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U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555

Subject: Oconee Nuclear Station, Unit 3
Docket Nos. 50-287
Licensee Event Report 287/2001-003, Revision 0
Problem Investigation Process Report No. O-01-04220

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 287/2001-003, Revision 0, addressing the discovery of minor reactor pressure vessel head leakage around several control rod drive nozzle penetrations.

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,

W. R. McCollum, Jr.

Attachment

IE22

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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)

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TITLE (4)
Minor Reactor Pressure Vessel Head Leakage From Several Control Rod Drive Nozzle Penetrations Due to Primary Water Stress Corrosion Cracking

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
11	12	01	2001	- 003 -	00	01	09	02	FACILITY NAME	DOCKET NUMBER

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)
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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
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

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

On November 12, 2001, a visual inspection of the top surface of the Oconee Nuclear Station Unit 3 Reactor Vessel (RV) head found evidence of small accumulations of boric acid deposited at the base of several control rod drive mechanism (CRDM) nozzles. This RV head inspection was performed as part of a planned surveillance activity during the end-of-cycle 19 refueling outage.

Following this visual inspection, nondestructive examination of the suspect nozzles revealed that seven (Nos. 2, 10, 26, 31, 39, 49, and 51) of the sixty-nine total nozzles required repair. Five of the seven repaired nozzles were confirmed to have a leakage pathway to the top of the RV head. The amount of boric acid around the five leaking nozzles was estimated to be no more than a few cubic inches. After confirming that the Reactor Coolant System pressure boundary had been degraded during power operations, an 8-hour notification was made at 0335 hours on November 12, 2001 in accordance with 10CFR50.72(b)(3)(ii)(A) reporting requirements.

The apparent root cause of the CRDM Nozzle leaks is primary water stress corrosion cracking. The seven CRDMs were repaired and the remaining 43 nozzles that [historically] had neither been previously examined nor repaired were inspected using an ultrasonic circumferential blade probe prior to exiting the refueling outage. This event is considered to have minimal safety significance with respect to the health and safety of the public.

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EVALUATION:

BACKGROUND

There are 69 Control Rod Drive Mechanism (CRDM) [EIIS:AA] nozzles [EIIS:NZL] that penetrate the Reactor Vessel (RV) [EIIS:RCT] 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 outside diameter (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 Oconee Nuclear Station Unit 3 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 RV head 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 RV head was accomplished by machining and buttering the J-groove with 182-weld metal. The RV head was subsequently stress relieved prior to the final installation of the nozzles.

EVENT DESCRIPTION

Unit 3 entered the scheduled end-of-cycle (EOC) 19 refueling outage on November 10, 2001. A visual inspection of the Unit 3 reactor vessel head was performed November 12, 2001 to identify any indications of leakage from the CRDM nozzle penetrations. A qualified visual inspection was performed through the nine access ports in the service structure support skirt of the reactor vessel head. The general cleanliness condition of the head was such that probable leak locations would be readily identified.

As a result of this inspection, four CRDM nozzle penetrations (Nos. 26, 39, 49 & 51) were identified with a high probability of leakage through the pressure boundary, either through the attachment weld or the nozzle wall. Additionally, three other nozzles (Nos. 2, 10 & 46) had boron crystal accumulation that could have been caused by, or masked, any minor leakage present. All seven nozzles were identified as requiring further inspections as specified by Duke's response to NRC Bulletin 2001-01. After confirming that the Reactor Coolant System (EIIS:AB) pressure boundary

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had been degraded during power operations, an 8-hour notification was made at 0335 hours on November 12, 2001 in accordance with 10CFR50.72(b)(3)(ii)(A) reporting requirements.

