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March 1, 2001

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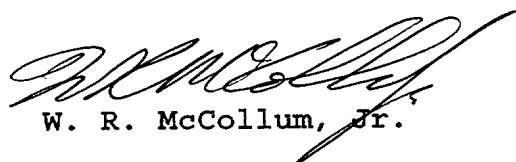
Subject: Oconee Nuclear Station, Unit 1
Docket No. 50-269
Licensee Event Report 269/2000-006, Revision 1
Problem Investigation Process No.: O-00-04134

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 269/2000-006, Revision 1, concerning the discovery of reactor coolant system pressure boundary leakage due to small cracks found in several small bore reactor vessel head penetrations. Revision 1 includes the final results of the root cause investigation into the event.

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

Very truly yours,



W. R. McCollum, Jr.

Attachment

IE 22

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Date: March 1, 2001

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cc: Mr. Luis A. Reyes
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INPO (via E-mail)

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If 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.

LICENSEE EVENT REPORT (LER)

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

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DOCKET NUMBER (2)

05000 - 269

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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
12	04	00	2000	006	01	3	01	01		05000
OPERATING MODE (9)			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
5			20.2201(b)		20.2203(a)(2)(v)		<input checked="" type="checkbox"/> 50.73(a)(2)(i)(B)		50.73(a)(2)(viii)	
POWER LEVEL (10)			20.2203(a)(1)		20.2203(a)(3)(i)		<input checked="" type="checkbox"/> 50.73(a)(2)(ii)		50.73(a)(2)(x)	
0%			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

L.E. Nicholson, Regulatory Compliance Manager

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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)

YES (If yes, complete EXPECTED SUBMISSION DATE).	<input checked="" type="checkbox"/> NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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 5 (of the 8) 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 around the suspect nozzles.

On December 4, 2000, an eddy current test was performed on the inside surface of the 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. Following completion of the root cause analyses, Primary Water Stress Corrosion Cracking was determined to be the primary failure mechanism of both the T/C nozzles and CRDM weld cracks. Prior to unit restart, the #21 CRDM weld was repaired and the 8 T/C nozzles removed and their resultant head penetrations permanently plugged. These actions were completed. This event is considered to have no significance with respect to the health and safety of the public.

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EVALUATION:

BACKGROUND

Thermocouple (T/C) 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) (See Figure 1). 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 (See Figure 2) 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 partial penetration 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.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)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 (See Figure 1). 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 182-weld metal. 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 permanently welded to the closure head using 182-weld metal (See Figure 3). 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."

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Boric acid deposits appeared to be located at the base of 5 (of the 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 T/C nozzles #1, #2, #3, #4 and #5. 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 (EC) test performed along the inside surface of the T/C nozzles revealed axial crack-like indications in all 8 of the T/C nozzles. These cracks were primarily axially oriented and located adjacent to (extending both above and below) the J-groove 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 EC 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. Although this event was discovered while in Mode 5, there is reasonable evidence, based on boron aging analysis, that the leaks started either during cycle 19 (July 1999) or cycle 18 (September 1998). 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).

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There was no operator intervention 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

Detailed analyses from the individual T/C and #21 CRDM root cause investigations performed concluded the root cause of the alloy 600 T/C nozzle leaks was primary water stress corrosion cracking (PWSCC). Similarly, the root cause of the alloy 600 CRDM Nozzle 21 leak was PWSCC of the Inconel 182-weld metal and the alloy 600 nozzle material. The bases for these conclusions are described below.

T/C Nozzles

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.

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On December 6, 2000, ultrasonic Test (UT) examinations from the inside surface of the T/C nozzles allowed the weld size to be determined and the axial crack-like indications (as determined by the prior EC examination) to be located relative to the welds. This work showed that most welds: (a) were significantly larger than specified on the design drawings, (b) lacked weld symmetry, (c) contained small areas of possible slag inclusions at the interface between the nozzles and welds, and (d) were undercut at the toes of the fillets where the weld joins the nozzle outside diameters.

