limits. The leak rate also met the criteria of the Emergency Plan to be declared a NOUE. The leak was entirely within the reactor building containment and no radioactive releases were made.

Analysis performed by Structural Integrity Associates, a consultant to the Failure Investigation Process team, concluded that, even with the existing crack, the 2A1 High Pressure Injection (HPI) line had enough remaining strength to provide a factor of safety greater than 2 under design basis event loads. This provides reasonable assurance that the line would not have catastrophically broken, even during a design basis event. It can therefore be concluded that the HPI system, and this line specifically, was still capable of performing its Emergency Core Cooling System function.

If the leak had not been discovered or actions taken to reduce pressure in the system by shutting down the unit, this leak potentially could have grown to a 2 ½ inch pipe break (approximately 0.025 square feet), which would have constituted a small break Loss Of Coolant Accident. Breaks at this location are bounded by analyses in the Oconee UFSAR and are addressed in Emergency Procedures, which give detailed guidance to the Operators for responding to this break. This guidance is to trip the reactor coolant pumps, if not already tripped, and redirect HPI flow so that the majority of the flow goes through the unbroken header into the RCS. This limits the amount of injection flow and RCS inventory that would be pumped out the broken line. The UFSAR analysis concludes that this break can be handled without core damage.

This event does not fit the normal precursor analysis method because the leak was not large enough to be considered an initiating event and the event did not cause failure of a mitigation system.

In summary, even though this was a significant event, there was no actual radiological impact. Appropriate analysis, systems, and guidance existed to adequately mitigate this event and the equivalent worst case UFSAR scenario. Therefore, the health and safety of the public was not affected by this event.