Ultrasonic test (UT) inspections of the inside diameter (ID) of nine CRDM housings were performed using the Framatome ANP "Top-Down Tool." Nozzles 26, 39, 49, & 51 were UT inspected due to being identified as having a high probability of leakage by the visual inspection; Nozzles 2, 10 and 46 due to masking; and Nozzles 29 and 31 for extent of condition (their CRDMs were removed to allow access for repair equipment). Ultrasonic Test (UT) scans were performed using a battery of transducers looking in both the axial and circumferential directions along with one zero degree straight beam transducer. Nozzles 29 and 46 had no UT indications. Nozzles 2, 26, 39, 49, and 51 all had indications that extended from below the weld to above the weld indicating a leak path in addition to various other ID and OD indications. Nozzle 2 had a circumferential indication in the nozzle above the J-groove weld. Nozzles 10 and 31 contained several OD indications located below the weld and extending slightly into the weld but showed no leak path.

From the underside of the reactor vessel head, a dye-penetrant test (PT) inspection of Nozzles 10, 31 and 46, including the fillet weld cap and partial penetration J-groove weld, was performed. PT results for nozzles 10 and 31 showed small nozzle OD flaws that ran up to the J-groove weld region at the weld to nozzle wall interface. No PT indications were found for Nozzle 46.

A conservative decision was made to repair Nozzles 10 and 31 although there were no indications of leakage from visual or UT inspections. This decision was primarily based on the small axial indications on the OD nozzle surface and the comparison of this data to previous ONS nozzle inspections that showed that these types of active PWSCC flaws could eventually result in a leakage pathway. Nozzle 29 was not repaired since there was no indication of leakage on the head and there were no indications recorded using the top-down UT tool. Nozzle 46 was not repaired since visual, UT and PT inspections showed no rejectable indications. Due to improvements in the UT inspection techniques in identifying and characterizing leakage paths along the nozzle OD volume, Eddy Current Test (ECT) inspections were not performed for any of the leaking or suspected leaking nozzles. This decision was also based on ALARA considerations since it was determined that ECT would have significantly increased the radiation dose to inspection personnel. Consequently, repairs made to nozzles were based on visual, UT and PT inspection results.

Nozzles 2, 10, 26, 31, 39, 49 and 51 were repaired utilizing a similar process used for the ONS-2 CRDM nozzle repairs performed in May 2001 (ref. LER 270/2001-002). The protruding portions of the nozzles and a length about 5 inches into the RV Head bore were removed by machining. A new pressure boundary weld was installed within the bore, inspected and surface conditioned with a water jet peening process.

As a result of the circumferential flaw found on Nozzle 2, an extended scope inspection was performed that utilized Framatome-ANP's ARAMIS equipment. ARAMIS is a remote "under the head" system designed to deliver an UT blade probe. The scope of the inspection involved the 43 remaining nozzles that [historically] had neither been previously repaired or volumetrically inspected. Circumferential blade probes were used to inspect the nozzle area from one inch above the top of the J-groove weld to one inch below the bottom of the J-groove weld. Thirty-six of the nozzles were inspected with 100 percent of the coverage area being examined. There were seven

nozzles where 100 percent inspection of the coverage area could not be achieved due to limited access inside the nozzle annulus (between the CRDM nozzle ID and the leadscrew support tube of the CRD mechanism itself). Approximate percentages of the coverage area inspected for these nozzles were:

Nozzle 62	82%
Nozzle 45	94%
Nozzle 69	75%
Nozzle 60	76%
Nozzle 42	94%
Nozzle 66	89%
Nozzle 48	99%

Overall results revealed no indication within the nozzle material for the 43 nozzles inspected. This nondestructive examination (NDE) was performed as added assurance that there were no existing circumferential flaws that could potentially pose a safety risk during the upcoming operating cycle.

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 (Reactor Coolant System). Consequently, this event is being reported pursuant to 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(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.

CAUSAL FACTORS

The apparent root cause of the indications found in seven Alloy 600 CRDM nozzles is Primary Water Stress Corrosion Cracking (PWSCC).

General cause of event discussion:

Alloy 600 is used extensively in nozzle applications in Reactor Vessel, Pressurizer [EIIS:PZR], hot and cold leg piping, and Steam Generator (EIIS:SG) tubing. It is recognized that small-bore nozzles have succumbed to numerous cracking incidents and the industry has evaluated and documented the results of many failure analyses. The conclusion resulting from this work is that the failure mechanism is a form of stress corrosion cracking referred to as PWSCC.