Liquid penetrant (LP) examinations of the J-groove welds (after boring out the nozzles), showed that some cracks had penetrated through the nozzle walls and that the orientation of these cracks was predominantly axial at the plane where the cracks penetrated into the welds. Four T/C nozzles (#4, #6, #7 and #8), had crack depths less than 0.163 inches into the 182-weld metal. Two other T/C nozzles (#1 and #3), had crack depths less than 0.288 inches into the weld. The remaining two nozzles (#2 and #5), had some cracks penetrating completely through the J-groove weld, the Inconel buttering and up to the vessel head low alloy steel. After all crack indications had been removed, the total exposed low alloy material was about 1.75 square inches for T/C #2 and 2.5 square inches for T/C #5.

Metallurgical lab testing of the two removed nozzle pieces (T/Cs #7 and #8), confirmed the presence of an axial through-wall intergranular crack on the piece removed from above the weld on nozzle #8. An axial partial through-wall intergranular crack was found on the piece removed below the weld on nozzle #7.

Eddy current, Ultrasonic (UT) and LP examinations revealed that the predominately axial cracks found fall within the weld region where finite element stress analyses (including the effects of welding residual stresses and operating conditions) predict high hoop stresses. The axial cracks are consistent with the fact that the analysis shows the hoop stress (that drives cracks in the axial orientation) to be higher than the axial stress (that drives cracks circumferentially) at high stress locations. The fact that the axial cracks have grown through the nozzle wall and into the J-groove weld is also consistent with analysis predictions that the high hoop stresses extend through the nozzle wall and into the weld metal. In summary, the deep and predominantly axially oriented

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cracks are compatible with the finite element analysis results, and with the root cause determination of PWSCC.

CRDM #21

An EC inspection of CRDM Nozzle 21 on December 7, 2000 did not identify any significant indications that suggested a through-wall leak. Following wire brushing of the CRDM nozzle weld and an area of approximately 2 inches around the nozzle, a liquid penetrant inspection of the nozzle, fillet weld cap and partial penetration J-groove weld was performed on December 9, 2000. Results of this test revealed two non-recordable rounded indications, less than 0.05 inch in diameter, on the face of the fillet weld. The indications were located at approximately the 7 o'clock position on the circumference of the nozzle weld (note that the 12 o'clock position is the uphill orientation and 6 o'clock the downhill). No other indications were noted either on the weld or on the surrounding nozzle base metal.

Following light metal removal in the area of the two rounded indications, a second LP inspection that expanded the examination area around the nozzle to 3-inches was performed on December 12, 2000. This test revealed no indications in the outer perimeter area, but clearly showed the two rounded indications and a new rounded indication approximately the same size as the other two. The new indication was aligned with the first two and was on the nozzle side of the fillet weld face. The spacing between the three indications was approximately equal. The decision was made to perform additional metal removal in the area of the three rounded indications and to perform another PT test. This test revealed a linear indication that was oriented transverse to the direction of the fillet weld and was approximately 0.75 inches long.

A boat sample of the weld from CRDM Nozzle 21 was taken on December 24, 2000 and sent to the Duke Engineering & Services Metallurgical Lab for analysis. The lab analysis concluded the cracking in the 182-weld metal was due to Primary Water Stress Corrosion Cracking (PWSCC).

After sample removal, the crack was ground out of the CRDM weld and nozzle material exposing a small area of low alloy, steel base metal. One branch of the crack grew from the root of the fillet weld into the nozzle itself, then grew axially along the alloy 600

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nozzle material (no more than 0.4 inch depth) and finally, back into the J-groove to the nozzle/weld annulus interface. It was subsequently concluded from the root cause investigation that this branch was likely the source of the CRDM #21 leak.

