Duke Power®

RONALD A JONES
Vice President
Oconee Nuclear Site

Duke Power
ONO1VP / 7800 Rochester Hwy.
Seneca, SC 29672

864 885 3158
864 885 3564 fax

August 18, 2005

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

Subject: Oconee Nuclear Station, Unit 3
Docket Nos. 50-287
Licensee Event Report 287/2001-001, Revision 1
Problem Investigation Process No.: O-01-00587

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 287/2001-001, Revision 1, concerning the discovery of Reactor Pressure Vessel Head Leakage Due to Stress Corrosion Cracks Found in Several Control Rod Drive Nozzle Penetrations. Revision 1 incorporates a clarification to the original report wording in regards to the level of corrosion observed on the Reactor Pressure Vessel Head and updates status of corrective actions taken including the replacement of the head.

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

Attachment

IE22

cc: Mr. William D. Travers
Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S. W., Suite 23T85
Atlanta, GA 30303

Mr. L. N. Olshan
Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

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

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

INPO (via E-mail)

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-2004)

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
Estimated burden per response to comply with this mandatory 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 and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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 an 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	2. DOCKET NUMBER	3. PAGE
Oconee Nuclear Station, Unit 3	05000- 0287	1 OF 12

4. TITLE
Reactor Pressure Vessel Head Leakage Due to Stress Corrosion Cracks Found in Nine Control Rod Drive Nozzle Penetrations

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									None	05000
02	18	2001	2001	- 001	01	08	18	2005	None	05000

9. OPERATING MODE: 5

10. POWER LEVEL: 0%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)		50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
B.G. Davenport, Regulatory Compliance Manager	(864) 885-3044

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

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

At 2100 hours on February 18, 2001, a visual inspection of the top surface of the Oconee Nuclear Station Unit 3 (ONS Unit 3) Reactor Pressure Vessel (RPV) head found evidence of small accumulations of boric acid deposited at the base of several control rod drive mechanisms (CRDMs). This RPV head inspection was performed as part of a normal surveillance during a planned maintenance outage.

The boric acid deposits were identified around nine (Nos. 3, 7, 11, 23, 28, 34, 50, 56, and 63) of the sixty-nine total CRDM nozzles. The amount of boric acid around each of the CRDM nozzles was estimated to be no more than a few cubic inches but ultimately signified that reactor coolant system pressure boundary leakage had occurred. After confirming that the Reactor Coolant System (RCS) pressure boundary had been compromised, a February 18, 2001, 8-hour notification was made to the Staff in accordance with 10CFR50.72(b)(3)(ii)(B) reporting requirements. Subsequent non-destructive testing was performed on a total of eighteen CRDMs in order to effectively evaluate, characterize the leak mechanism, and determine extent of the condition.

Visual inspection of the reactor vessel head surface revealed a small area of head surface corrosion adjacent to nozzle 34. No other significant areas of corrosion were observed. Subsequent structural analysis of the area of corrosion concluded that slight degradation to the RPV head did not adversely affect the structural integrity of the component.

The apparent root cause of the nine CRDM Nozzle leaks is primary water stress corrosion cracking (PWSCC). The nine leaking CRDMs have been repaired. This event is considered to have minimal safety significance with respect to the health and safety of the public.

EVALUATION:

BACKGROUND

There are 69 Control Rod Drive Mechanism (CRDM) [EIIS:AA] nozzles [EIIS:NZL] that penetrate the reactor pressure vessel (RPV) [EIIS:RCT] head (see Figure 1). The CRDM nozzles are approximately 5-feet long and are welded to the RPV head at various radial locations from the centerline of the RPV 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 RPV head (see Figure 2).

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 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 RPV head bore and the OD of each nozzle. The nozzles were shrink 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 (see Figure 2). 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 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 RPV head was accomplished by machining and buttering the J-groove with 182-weld metal. The RPV head was subsequently stress relieved.

EVENT DESCRIPTION

At 2100 hours on February 18, 2001, a visual inspection of the top surface of the Oconee Nuclear Station Unit 3 (ONS Unit 3) Reactor Pressure Vessel (RPV) head found evidence of small accumulations of boric acid deposited at the base of several Control Rod Drive Mechanisms (CRDMs). This RPV head inspection was performed as part of a normal surveillance during a planned maintenance outage.

