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May 19, 2003

U.S. Nuclear Regulatory Commission
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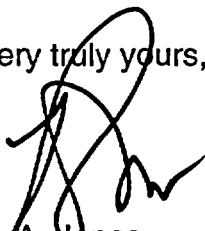
Subject: Oconee Nuclear Station
Docket No. 50-287
Licensee Event Report 287/2003-01, Revision 0
Problem Investigation Process No.: O-03-02557

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 287/2003-01, Revision 0, addressing the discovery of apparent reactor pressure vessel head leak from a control rod drive nozzle.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(i)(B) and (a)(2)(ii)(A). For this event, the overall safety significance of this event was minimal and there was no actual impact on the health and safety of the public.

Very truly yours,



R. A. Jones
Vice President
Oconee Nuclear Site

Attachment

IE22

Document Control Desk

Date: May 19, 2003

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cc: Mr. Luis A. Reyes
Administrator, Region II
U.S. Nuclear Regulatory Commission
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Mr. L. N. Olshan
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Mr. M. C. Shannon
NRC Senior Resident Inspector
Oconee Nuclear Station

INPO (via E-mail)

NRC FORM 366 (7-2001)		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004						
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)						Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.				
1. FACILITY NAME Oconee Nuclear Station, Unit 3					2. DOCKET NUMBER 050- 0287		3. PAGE 1 OF 5			
4. TITLE Apparent Reactor Pressure Vessel Head Leakage From A Control Rod Drive Nozzle										
5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTI AL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
05	02	03	2003 - 01	00		05	19	03	None	
9. OPERATING MODE 5			11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)							
10. POWER LEVEL 0			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)	
			20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)	
			20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)	
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)	
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A	
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)			
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)			
			20.2203(a)(2)(v)		<input checked="" type="checkbox"/> 50.73(a)(2)(i)(B)		50.73(a)(2)(vii)			
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)			
20.2203(a)(3)(i)		<input checked="" type="checkbox"/> 50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)						
12. LICENSEE CONTACT FOR THIS LER										
NAME L.E. Nicholson, Regulatory Compliance Manager						TELEPHONE NUMBER (Include Area Code) (864) 885-3292				
13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT										
CAUSE	SYSTEM	COMPONENT	MANU- FACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANU- FACTURER	REPORTABLE TO EPIX
B6a	RCS	NZL	B&W	Y						
14. SUPPLEMENTAL REPORT EXPECTED							15. EXPECTED SUBMISSION DATE		MONTH	DAY
YES (If yes, complete EXPECTED SUBMISSION DATE).					<input checked="" type="checkbox"/> NO					
16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)										
<p>Unit 3 entered its scheduled end-of-cycle 20 refueling outage on April 26, 2003. On May 2, 2003, a visual inspection of the bare reactor vessel head (RVH) was performed. Results of the visual inspection revealed two (2) control rod drive (CRD) mechanism (CRDM) nozzles that were suspected of leakage. Of these, CRDM No. 4 was observed to contain a very thin white coating on the nozzle and CRDM No. 7 appeared to have a small accumulation of boron on the head adjacent to the annulus region. In addition, approximately 6 to 8 CRDMs could not be visually inspected as they were masked by deposits from a Component Cooling (CC) system leak above the RV head.</p> <p>Subsequent evaluation of the prior refueling outage RVH inspection videotape showed that the CRDM No. 7 deposits were not associated with a new leak but rather were remnants from a prior refueling outage leak and repair campaign where the boron residue had not been completely removed from the RVH during the wash down process. The CRDM No. 4 boron deposit appeared fresher, exhibited characteristics similar to prior RVH leaks and as such, was conservatively identified as a leak. The apparent root cause of the nozzle leak is primary water stress corrosion cracking.</p> <p>This RVH will be retired from service and replaced with a new RVH prior to unit restart. For this event, the overall safety significance of this event was minimal and there was no actual impact on the health and safety of the public.</p>										

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
Oconee Nuclear Station, Unit 3	0500287	2003	- 01	- 00	2	OF 5

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

EVALUATION:

BACKGROUND

There are sixty-nine (69) Control Rod Drive (CRD) Mechanism (CRDM) [EIS:AA] nozzles [EIS:NZL] that penetrate the Reactor Vessel Head (RVH) [EIS:RCT]. The CRDM nozzles are approximately 5-feet long and are welded to the RVH at various radial locations from the centerline of the RVH. The nozzles are constructed from 4-inch outside diameter (OD) Alloy 600 material. The lower end of the nozzle extends about 6-inches below the inside of the RVH.

The Alloy 600 used in the fabrication of CRDM nozzles was procured in accordance with the requirements of Specification SB-167, Section II to the 1965 Edition including Addenda through summer 1967 of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code. The product form is tubing and the material manufacturer for the Oconee Nuclear Station (ONS) Unit 2 CRDM nozzles was the Babcock and Wilcox (B&W) Tubular Products Division.