PWSCC can initiate on Alloy 600 surfaces exposed to primary water at high temperatures that have high residual stresses due to welding. Cold working of the surface by machining, grinding or reaming operations prior to welding may result in higher residual stress.

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 small-bore Alloy 600 nozzle (including CRDM nozzles) attached with a partial penetration weld possesses these characteristics. In PWR applications, numerous small-bore Alloy 600 nozzles and Pressurizer heater sleeves have experienced leaks attributed to PWSCC. Generally, these components are exposed to 600 degree F or higher temperatures and primary water, as were these CRDM nozzles.

Specific discussion regarding the apparent cause of event reported in this LER:

For this event, the apparent root cause of PWSCC is substantiated based on,

1. Comparison of the current NDE data with ONS CRDM inspections as documented in previous ONS-1, -2, and –3 root cause evaluations.
2. Correlation of the current crack location and orientation with previous Finite Element Analyses (FEA) documented in the events referenced above, and
3. The recent history of CRDM cracking found in Alloy 600 weld metal attributed to PWSCC at ONS and other Pressurized Water Reactors.

The investigation into the ONS-3 RV head leakage revealed information that continued to support conclusions documented during the ONS-3 and ONS-2 RV Head repairs in early 2001. Six supporting points from this most recent ONS-3 outage include:

1. Although both ID and OD cracks were observed, most of the cracks appeared on the nozzle OD.
2. Minor circumferential cracking above the structural J-groove weld was found (Nozzle 2).
3. The single circumferential flaw above the weld nozzle had adjacent axial cracking.
4. Axial cracking was present without adjacent circumferential flaws.
5. The current nozzles had fewer, smaller and shallower flaws than earlier ONS-3 nozzles.
6. The volume of nozzle leakage is small for these leaks and a visual inspection of a clean RV head is still the best means to determine leakage.

NDE (primarily UT) revealed that the leak paths were within the nozzle wall region where FEA (including the effects of welding residual stresses and operating conditions) predict high hoop stresses. The crack geometry is consistent with the analysis that shows the hoop stress (that drives cracks in the axial orientation) is higher than the axial stress (that drives cracks circumferentially) at high stress locations. Crack growth into the nozzle wall is also consistent with analysis predictions that high hoop stresses extend through the weld material and into the nozzle wall. The mostly axially oriented cracks are consistent with FEA results, and with a root cause determination of PWSCC. The single circumferential CRDM nozzle flaw was OD initiated.

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The current ONS-3 failure modes are consistent with the previous 2001 ONS-3 CRDM cracking (ref. LER 287/2001-001) and the 2001 ONS-2 cracking (ref. LER 270/2001-002). The initial UT inspections identified other indications in the weld material. These indications were determined to be acceptable pre-existing indications in the J-groove weld that were first identified due to recent technological advances in UT NDE.

CORRECTIVE ACTIONS

Immediate:

An assessment team was assembled to investigate the event including apparent cause(s), necessary corrective actions, and past/future unit operational impacts.

Subsequent:

1. Seven CRDM nozzles were repaired (5 nozzles had leaked and 2 had not leaked but NDE revealed suspect indications).
2. The remaining CRDMs that were [historically] neither repaired nor volumetrically inspected were UT inspected prior to unit restart.
3. An operability assessment was performed which concluded that CRDM nozzle cracks that could initiate and propagate during the upcoming power production cycle did not pose a significant safety issue.

Planned:

The PWSCC of Alloy 600 and Alloy 182 weld materials does not easily lend itself to identifying specific corrective actions to prevent recurrence. In the short term and as committed in Duke's response to NRC Bulletin 2001-01, CRDM inspections will be performed during future refueling outages. This management action plan will be in-effect until the RV heads are replaced on all three units. The current long-term solution for the elimination of the CRDM nozzle PWSCC issue is to replace the RV Heads presently scheduled to begin in 2003.