Boric acid deposits were identified around six of the sixty-nine total CRDM nozzles (Nos. 11, 23, 28, 34, 50, and 56). After washing down the RPV head, a February 25, 2001, follow-up inspection confirmed the presence of boric acid deposits on these six nozzles and revealed "suspicious" boron deposits on CRDM nozzles Nos. 3, 7 and 63 (see Figure 1). The amount of boric acid around each of the CRDM nozzles was estimated to be no more than a few cubic inches. After confirming that the Reactor Coolant System (RCS) [EIIS:AB] pressure boundary had been compromised, a

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit 3	05000-287	2001	001	01	3 OF 12

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

February 18, 2001, 8-hour notification was made to the Staff in accordance with 10CFR50.72(b)(3)(ii)(B) reporting requirements.

Subsequent surface dye-penetrant test (PT) inspections of the nine nozzles' weld area and Outside Diameter (OD) identified several deep axial cracks that had initiated near the toe of the fillet weld had propagated radially into the nozzle material as well as axially along the OD surface. Some of these cracks had reached the bottom end of the CRDM nozzle housing. These potentially deep cracks were the most likely leakage pathway that lead to the visible accumulations of boric acid crystals found on the Unit 3 RPV head. Eddy Current (EC) examination of the nine leaking CRDM nozzles revealed cracks on several of the nozzles.

Ultrasonic Test (UT) examinations were used to size the EC indications and determine the through-wall extent of other indications that EC could not resolve. The UT results confirmed the existence of deep (some through wall) cracks in all nine leaking CRDM nozzles. The PT, EC and UT results were combined to determine the extent of the existing cracking and develop nozzle specific repair plans for each of the leaking CRDM nozzles. Results from these NDE inspections are given in Attachment 1.

In addition to the original nine leaking CRDMs, nine additional nozzles (Nos. 4, 8, 10, 14, 19, 22, 47, 64 and 65) from the same heat[1] of material as the initial nine CRDMs were EC and UT inspected for "extent of condition" purposes (see Figure 1). Results of this inspection are also given in Attachment 1.

Visual inspection of the reactor vessel head surface revealed a small area of head surface corrosion adjacent to nozzle 34 (See Figure 3). No other significant areas of corrosion were observed. Subsequent structural analysis of the area of corrosion concluded that slight degradation to the RPV head did not adversely affect the structural integrity of the component.

Although the leakage of primary coolant through the CRDM nozzles was so minimal that it was detectable only by the observed accumulation of boric acid crystals on the RPV head, 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 RCS leakage from the CRDM nozzle penetrations resulted in 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).

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.

[1] ONS Unit 3 has two CRDM heats of material distributed between the 69 CRDM Nozzles

CAUSAL FACTORS

The apparent root cause of the nine alloy 600 CRDM nozzle leaks is Primary Water Stress Corrosion Cracking (PWSCC).

General cause of event discussion:

Alloy 600 is used extensively in nozzle applications in the reactor vessel and Pressurizer [EIIS:PZR]. It is also used for hot and cold leg piping as well as steam generator tubing in Combustion Engineering and Babcock and Wilcox fabricated plants. It is recognized these small-bore nozzles have suffered 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 is generally thought to initiate at the nozzle inside surface adjacent to the partial penetration J-groove welds. This area has been shown to have high residual stresses resulting from the weld process and, in some cases, from surface distress due to machining, grinding or reaming operations. In thin wall product forms, this area could also have an altered microstructure from welding (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 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 are the ONS Unit 3 CRDM nozzles.

Specific discussion regarding cause of event reported in this LER:

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

1. Metallographic examination of CRDM nozzle samples found PWSC cracks,
2. Correlation of crack location and orientation with Finite Element Analyses (FEA) indicating high tensile stress, and
3. The recent history of cracking found in alloy 600 weld metal attributed to PWSCC at the ONS, i.e., this is the second reportable instance of PWSCC at ONS resulting in leakage. ONS Unit 1 was the first occurrence (Ref.: LER 269/2000-006-01).

