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

Subject: Oconee Nuclear Station, Unit 1
Docket Nos. 50-269
Licensee Event Report 269/2002-003, Revision 0
Problem Investigation Process Report No. O-02-01402

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 269/2002-003, Revision 0, addressing the discovery of minor reactor pressure vessel head leakage due to Primary Water Stress Corrosion Cracking of an Alloy 600 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,

Bruce Hamilton /by

W. R. McCollum, Jr.

Attachment

IE22

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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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Minor Reactor Pressure Vessel Head Leakage Due to Primary Water Stress Corrosion Cracking of An Alloy 600 Control Rod Drive Nozzle

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
4	1	02	2002 -	003 -	00	05	22	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)**

Unit 1 entered its scheduled end-of-cycle 20 refueling outage on March 23, 2002. With the reactor vessel head (RVH) still installed on the vessel, a qualified visual inspection of the bare RVH was performed on March 24, 2002. From this visual inspection, two CRDM nozzle penetrations (Nos. 1 and 7) were identified with boron deposits on the CRDM tube. In accordance with Duke's response to NRC Bulletin 2001-01, both nozzles were categorized as masked and required further inspections. To avoid impacting the outage schedule, nozzles 5, 8 and 9 were subsequently added to the inspection list due to uncertainty regarding the ability to perform a full 360 degree visual inspection with the head installed.

Results of an ultrasonic test inspection on the five nozzles revealed axial indications on the outside diameter of nozzles 1, 7 and 8. These indications, which centered on the lower nozzle to J-groove interface, did not extend through the tube wall or reveal a leak path by extending above the J-groove weld. Follow-up dye-penetrant exams confirmed that nozzle 7 had leaked and that nozzle 1 repairs were not required. From past nozzle repair data, a conservative decision to repair nozzle 8 was made although a leakage path was not identified. On April 1, 2002, after confirming that the Reactor Coolant System pressure boundary had been degraded during power operations, an 8-hour notification was made at 2123 hours in accordance with 10 CFR 50.72(b)(3)(ii)(A) reporting requirements. The apparent root cause of the nozzle leaks is primary water stress corrosion cracking. The two nozzles were repaired prior to unit restart. This event had minimal safety significance with respect to the health and safety of the public.

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

EVALUATION:

BACKGROUND

There are 69 Control Rod Drive Mechanism (CRDM) [EIIS:AA] nozzles [EIIS:NZL] that penetrate the Reactor Vessel Head (RVH) [EIIS: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 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 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

Unit 1 entered its scheduled end-of-cycle 20 (1EOC20) refueling outage on March 23, 2002. As part of the outage and with the reactor vessel head (RVH) still installed on the vessel, a qualified visual inspection of the bare RVH was performed on March 24, 2002. The purpose of this inspection was to look for any preliminary indications of primary leakage from the unit's 69 Control Rod Drive Mechanism (CRDM) nozzle penetrations. To aid with the visual inspection, Oconee had previously installed nine access ports in the RVH service support skirt to allow the inspectors an unobstructed view of the RVH nozzles. In addition to the 69 CRDM penetrations, the Unit 1 RVH also contains 8 thermocouple (T/C) nozzle penetrations (Figure 1). These T/C nozzles are no longer used and they were permanently plugged during the 1EOC19 nozzle repair campaign (Ref. LER 269/2000-006-01).

From this visual inspection, no leakage was noted from the T/C nozzles. However, two CRDM nozzle penetrations (Nos. 1 and 7, see Figure 1) were identified with boron deposits on the CRDM tube. At the time, it could not readily be determined if pressure boundary leakage had occurred or if the boron deposits were attributable to other sources such as flange joint leakage. In accordance with Duke's response to NRC Bulletin 2001-01, both nozzles were categorized as masked and required further inspections.

Following this initial inspection, a question was raised regarding the ability to effectively see around certain nozzles located high on the RVH. In particular, there were several nozzles whose full periphery would have been partially obstructed by the mirror insulation support beams that traverse the head. With the head installed, this configuration could have prevented a 360 degree perspective of these nozzles. To prevent outage impact while resolving these concerns, three additional nozzles (Nos. 5, 8 and 9) were subsequently identified as requiring additional nondestructive examination (NDE) inspections (Figure 1).

After the RVH was removed and placed on the stand, ultrasonic test (UT) inspections of the five nozzles from inside the nozzle diameters were performed. The scans looked for both axial and circumferential flaws in the nozzle wall from the bottom of the nozzle penetration to the top of the RVH surface.