Each nozzle was machined to final dimensions to assure a match between the RVH bore and the OD of each nozzle. The nozzles were shrunk fit by cooling to at least minus 140 degrees F., inserted into the closure head penetration and then allowed to warm to room temperature (70 degrees F minimum). The CRDM nozzles were tack welded and then permanently welded to the closure head using 182-weld metal. The manual shielded metal arc welding process was used for both the tack weld and the J-groove weld. During weld buildup, the weld was ground, and dye penetrant test (PT) inspected at each 9/32 inch of the weld. The final weld surface was ground and PT inspected.

The weld prep for installation of each nozzle in the RVH was accomplished by machining and buttering the J-groove with 182-weld metal. The RVH was subsequently stress relieved prior to the final installation of the nozzles.

EVENT DESCRIPTION

Oconee Nuclear Station Unit 3 (ONS-3) entered its scheduled end-of-cycle 20 refueling outage on April 26, 2003. On May 2, 2003, a visual inspection of the bare reactor vessel head (RVH) was performed, while bolted to the vessel, in order to determine if any of the sixty-nine (69) Control Rod Drive (CRD) Mechanism (CRDM) nozzle penetrations had developed a reactor coolant leak during the prior operating cycle. This inspection was performed looking through the nine access ports in the service structure support skirt on the RVH.

Results of the visual inspection revealed two (2) CRDM nozzles that were suspected of leakage. Of these, CRDM No. 4 was observed to contain a very thin white coating on the nozzle and CRDM No. 7 appeared to have a small accumulation of boron on the head adjacent to the annulus region. In addition, approximately 6 to 8 CRDMs could not be visually inspected as they were masked by deposits from a Component Cooling (CC) system [EIS:CC] leak above the RVH.

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		2003	- 01	- 00	

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

Subsequent evaluation of the prior refueling outage RVH inspection videotape showed that the CRDM No. 7 deposits were not associated with a new leak but rather were remnants from a prior refueling outage leak and repair campaign where the boron residue had not been completely removed from the RVH during the wash down process. The CRDM No. 4 boron deposit appeared fresher, exhibited characteristics similar to prior RVH leaks and as such, was conservatively identified as a leak. The apparent root cause of the nozzle leak is primary water stress corrosion cracking.

On May 2, 2003, after confirming that during power operations the Reactor Coolant System [EIS: RCS] pressure boundary had been degraded, an 8-hour notification (No. 39821) was made at 1957 hours (Eastern Time) in accordance with 10 CFR 50.72(b)(3)(ii)(A) reporting requirements.

Reportability

Technical Specification Limiting Condition for Operation 3.4.13(a) limits RCS operational leakage to "No pressure boundary leakage" while in MODES 1 through 4. This event also represents a degradation of one of the plant's principal safety barriers. Consequently, this event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(A) reporting requirements.

No operator intervention was required as a result of this event. Prior to the discovery of this event, Unit 3 was in cold shutdown (Mode 5) at 0 percent power and Units 1 and 2 were in Mode 1 operating at approximately 100 percent power.

ROOT CAUSE

Based on prior RVH evaluations described in previous, similar reported events (see below), the apparent root cause of the leaking Alloy 600 CRDM nozzle was Primary Water Stress Corrosion Cracking (PWSCC).

CORRECTIVE ACTIONS

The current RVH will be retired and replaced with a new RVH prior to ONS-3 restart.

SAFETY ANALYSIS

There were no actual safety consequences as a result of this event. The leakage of primary reactor coolant through the CRDM nozzle was so small that it was detectable only by the extremely small accumulation of boric acid crystals observed on the RVH. The total leakage from the CRDM nozzle did not exceed Technical Specification limits for unidentified RCS inventory loss. At no time during the operating cycle did the reactor building or area radiation alarms actuate as a result of this event. Neither the small amount of boric acid crystal deposits observed around the nozzle nor the deposits from the Component Cooling system leak caused visible corrosion damage to the RVH.

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

Since the current RVH will be retired from service and encapsulated for storage in the newly constructed steam generator / reactor head retirement facility, non-destructive examination (NDE) of the leaking nozzle was not performed primarily for personnel safety reasons and to minimize radiation exposure to workers in accordance with ALARA principles. The CRDM nozzle repairs made during the previous repair campaigns are shown on Figure 1.

ADDITIONAL INFORMATION

This event did not include a Safety System Functional Failure nor involve a personnel error. There were no releases of radioactive materials, radiation exposures in excess of limits or personnel injuries associated with this event. This event is considered reportable under the Equipment Performance and Information Exchange (EPIX) program. Energy Industry Identification System (EIIIS) codes are identified in the text as [EIIIS:XX].

