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

Subject: Oconee Nuclear Station, Unit 1
Docket No. 50-269
Licensee Event Report 269/2000-006, Revision 0
Problem Investigation Process No.: O-00-04134

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 269/2000-006, Revision 0, concerning the discovery of minor reactor coolant system pressure boundary leakage due to stress corrosion cracks found in several small bore reactor vessel head penetrations. This event was determined to be reportable on December 4, 2000.

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

Very truly yours,

W. R. McCollum, Jr.

Attachment

IE22

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
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Mr. M. C. Shannon
NRC Senior Resident Inspector
Oconee Nuclear Station

INPO (via E-mail)

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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TITLE (4)
Reactor Coolant System Pressure Boundary Leakage Due To Cracks Found in Several Small Bore Reactor Vessel Head Penetrations

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 05000
12	04	00	2000	- 006 -	00	1	02	01	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
L.E. Nicholson, Regulatory Compliance Manager	(864) 885-3292

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B6a	RCS	NZL	B&W	Y					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		3	30	2001

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

On November 25, 2000, at 0300 hours, with Oconee Unit 1 in Mode 5 and preparing to enter refueling outage 19, a periodic visual inspection of the top surface of the Reactor Pressure Vessel (RPV) head revealed small amounts of boric acid deposited on the vessel head surface. These deposits appeared to be located at the base of at least 4 unused thermocouple (T/C) and the #21 Control Rod Drive Mechanism (CRDM) nozzles at points where they penetrate the RPV head surface. A more detailed video inspection, performed on December 1, 2000, confirmed the presence of the boric acid crystals around the suspect nozzles.

On December 4, 2000, an eddy current test was performed on the inside surface of all 8 T/C nozzles and revealed axial crack-like indications on the ID of the nozzles in the vicinity of the partial penetration weld (on the underside of the RPV head). On December 9, 2000, dye penetrant (PT) testing on CRDM #21 identified two very small pin hole indications. After lightly grinding and performing another PT, a 0.75-inch radial indication running at a slightly skewed angle across the fillet weld was identified. Although the root cause investigation is ongoing, Primary Water Stress Corrosion Cracking is considered to be the primary failure mechanism. As a planned corrective action, following characterization of the failure mechanism, appropriate repairs will be made to the #21 CRDM weld and the T/C nozzles. These repairs will be accomplished prior to unit restart. This event is considered to have no significance with respect to the health and safety of the public.

EVALUATION:

BACKGROUND

Thermocouple Nozzles

A T/C nozzle is a three-quarter inch diameter schedule 160 pipe machined to a controlled diametrical fit with the bore in the reactor pressure vessel (RPV) head [EIIS: RCT]. The nozzle [EIIS: NZL] material is SB-167 (alloy 600). A total of eight thermocouple [EIIS: THC] nozzles were installed in the Unit 1 RPV head. These nozzles are located outboard of the head's Control Rod Drive [EIIS: DRIV] Mechanisms (CRDMs). The material was procured to the 1965 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code.

The original T/C nozzles were intended to provide instrumentation access into the vessel in order to verify that the internal reactor vessel plenum vent valves were not leaking. This action was subsequently determined to be unnecessary and blind flanges were added to the T/C nozzles that established the Reactor Coolant System (RCS) [EIIS: AB] pressure boundary. The nozzles serve no current function other than part of the RCS pressure boundary. Similar T/C nozzles and penetrations do not exist on either the ONS-Unit 2 or ONS-Unit 3 RPV heads.

A typical T/C head penetration consists of a 1.03-inch outside diameter (OD) by 0.218-inch nominal wall alloy 600 pipe that is inserted vertically into the RPV head and connected to the inside diameter (ID) surface by a J-groove weld. All 8 thermocouple penetrations have an overall length of approximately 62-inches. Approximately 8 inches of each penetration extends past the ID surface of the J-groove weld located on the inside surface of the RPV head. Each thermocouple penetration has a flange welded to the top of the pipe that extends the penetration length by approximately 2-inches.

The alloy 600 was procured from Huntington Alloys as cold drawn, ground, and annealed pipe. While specific heat treatment records are not available, the typical final annealing temperature for alloy 600 materials produced by Huntington Alloys is known to be 1600 degrees F. minimum.