Additionally, a Duke Engineering and Services (DE&S) metallurgical evaluation report on ONS Unit 3 CRDM Housing cracks concluded that,

1. The cracking resulted from a stress driven intergranular corrosion mechanism,
2. There was no indication of aggressive chemical species on the crack face,
3. The primary driving force in the region of cracking appears to be due to residual surface stress from cold deformation after the parts were annealed, and
4. The apparent corrodent was the primary coolant in the reactor coolant system.

The leaking cracks' path, as characterized by the UT and PT examinations, fell within the nozzle wall region where FEA (including the effects of welding residual stresses and operating conditions) predict high hoop stresses. The crack geometry was consistent with the analysis that shows the hoop stress (that drives cracks in the axial orientation) was higher than the axial stress (that drives cracks circumferentially) at high stress locations. Crack growth into the nozzle wall was also consistent with analysis predictions that high hoop stresses extended through the weld material and into the nozzle wall. The deep and mostly axial oriented crack was consistent with FEA results, and with a root cause determination of PWSCC.

CORRECTIVE ACTIONS

Immediate:

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

Subsequent:

1. A combination of eddy current, ultrasonic, and dye penetrant inspections were performed on each of the nine leaking CRDMs (Nos. 3, 7, 11, 23, 28, 34, 50, 56 and 63).
2. Nine additional CRDM nozzles (Nos. 4, 8, 10, 14, 19, 22, 47, 64 and 65), of the same heat of material as the initial nine CRDM nozzles, were also EC and UT inspected. These nine additional CRDM nozzles were inspected to support "extent of condition" evaluations.
3. CRDM Nozzle material from the lower portions of seven CRDMs (Nos. 3, 7, 11, 23, 28, 50 and 56) and a small sample that included a circumferential crack above the J-groove weld region from CRDM Nozzle 56, were removed and sent to the DE&S Metallurgical Lab in Huntersville, NC for analysis and evaluation.
4. The nine leaking CRDMs were repaired (as described below).

   The general repair process was to remove all crack indications and weld repair the individual excavation(s) for each CRDM nozzle. The cracks were first ground out of the nozzle material (initially by manual grinding, later by air arc gouging followed by shallow

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit 3	05000-287	2001	001	01	6 OF 12

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

surface grinding), sometimes exposing a small area of low alloy steel base metal for some nozzles. The final surface was PT examined prior to preheating for the weld repair process.

During the crack indication removal process and subsequent nondestructive examination, some cracks were "chased" from their surface location (identified by PT) into the weld and/or nozzle material. These original indications were followed to confirm the leak path through the pressure boundary had been eliminated. While removing these indications, other pre-existing linear indications in the weld and nozzle material were also identified and removed.

Following weld repair of the individual excavations, a protective weld overlay of alloy 52/152 material (alloy 690 type) was deposited over the CRDM repairs where original weld material (82/182 alloy 600) remained after the repair process. Five of the repaired nozzles (Nos. 3, 7, 11, 50, and 56) had this protective overlay. Nozzles 23, 28, 34, and 63 did not require a weld overlay since the entire weld was replaced with 52/152 material. This nonstructural weld material acts as a protective layer to the existing material, improving its resistance to PWSCC attack.

Planned:

Revision 1 Update: Replacement of the RPV heads and subsequent commitments made in association with NRC Bulletins 2001-01, 2002-01, and 2002-02 have addressed both the short and long-term corrective actions described below:

1. Although repairs have been completed for the nine leaking CRDM nozzles, the potential for future leakage events of alloy 600 CRDM nozzle components (including the 182 weld material) on the existing RPV head, due to PWSCC, remains a concern for all three Oconee Units. An aggressive management plan that focuses on continued RPV head nozzle inspections and repairs was determined to be the best approach to address PWSCC in the short-term.

2. In the long-term, the RPV heads will be replaced at all three Oconee Units to prevent recurrence of this event.

These short and long-term actions as well as other planned corrective actions are being addressed via the Oconee Corrective Action Program. There are no NRC Commitment items contained in this LER.