Results of the UT inspection revealed partial through-wall outside diameter (OD) cracks for nozzles 1, 7 and 8. The nozzle 7 and 8 findings appeared to be very similar to those seen from previous nozzle inspections on the other Oconee units. There were five flaws and a potential leak path identified for nozzle 7 and one axial flaw but no identified leakage path for nozzle 8. Nozzle 1 showed three minor indications in a region of rough weld contour while nozzles 5 and 9 did not show any UT indications. Since the UT scans for nozzles 5 and 9 were clear, no further inspection or repair was required.

A liquid dye-penetrant test (PT) of the bore of nozzle 7 (after removal of the nozzle) revealed two axial indications high in the original weld. This indicated a potential leak path through the weld. The video inspection for CRDM flange leaks found no leaking flanges and a review of a video of the top of the head made during the previous outage (1EOC19) did not show evidence of the new deposit around nozzle 7. Consequently, there was strong evidence that nozzle 7 was leaking through the weld and that the leak initiated during the past operating cycle.

A PT exam was also performed on nozzle 1 to confirm the condition of the nozzle OD surface. The PT of the fillet weld cap, partial penetration J-groove weld, and the nozzle OD surface, showed no recordable or rejectable PT Indications. Consequently, nozzle 1 did not require repair.

From Oconee's previous nozzle repair data, a conservative decision was made to repair nozzle 8. This decision was based on the UT axial indications on the OD nozzle surface that, when compared to previous ONS inspections, would tend to indicate that this type of active PWSCC flaw would eventually result in nozzle leakage through the weld material.

On April 1, 2002, after confirming that the Reactor Coolant System [EIIS: RCS] pressure boundary had been degraded during power operations, an 8-hour notification was made at 2123 hours in accordance with 10 CFR 50.72(b)(3)(ii)(A) reporting requirements.

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

Due to improvements in the UT inspection techniques in identifying leakage paths along the nozzle OD volume and the increased radiation dose to inspection personnel, eddy current inspections were not performed on the unit 1 nozzles. After the head was cleaned and placed on the stand, a visual was performed looking for wastage using a video camera and then a boroscope, where required, to inspect 360 degrees around each nozzle. This inspection did not identify any RVH wastage.

Repair Process

Nozzles 7 and 8 were repaired utilizing a similar process used for previous Oconee Units 2 and 3 nozzle repair campaigns. From under the RVH, the protruding portions of the nozzles, and a length of about 5 inches into the RVH bore, were removed by machining. A new pressure boundary weld was installed within the bore, inspected, and surfaced conditioned with a water jet peening process.

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 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 1 was in cold shutdown (Mode 5) at 0 percent power and Units 2 and 3 were in Mode 1 operating at approximately 100 percent power.

CAUSAL FACTORS

The apparent root cause of the leaking Alloy 600 CRDM nozzle (No. 7) was 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.

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

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:

The observations, inspection results and stress analyses performed for the nozzles clearly support the conclusion that PWSCC is the apparent failure mechanism candidate. All the cracks initiated at the nozzle outside diameter surface, along the intersection between the J-groove fillet weld and the nozzle on the underside of the reactor vessel head. The following is a review of the three principle conditions required for PWSCC, (1) a susceptible material, (2) high tensile stresses and (3) an aggressive environment.

All of the CRDM nozzles exhibit, to some degree, all three of the characteristics necessary for PWSCC cracking. The material characteristic is confirmed by the moderate sensitization potential and the high stress characteristic is confirmed by the Finite Element Analysis (FEA) and the tensile sample tests. The environmental characteristic is confirmed by the primary water environment, supplemented by the Unit 1 past history of periods of resin (sulfate) intrusion. Although none of these characteristics is extreme, the combination of moderate effects would lead to a conclusion of moderate PWSCC potential. This comparison results in solid evidence supporting PWSCC as the apparent cause of the ONS 1 CRDM nozzle cracks.

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:

CRDM Nozzles 7 and 8 were repaired.

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 nozzle inspections will be performed during future refueling outages. This management action plan will be in-effect until the RVHs are replaced on all three units. The current long-term solution for the elimination of the CRDM nozzle PWSCC issue is to replace the RVHs beginning with Unit 3 in 2003.

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

ONS has other numerous Alloy 600 components in service exposed to the primary system environment. These components, all susceptible to varying degrees to similar cracking found on the RVH penetrations, will be evaluated as part of the ongoing ONS Alloy 600 Program.

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 small accumulation of boric acid crystals observed on the RVH. 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 nozzles had caused no visual corrosion damage to the vessel head.