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/or dye penetrant inspections were performed on all eight (8) T/C nozzles and CRDM #21.
2. Seven additional CRDMs were eddy current tested and 18 total CRDM nozzles were ultrasonic tested (for extent-of-condition purposes).
3. Two removed nozzle pieces (T/Cs #7 and #8) and a boat sample from the CRDM #21 weld were sent to a metallurgical Lab for analysis.
4. Prior to unit restart,
 - All eight thermocouple nozzles were removed by boring them out from the underside of the vessel head and cutting a 4-inch long, 1.375-inch diameter counter bore in the lower head region. The counter bore diameter was larger than the nominal 1.035-inch diameter of the original hole in the vessel head.
 - For nozzle locations #1, #3, #4, #6, #7 and #8, all remaining indications in the partial penetration welds were removed, and 4-inch long alloy 690 plugs were welded to the existing J-groove weld metal from the underside of the head. The plugs fit the counter bore diameter and are thus larger than the original hole in the vessel head so that the plugs cannot be ejected if the repair welds fail.

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- For nozzles #2 and #5, the cracks extended up to the low alloy, steel base metal of the reactor vessel head. It was undesirable to perform an under-the-head temper bead weld repair at this location. These nozzles were repaired using a 9-inch long alloy 690 plug installed from the underside of the vessel head and welded to the top surface of the vessel head using a temper bead technique. The bottom portions of these plugs also fit the counter bore diameter and are therefore larger in diameter than the original holes through the vessel head such that they cannot be ejected if the repair welds fail.
5. The CRDM #21 nozzle weld cracks and its branches were ground out and the final weld repair performed in accordance with the ASME Code using the temper bead technique. This repair was completed prior to unit restart.
6. On February 18, 2001, an opportunity arose to visually inspect the Unit 3 reactor vessel head while the unit was taken to cold shutdown for planned maintenance to repair a leaking Pressurizer code safety relief valve. During this head inspection, six (6) Unit 3 CRDMs were identified as having boron deposits around their base in a similar fashion to what was discovered on CRDM #21 (Unit 1). After confirming that a pressure boundary leak had occurred, a prompt report was made to the Staff early on February 19, 2001 in accordance with 10CFR50.72 (b)(3)(ii)(B) reporting requirements. Pending results of a (Unit 3) FIP Team investigation, it is planned to report this new finding as either a supplement to this report or as a separate LER.

There are no NRC Commitment items contained in this LER.

Note: Although all eight (8) of the T/C nozzles were removed and plugged and repairs made to the #21 CRDM weld, the potential for future leakage events due to PWSCC of the remaining alloy 600 CRDM nozzle components and 182-weld metal remains a concern for all three Oconee Units. A management plan for the alloy 600 CRDM nozzles and 182-weld metal that focus' on new inspections and proactive corrective actions (repairs) was determined to be the best approach to prevent recurrence of this event. This management plan as well as other planned corrective actions are being addressed via the Oconee Corrective Action Program.

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SAFETY ANALYSIS

T/C Nozzles

Safety analyses submitted by NEI for the industry to the NRC in 1993 show that axial cracks in RPV head nozzles do not pose a safety risk. Specifically,

- 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.

PWSCC cracks are self-limiting by their nature. A crack will grow only as long as there is a driving stress present. At the point stress is relieved by the crack or the crack reaches a region of lower stress, the crack growth is arrested. At this point, the crack, if in a thin walled material, will produce a leak. There is no possibility of a nozzle ejection.

Thermocouple leakage rates were minimal and did not approach Technical Specification limits for unidentified RCS inventory loss. Additionally, there were no reactor building or area radiation alarms, which confirms that RCS boundary leakage was very low. The small amounts of boric acid crystal deposits found around the T/C nozzles had caused no observable corrosion to the vessel head.

CRDM #21

The degraded condition of CRDM Nozzle #21 did not represent a challenge to the nuclear safety of the plant. These cracks were in the weld/base metal and had an axial/radial orientation that would resist CRDM nozzle ejection. As predicted by the stress analysis (and the fact that PWSCC does not occur in carbon steel material) the crack did not extend a significant depth into the vessel head material, but rather grew until it resulted in a small boric acid leak as detected during normal shutdown surveillance walkdowns.

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Based on a boron acid aging evaluation, the leak most likely began either during cycle 18 (September 1998) or cycle 19 (July 1999).