SAFETY ANALYSIS

It was determined that the orientation of the cracks, as they traversed through the nozzle material, was primarily axial and the branching observed was typical of a PWSCC. The predominant direction of the crack was along the axis of the nozzle with crack penetration into the wall of the

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit 3	05000-287	2001	001	01	7 OF 12

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

nozzle. The localized circumferential cracks found in CRDM nozzles 11, 23, 34, 50, and 56 were generally associated with axial cracks.

As concluded in a Framatome Technologies Incorporated (FTI) safety assessment report, any circumferential flaw above the weld on the outside surface of the nozzle should not be considered a significant safety concern. Specifically, it was determined that a short, isolated flaw would take more than ten (10) years to grow through-wall, while a long circumferential (where multiple flaws have joined) could grow from the outside surface to the inside surface in about three and one-half (3.5) years. In neither case would the structural integrity of the nozzle be compromised to the point that the nozzle would fail by ejection.

Circumferential cracking was also observed on the outside surface of CRDM nozzles at ONS Unit 3 at the toe of the fillet weld that forms part of the structural attachment to the RPV head. Since these cracks are located at or below the weld, and not in the reactor coolant pressure boundary, they are not considered to be a significant safety concern from the standpoint of gross structural failure or release of radioactive water. Due to the proximity of associated through-wall cracking below the weld, there was a concern that a through-wall circumferential crack could link up with two or more through-wall axial cracks and form a loose part that could potentially enter a control rod guide tube and prevent a [single] control rod assembly from being fully inserted. However, this type of anomaly has been evaluated and is bounded by the ONS LOCA and non-LOCA accident analyses which assume that the highest worth control rod assembly does not insert into the core. Multiple control rod assembly insertion failures, due to loose part intrusion, were evaluated and determined not to be credible scenario.

Additionally, based on experience at ONS Unit 3, circumferential and axial cracking below the weld is accompanied by through-wall cracking at and above the weld, as evidenced by the boric acid crystal deposits on the RPV head. It was subsequently concluded from these results and observations, that detectable leakage would precede the development of a loose part.

In summary, the degraded condition of CRDM Nozzles did not represent a challenge to the nuclear safety of the plant or jeopardize the health and safety of the public. The majority of the cracks were located in the nozzle base metal and were axially orientated. As predicted by stress analysis, and the fact that PWSCC does not occur in carbon steel material, the cracks did not extend into the reactor vessel head's low alloy steel but rather grew until it resulted in an observable leak that was detected during a normal shutdown surveillance walkdown. The leakage of primary coolant through the CRDM nozzles was so minimal that it was detectable only by the observed accumulation of boric acid crystals on the RPV head. The total leakage from the CRDM nozzles did not exceed Technical Specification limits for unidentified RCS inventory loss. No Reactor Building or area radiation alarms sounded.

Visual inspection of the reactor vessel head surface revealed a small area of head surface corrosion adjacent to nozzle 34. No other significant areas of corrosion were observed. The corrosion in question was attributed to the effects of primary system leakage. Dimensions of the area were taken and used to provide input for a structural analysis of the effect of the corrosion on the integrity of the head. Results of the analysis[2] revealed that the RPV head was not adversely affected from the corrosion and would continue to satisfy the stress and fatigue requirements (as described in ASME Code, Section III, 1989 edition without addendum), for an additional 14 years (~9 cycles) of operation.

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

Other than the recent ONS Unit 1 LER (269/200-006-01) that reported RCS pressure boundary leakage due to PWSCC failure of several of the RPV head thermocouple and CRDM #21 penetrations, there were no other LERs over the last two years that reported past PWSCC of alloy 600 components or leaks involving RPV head penetrations.

This type of cracking phenomena is not new either to the domestic or worldwide nuclear industry. From the recent discovery at ONS Unit 3, as well as the previous discovery of PWSCC at ONS Unit 1, the Oconee Units will remain susceptible to future PWSC cracking of alloy 600 components. Until a planned corrective action to replace all of the Oconee RPV heads is implemented, this type of event is expected to recur.