SIMILAR EVENTS

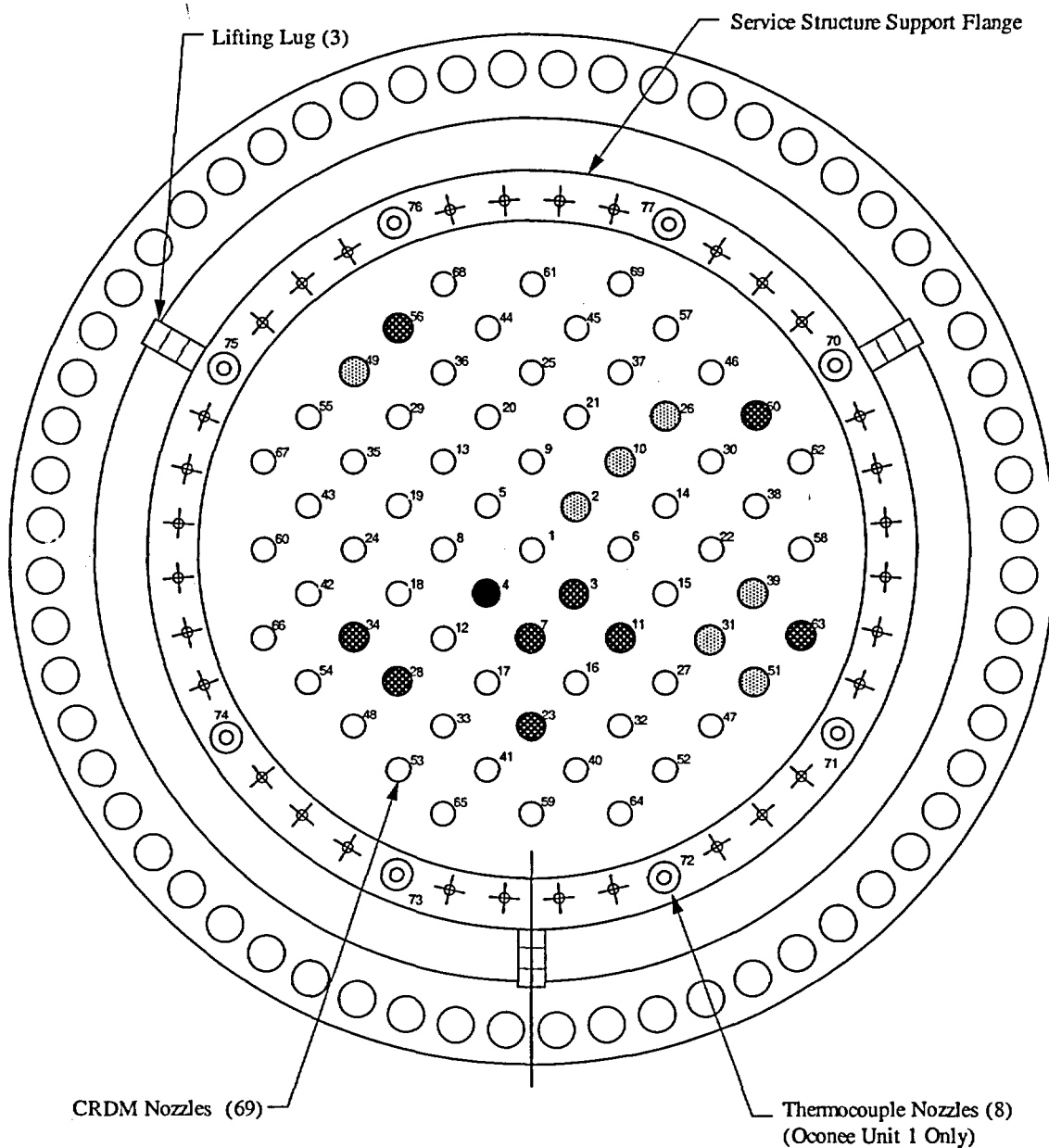
Over the last two and one half years, similar event LERs have been submitted for all three Oconee units beginning with Unit 1 in December 2000 (LER 269/2000-06) and the last for Unit 2 in December 2002 (LER 270/2002-02). To date, three (3) Unit 1, two (2) Unit 2, and three (3) Unit 3 LERs, have been submitted to the NRC which have reported PWSSC of Alloy 600 CRDM and/or thermocouple nozzles (Unit 1 only). Prior to these LERS, there have been no other reportable events that involved PWSSC of Alloy 600 components or RVH penetration leaks.

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FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)	
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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

Figure 1 – Oconee Unit 3 RVH Map



- Nozzles 3, 7, 11, 23, 28, 34, 50, 56, and 63 were repaired (maintenance outage)
- ▨ Nozzles 2, 10, 26, 31, 39, 49 and 51 were repaired (EOC 19 refueling outage)
- Nozzle 4 was not repaired since the RVH is being retired (EOC 20 refueling outage)