CRDM Nozzles

There are 69 Control Rod Drive Mechanism (CRDM) [EIIS: AA] nozzles that penetrate the RPV head. The CRDM nozzles are approximately 5-feet long and are welded to the RV head at various radial locations from the centerline of the RV head. The nozzles are constructed from 4-inch OD alloy 600 material. The lower end of the nozzle extends about 6-inches below the inside of the RV head.

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 ASME B&PV Code. The product form is tubing and the material manufacturer for the ONS Unit 1 CRDM nozzles was the Babcock and Wilcox (B&W) Tubular Products Division.

The weld prep for installation of each nozzle in the RPV head was accomplished by machining and buttering the J-groove with alloy 182. The RPV head was subsequently stress relieved.

Each nozzle was machined to final dimensions to assure a match between the RPV head bore and the OD of each nozzle. The nozzles were shrink fitted 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 welded to the closure head with alloy 182-weld metal. The shielded manual 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 (PT) inspected at each 9/32 inch of the weld. The final weld surface was ground and PT inspected.

EVENT DESCRIPTION

On November 25, 2000, at 0300 hours, with Oconee Nuclear Station (ONS) Unit 1 in Mode 5 and preparing to enter refueling outage 19, a periodic visual inspection of the top surface of the Reactor Pressure Vessel (RPV) head revealed small amounts of boric acid deposited on the vessel head surface. The RPV head inspection was performed in response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

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

The deposits appeared to be located at the base of at least 4 (of 8) unused thermocouple (T/C) nozzles and 1 (of 69) Control Rod Drive Mechanism (CRDM) nozzle at points where they penetrate the RPV head surface. A more detailed video inspection, performed on December 1, 2000, confirmed the presence of the boric acid crystals around the base of the suspect nozzles. The maximum volume of boric acid observed was around the base of CRDM #21 and was estimated to be approximately 0.5 cubic inches. The amounts around the T/Cs were much less.

Although a December 1, 2000, RPV head inspection (while on the stand) did not reveal any apparent leak path for the boric acid to have originated from a CRDM flange leak, a December 4, 2000, eddy current test (ECT) performed along the inside surface of the T/C nozzles revealed axial crack-like indications in all 8 of the T/C nozzles. These small cracks were primarily axially oriented and located in the vicinity of the T/C's partial penetration weld on the underside of the RPV head. After confirming that Reactor Coolant System (RCS) pressure boundary leakage had occurred, a 4-hour notification was made to the Staff in accordance with 10CFR50.72(b)(2)(i) reporting requirements.

On December 7, 2000 an ECT inspection of CRDM nozzle #21 was performed and did not locate any surface indications or potential leakage pathways. A December 9, 2000 dye penetrant test (PT) of the under head region, including the nozzle fillet weld cap and partial penetration J-groove, identified two very small pin hole indications. After lightly grinding and performing another PT, a 0.75-inch radial indication running at a slightly skewed angle across the fillet weld was identified.

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. Additionally, the cracks found in the T/C and CRDM nozzles indicated a degradation of one of the plant's principal safety barriers. Accordingly, this event is being reported pursuant to 10CFR50.73(a)(2)(i)(B), and 10CFR50.73(a)(2)(ii)(A).

Prior to the discovery of this event, Unit 1 was in cold shutdown (Mode 5) at 0% power. Additionally, there was no operator intervention required as a result of this event.

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

CAUSAL FACTORS

Although the root cause investigation is ongoing, Primary Water Stress Corrosion Cracking (PWSCC) is considered to be the most likely failure mechanism of the alloy 600 T/C nozzles and for CRDM #21 weld.

Alloy 600 is used in nozzle applications in the reactor vessel, pressurizers, hot leg and cold leg piping, and for steam generator tubing. It is recognized in the industry that small-bore, alloy 600 shapes are susceptible to PWSCC and have experienced numerous cracking incidents as evaluated and documented in numerous failure analyses.

Industry data has shown that PWSCC (in nozzle applications) generally initiates at the inside surface of the nozzle opposite the partial penetration (J-groove) weld. This area has been shown to have very high residual stresses resulting from the weld process and in some cases from surface distress from machining, grinding or reaming operations. In thin wall product forms, this area could also have an altered microstructure from welding (i.e., weld heat affected zone). It is well established that PWSCC can occur in materials provided that three conditions are present:

(1) susceptible material,
(2) high tensile stress, and
(3) an aggressive environment.