Potential Safety Consequences

The investigation into the ONS 1 RVH leakage revealed information that continued to support the conclusions documented in previous Oconee Units 1, 2, and 3 reported events. Four supporting points from this most recent ONS 1 outage include:

1. All of the nozzle cracks were axial and appeared on the nozzle OD surface near the weld.
2. No circumferential cracking was found.
3. There were no through-wall flaws. Eight of 9 flaws identified were relatively small and shallow.
4. None of the nozzles exhibited a large quantity of boron indicative of past gross leakage.

Ultrasonic testing revealed a potential leak path within the nozzle 7 wall region where FEA, including the effects of welding residual stresses and operating conditions, predict high hoop stresses. The nozzle 7 and 8 axial crack geometries were 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 (although small) is also consistent with analysis predictions that high hoop stresses extend through the weld material and into the nozzle wall. The mostly axially oriented OD cracks are consistent with FEA results, and with a root cause determination of PWSCC.

Visual inspection of the top of the RVH identified no gross leakage and UT examination revealed no circumferential flaws. Even if circumferential flaws were found, the latest revision of Framatome-ANP's Safety Evaluation 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.

Consequences of Failure (Failure Scenario)

The leaking nozzle reported is not considered to be a safety concern since the PWSCC cracks involved were axial in orientation. The cracking into the housing material is reasonably consistent with the results of elastic-plastic FEA of the CRDM housings that include modeling of both welding residual and operating stresses. The probability of a nozzle being ejected is very low.

In addition, previous Units 1, 2 and 3 CRDM nozzle and ONS 1 T/C repair campaigns generally support the following several factors: (1) that leak rates from cracks within the weld/housing regions of nozzles are low, (2) that axial cracks extending beyond the weld and housing regions will leak and be detected by visual examination and leak-before-break, before there is a risk of nozzle failure, (3) the leakage from cracked weld/housing material 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, and (4) axial (particularly through wall) cracks are much more numerous than circumferential cracks.

The rod ejection accident is a possibility if this leakage is not detected and repaired. At Oconee, all RVHs are cleaned to ensure that boron corrosion is controlled and minimized. Large "portal" openings have been made in the service structure to improve the RV bare head inspections and to date, all identified or suspected leaking nozzles have been repaired. The potential for a rod ejection accident has not increased as a result of these axial cracks. Also, the rod ejection accident has been thoroughly evaluated in the ONS UFSAR and the ONS operators have been trained on the procedure and response to this accident.

The risk of the gross RVH corrosion, similar to that reported at Davis-Besse, is very improbable at the Oconee units. This is due to the following reasons:

1. A complete visual inspection of the Oconee RVH bare metal, below the mirror insulation and utilizing the nine access ports located in the service structure, assures that each nozzle is effectively viewed from several angles. There is reasonable assurance that any areas of head corrosion would be visible. The recent visual inspection did not show any visible signs of corrosion or wastage on the head surface.
2. Since the RVHs are cleaned at each outage, any new boron deposits will be distinguishable during future visual inspections.
3. There is reasonable assurance that the post machining NDE and the actual weld repair method (used for nozzles 7 and 8) would have identified any occurrence of head wastage due to corrosion or erosion. It is also reasonable to assume that this method would have also found any wastage below the surface of the head and between the nozzle's exterior wall and the head bore ID.
4. The PT performed on nozzle 7 would have aided in the identification of voided areas in the RVH material.
5. The automated repair welding requires essentially a zero gap between the head and the machined beveled nozzle. Any gap requires weld deposition modification, or build-up, prior to performing the repair welding.

A risk assessment was completed for the three Oconee units to evaluate the time period prior to reactor vessel head replacement. The risk assessment concluded that the increase in core damage frequency due to the probability of a circumferential crack reaching the critical flaw size and causing a loss of coolant accident is below the point of being risk significant for any of the Oconee units during each of these reactor years. In addition, the revised ONS Updated Final Safety Analysis Report Chapter 15 accident analysis, using actual ONS core data, concludes that there would be no failed fuel due to a postulated rod ejection accident from a potentially failed CRDM nozzle.

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

Over the last two years, similar LERs have been submitted for all three Oconee units. The last, LER 287/2001-003-00, reported five PWSCC related RVH penetration leaks that required repair. Prior to this, LER 269/2000-006-01 reported RCS pressure boundary leakage due to PWSCC failure of several thermocouple and one CRDM (No. 21) RV head penetrations. Licensee Event Reports 287/2001-001-00 and 270/2001-002-00 reported CRDM nozzle leakage events at ONS Units 3 and 2 respectively. Other than these recent LERS, there have been no other reportable events that involved PWSCC of Alloy 600 components or RVH penetration leaks.