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

[2] OC-3 RV Head Erosion/Corrosion at CRDM Nozzle #34, Framatome ANP, Inc., #32-5011933-00, dated April 17, 2001

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

Attachment 1

CRDM Nozzle NDE Results

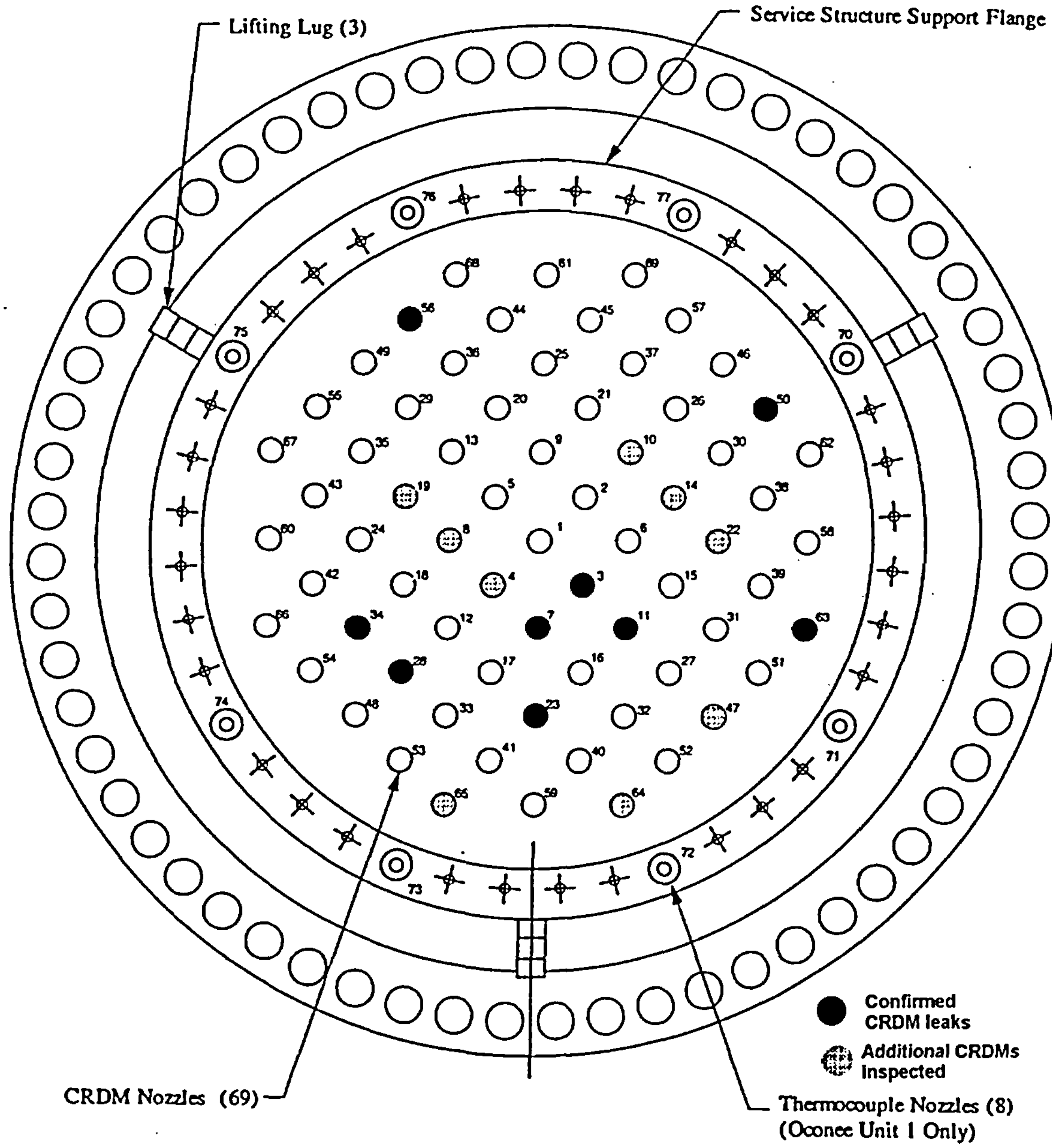
The first seven items are for initial nine leaking CRDM locations and the remaining two items are based on the result of the inspection of the nine "extent of condition" CRDM nozzles.