Virtually any alloy 600 small-bore nozzle attached with a partial penetration weld that sees RCS inventory will have these characteristics.

CORRECTIVE ACTIONS

Subsequent:

1. A Failure Investigation Process (FIP) Team was immediately assembled to assess the event including its cause(s), necessary corrective actions, and past/future unit operational impacts.

2. A combination of eddy current, ultrasonic and/or dye penetrant inspections were performed on the 8 T/C nozzles and CRDM #21.

3. Seven additional CRDMs were eddy current tested and 18 CRDM nozzles were UT tested (for extent-of-condition purposes).

Planned:

1. Following characterization of the failure mechanism, appropriate repairs will be made to the #21 CRDM weld and the T/C nozzles. These repairs will be accomplished prior to unit restart.
2. This LER will be supplemented by March 30, 2001 to provide the final characterization information described above as well as any additional corrective actions.

These corrective actions are considered NRC commitments. At this time, there are no other NRC Commitment items contained in this LER.

SAFETY ANALYSIS

As described in NUREG/CR-6245, an axial crack is not likely to grow above the RPV head to a critical length because the stresses are not high enough to support the growth away from the attachment weld. The small volume of boric acid crystal deposits confirms that the leak rates were very low. Also, there was no evidence of significant corrosion products in the boric acid crystals suggesting that boric acid corrosion in the T/C nozzle annulus was low (as would be expected with very low leak rates at high temperature).

Additionally, Safety Analyses submitted by the industry in the early 1990's show that RPV head nozzle axial cracks do not pose a safety risk since,

- leak rates from cracks within the annulus region of nozzles are low,
- axial cracks extending beyond the annulus will be detected by leak-before-break before there is a risk of failure, and
- leakage from cracked nozzles is predicted to result in boric acid corrosion rates sufficiently low that the leakage could continue for up to six years without affecting the structural integrity of the RPV head.

Evaluation of the 0.75-inch crack found in the #21 CRDM weld determined that it was generally following a growth pattern characteristic of axial cracks that have been extensively analyzed in CRDM nozzles. The primary difference is that the crack is in the weld and has partially penetrated into the nozzle wall but never reached the ID of the nozzle (i.e., it did not penetrate through the nozzle wall). The crack has followed a predominately axial direction as it traversed through the weld material. This crack did not pose a significant safety risk in that, similarly to an axial crack, it exhibited the same characteristics, e.g., had a very low leak rate and followed the same leak-before-break type growth pattern.

Consequently, the overall safety significance of this event was determined to be minimal and there was no actual impact on the health and safety of the public due to this event.

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

ADDITIONAL INFORMATION

This event did not include a Safety System Functional Failure.

There were no releases of radioactive materials, radiation exposures or personnel injuries associated with this event.

This event is considered reportable under the Equipment Performance and Information Exchange (EPIX) program.

SIMILAR EVENTS

A review of LERs over the past two years did not identify any occurrences of past PWSCC, alloy 600 material cracks, or leaks involving RPV head penetrations, e.g., thermocouples and/or CRDMs. There were two (2) reports that dealt with reactor coolant system leaks. LER 269/98-02-001 reported a RCS nonisolable weld leak involving a 1-inch Schedule 160 Type 316 stainless steel Pressurizer Surge Line drain line. The root cause of the weld failure was mixed mode externally initiated stress corrosion cracking and fatigue propagation after the crack was initiated. A second LER, 269/2000-001-00 involved a nonisolable vapor leak on a 1.5-inch Schedule 160 1B2 cold leg loop drain line 90-degree elbow on a low point of the RCS. The root cause for this event was attributed to thermal fatigue due to cyclical turbulent penetration of hot RCS water into the stagnant drainpipe.

In conclusion, a review of these LERs did not reveal information that generally matched the criteria reported in this LER. Additionally, the corrective actions in these reports would not have been expected to identify or correct the event identified in this LER. Therefore, this event is considered to be non-recurring.

Energy Industry Identification System (EIIS) codes are identified in the text as [EIIS:XX].