The boat specimen removed from ONS 1 confirmed PWSCC as the mechanism that resulted in the CRDM #21 leak. Leakage from the crack was small due to the relatively tight PWSCC cracks. Similar to the T/C leakage, the CRDM leakage did not exceed Technical Specification limits for unidentified RCS inventory loss, no reactor building or area radiation alarms sounded, and the small amounts of boric acid crystal deposits that were observed around CRDM Nozzle #21 had caused no observable corrosion to the vessel head.

In conclusion, 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.

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.

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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].

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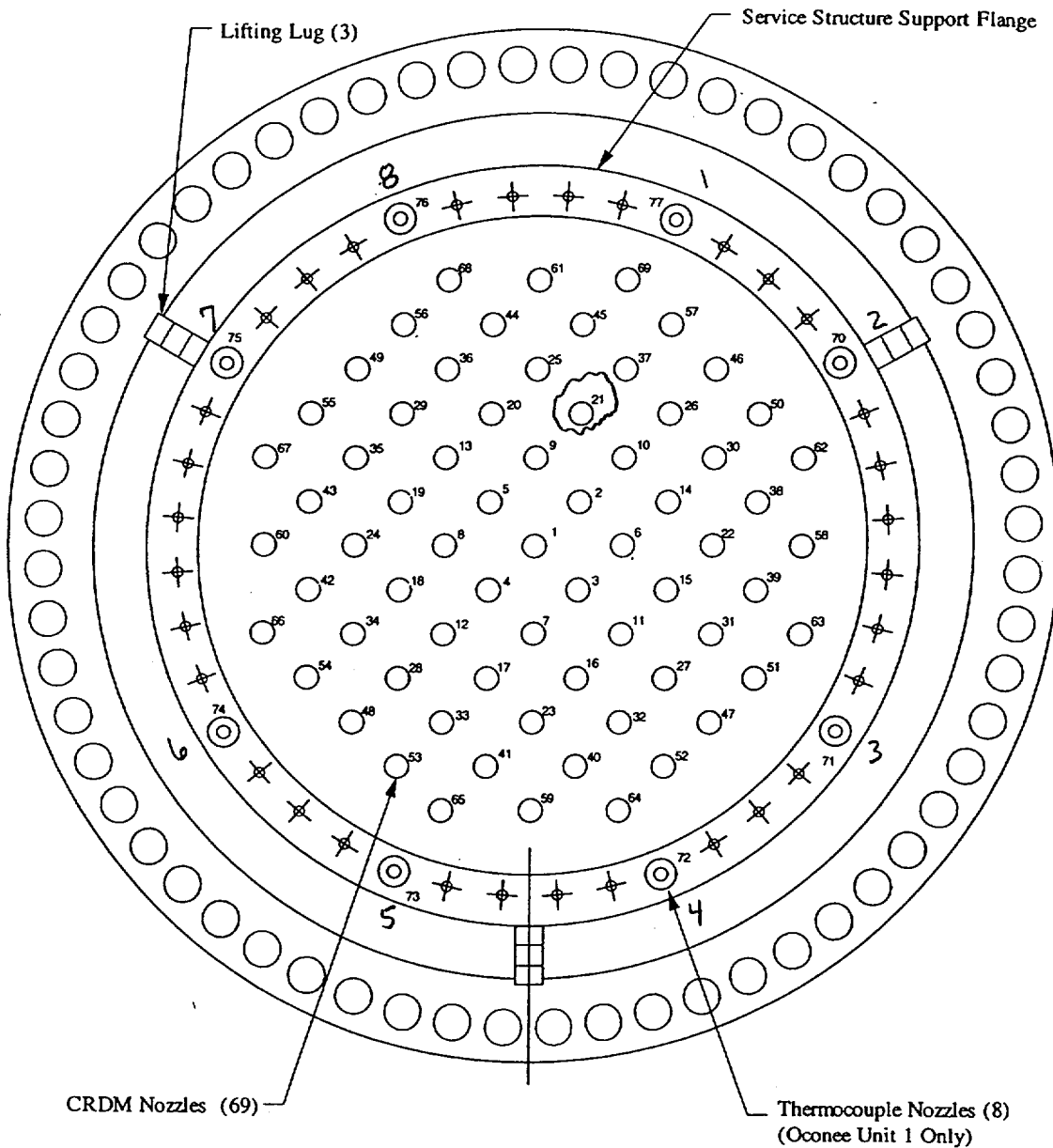
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Oconee Nuclear Station, Unit 1	05000-269	2000	006	01	13	OF	15