1. Of the total of 47 original indications in nine leaking nozzles, 19 were OD initiated axial flaws (19/47 = 40%) that were not through wall.
2. There were 16 OD initiated indications (34%) that were axial through wall flaws.
3. There were nine circumferential flaws (19%), one ID initiated, eight OD initiated. Only two of the OD circumferential flaws (nozzles 50 and 56) were above the J-goove weld.
4. Every circumferential OD flaw (eight flaws in four nozzles) had at least one axial through wall flaw. Two of the OD circumferential indications were through wall or near-through wall (nozzles 50 and 56) flaws.
5. Three nozzles (3, 28 and 63) had five or more axial flaws with no circumferential flaws.
6. There were also three ID initiated axial flaws (6%) that were not through wall.
7. All OD initiated circumferential indications are significantly deeper than the ID initiated flaws. OD circumferential flaws average 70% through wall for eight indications and the single ID flaw was 13% through wall.
8. All of the nine CRDMs inspected for extent of condition by EC had only Cluster Flaws above and/or below the J-groove weld. These flaws are similar in size extent and depth to those observed on numerous ONS Unit 1 and ONS Unit 2 CRDM ID surfaces. The maximum depth found was 1.75 mm for Nozzle 10.
9. None of the nine CRDMs inspected for extent of condition using UT inspection techniques had recordable OD Flaws. One (CRDM 4) had four shallow axial flaws (max. 1.37 mm deep) above the weld at the downhill (6:00) position.