These short and long-term corrective action commitments have previously been furnished to the NRC and there are no new commitments being made in this report. These as well as other pertinent corrective actions are addressed and being managed via the Oconee Corrective Action Program.

SAFETY ANALYSIS

Actual Safety Consequences

There were no actual safety consequences as a result of this event. The leakage of primary reactor coolant through the CRDM nozzles was so minimal that it was detectable only by the extremely

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small accumulation of boric acid crystals observed on the RV head. The total leakage from the CRDM nozzles did not exceed Technical Specification limits for unidentified RCS inventory loss. At no time during cycle operation did the reactor building or area radiation alarms actuate as a result of this event. The small amounts of boric acid crystal deposits observed around the CRDM Nozzles had caused no detectable corrosion to the vessel head.

Potential Safety Consequences

Worst case scenario for this event was a rod ejection accident. However, for this accident to occur, extensive circumferential cracking above the CRDM nozzle weld would be necessary. As evidenced from NDT results from the 7 repaired and 43 inspected nozzles, only one nozzle (No. 2) exhibited circumferential cracking, but this crack had not progressed to a point where it would pose a safety concern.

Framatome-ANP previously performed an analysis assuming an above the weld circumferential flaws was through wall and extended 180° around. This analysis showed there was sufficient margin (with a safety factor of 3) to preclude gross net-section failure. The fact that visual inspection of the top of the RV head identified a nozzle containing a circumferential flaw supports the latest revision of Framatome-ANP's Safety Evaluation, which asserts that nozzles will be identified by leakage before circumferential flaws become a safety issue. The basis for this conclusion is the fact that an axial through-wall or through-weld flaw is required before the circumferential flaw can initiate and begin to grow.

Inspection of the leaking nozzles revealed that the PWSCC cracks responsible for the leaks were predominately axial in orientation. The cracking into the housing material was consistent with the results of elastic-plastic finite-element stress analysis of the CRDM housings that include modeling of both welding residual and operating stresses. Previous evaluation results from ONS 1, 2 and 3 CRDM nozzle and ONS 1 thermocouple nozzle leak events demonstrated that leak rates from cracks within the weld/housing regions of nozzles are low and that axial cracks extending beyond the weld and housing regions will leak (be detected by visual examination and leak-before-break) before there is a risk of failure. Leakage from cracked weld/housing material is predicted to result in boric acid corrosion rates sufficiently low that the leakage could continue for a period of time without affecting the structural integrity of the RPV head.

The degraded condition of RCS pressure boundary did not represent a challenge to the nuclear safety of the plant or jeopardize the health and safety of the public. As predicted by stress analysis and the fact that PWSCC does not occur or propagate into carbon steel material, the cracks did not extend into the reactor vessel head's low alloy steel but rather grew in the Alloy 600 or Alloy 182 material until they resulted in observable leaks that were detected during a planned refueling surveillance walkdown.

For this event, the majority of the nozzle cracks were located in the base metal and were axially oriented. Prior to restart of the unit, all nozzles that showed evidence of leakage were repaired and all non-repaired nozzles were inspected and no above the weld circumferential cracks were found. Based on this information, there were reasonable assurances that there were no nozzle circumferential cracks at unit startup. For the upcoming cycle, if a CRDM axial crack were to

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develop into a circumferential crack, there are also reasonable assurances that these cracks would not grow at a rate that could pose a safety concern prior to replacing the RV head at the next scheduled refueling outage.

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.

SIMILAR EVENTS

Within the last year, LER 269/2001-006-01 reported RCS pressure boundary leakage due to PWSCC failure of several thermocouple and one CRDM (No. 21) RV head penetrations. In addition, LERs 287/2001-001-00 and 270/2001-002 reported similar CRDM nozzle leakage events at ONS Units 3 and 2 respectively. Prior to these reports, there were no other LERs over the last three years that reported past PWSCC of Alloy 600 components or leaks that involved RV head penetrations. PWSCC is not new either to the domestic or worldwide nuclear industry. However, findings from ONS CRDM examinations have revealed OD initiated flaws in addition to the ID initiated flaws reported by most of the industry.

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