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

Figure 1

Reactor Vessel Closure Head Map



LICENSEE EVENT REPORT (LER)

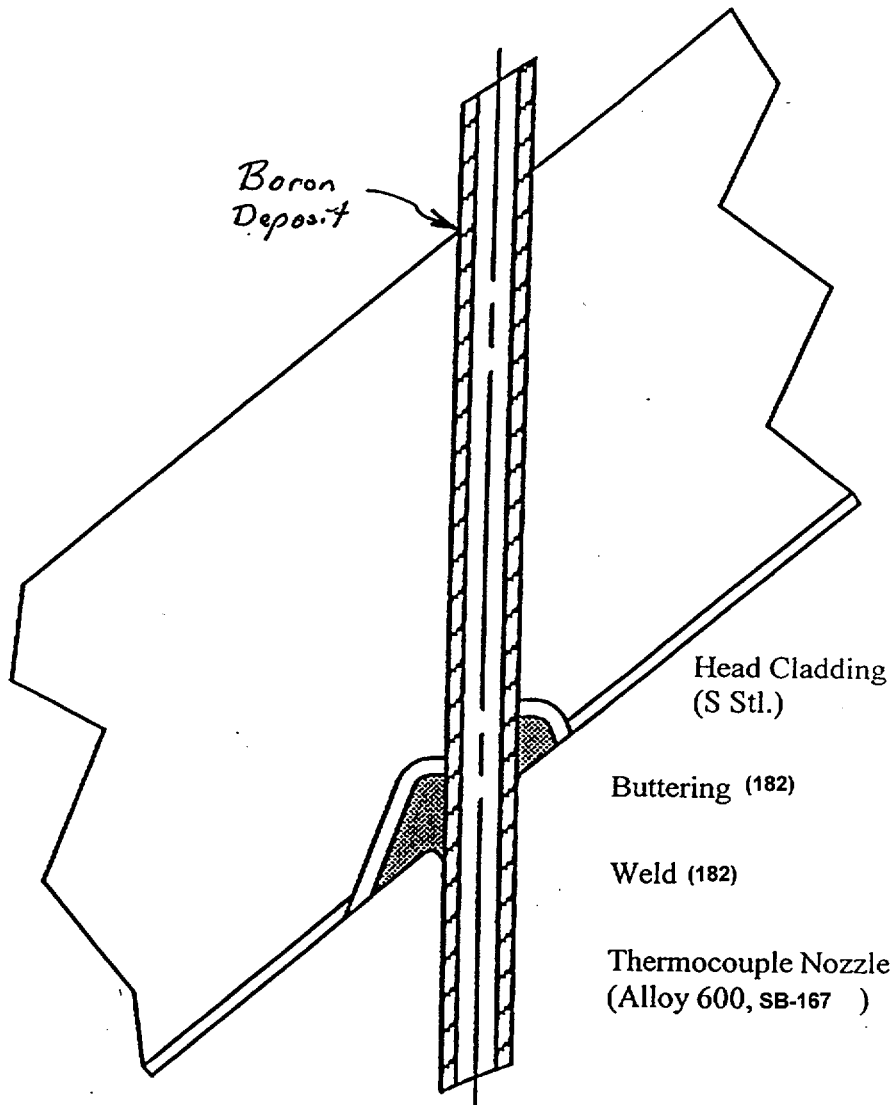
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Oconee Nuclear Station, Unit 1	05000-269	2000	006	01	14	OF	15

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Figure 2

Oconee 1 Thermocouple Penetration (typ.)



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TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Oconee Nuclear Station, Unit 1	05000-269	2000	006	01	15	OF	15

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Figure 3

Oconee CRDM Nozzle Penetration (typ.)