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

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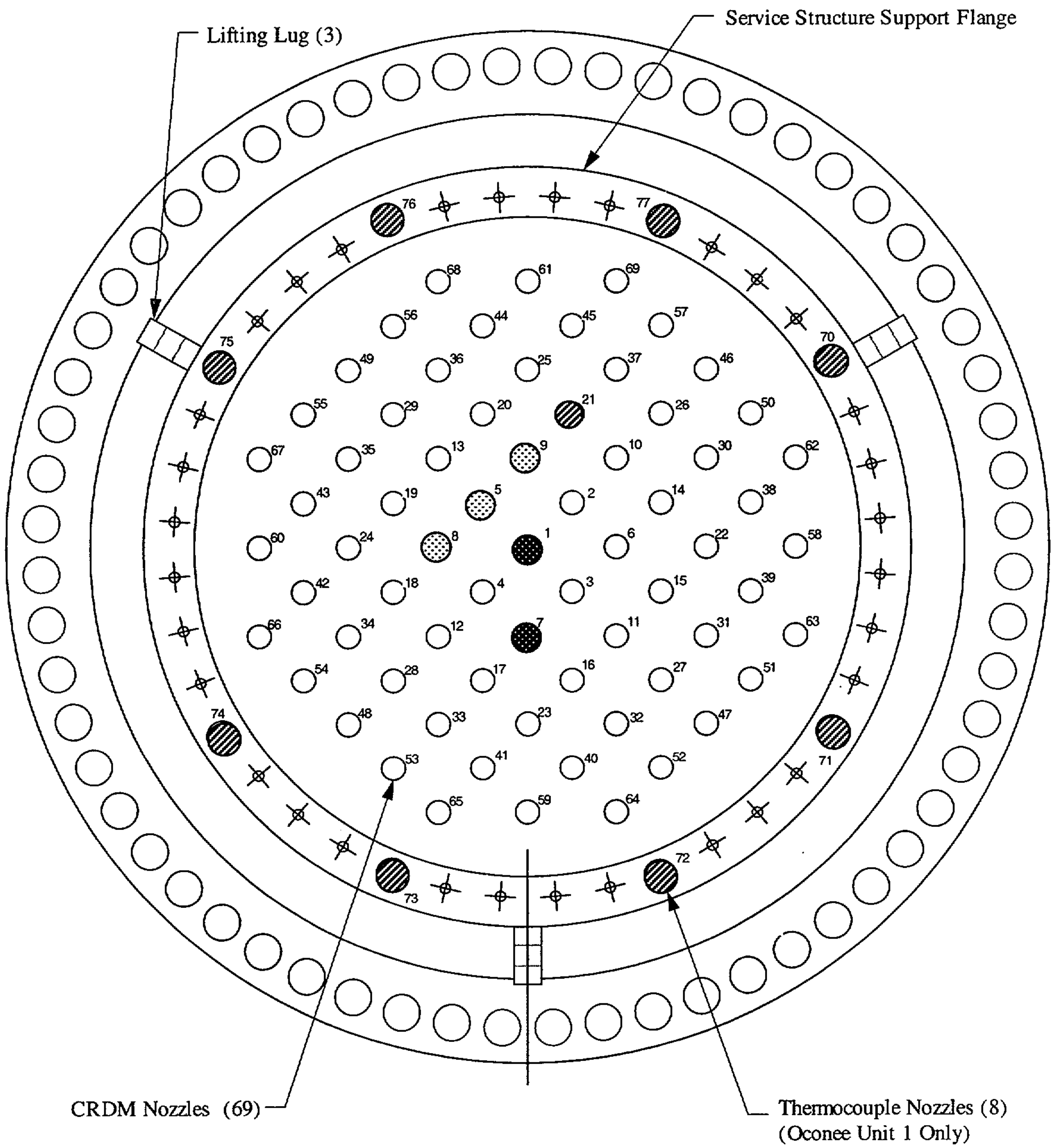
TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Figure 1 – Oconee Unit 1 RVH Map



- Nozzles 1 and 7 were initially masked; Nozzle 7 was found to be leaking and was repaired
- Nozzles 5, 8 and 9 identified as being obstructed from 360° view; Nozzle 8 was repaired
- From 1EOC19: Nozzle 21 (leaking/repaired) and 8 Thermocouple Nozzles (leaking/plugged)