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

Figure 1

Reactor Vessel Closure Head Map

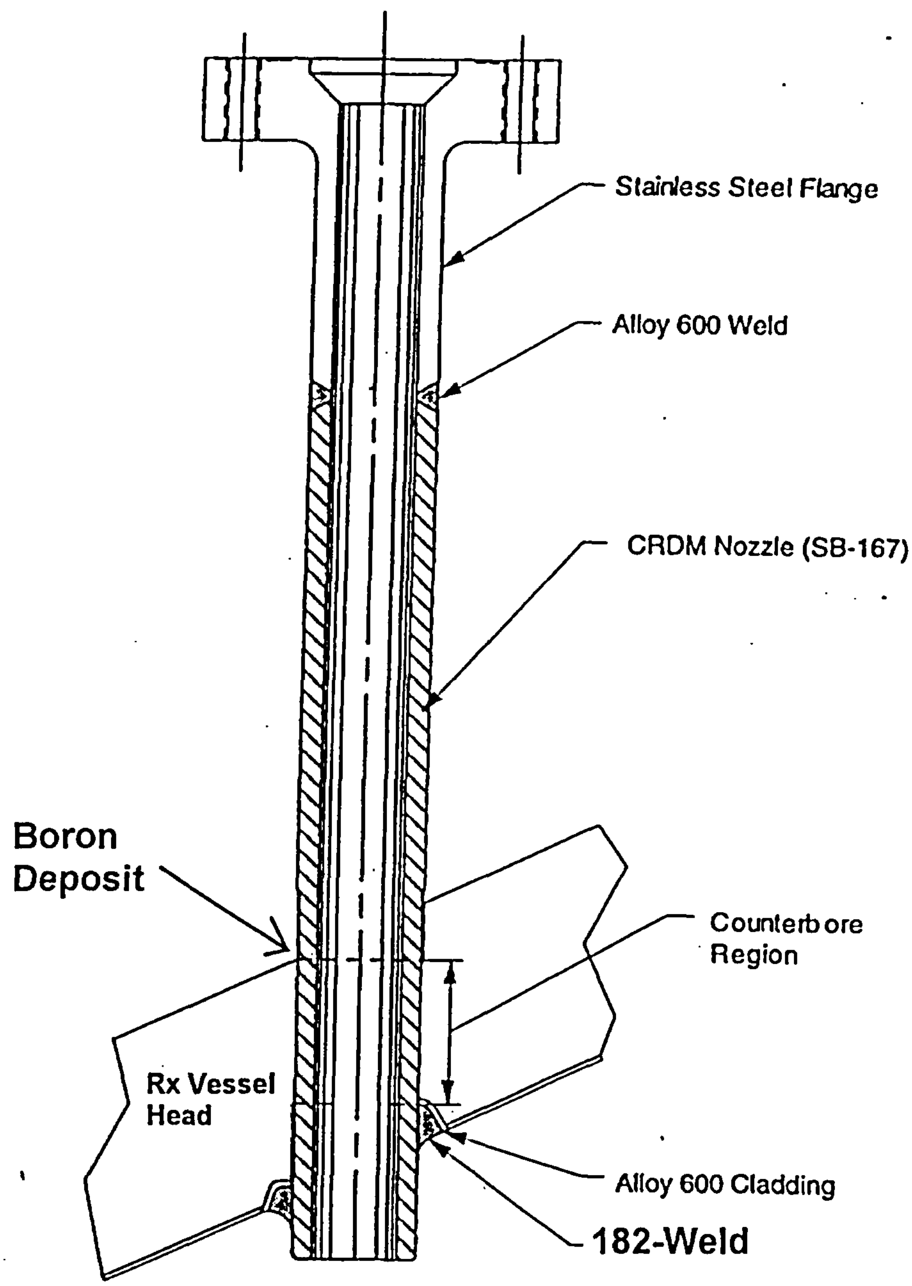


FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit 3	05000-287	2001	001	01	11 OF 12

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

Figure 2

Oconee CRDM Nozzle Penetration (typ.)

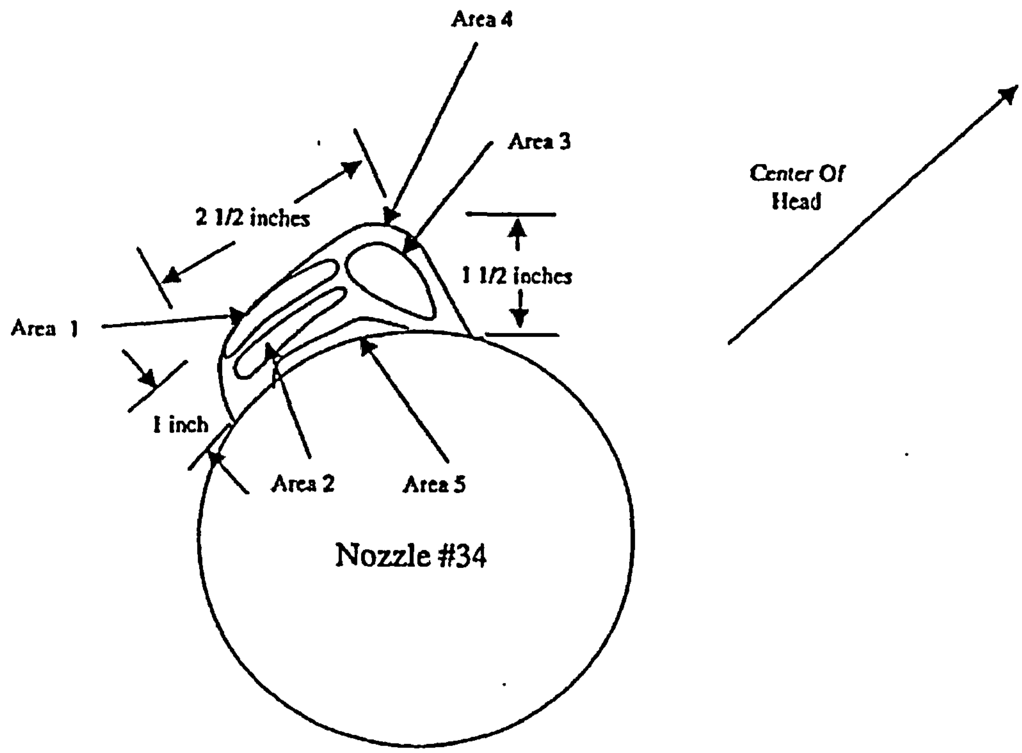


FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oconee Nuclear Station, Unit 3	05000-287	2001	001	01	12 OF 12

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

<u>Figure 3</u>

Area of Corrosion near Nozzle #34



Area	Dimensions in inches
1	1/4 wide x 1-1/2 long x 1/4 deep
2	3/8 wide x 1-1/2 long x 1/16 deep (island)
3	1/4 wide x 3/4 long x 3/16 deep
4	1/8 or less (general area)
5	1/8 wide x 1-1/2 long x 1/2 deep

Dimensions are bounding