UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

January 27, 2003

Craig G. Anderson, Vice President,
Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, Arkansas 72801-0967

SUBJECT: ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 - NRC INTEGRATED INSPECTION REPORT 50-313/02-05; 50-368/02-05

Dear Mr. Anderson:

On December 28, 2002, the NRC completed an inspection at your Arkansas Nuclear One, Units 1 and 2, facility. The enclosed report documents the inspection findings, which were discussed with you and other members of your staff on January 3, 2003, and as described in Section 4OA6.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the NRC has identified two findings which are presently characterized as unresolved, requiring additional NRC review. The first one involves the failure to provide complete and accurate information in response to NRC Generic Letter 97-01 regarding a 1988 event which resulted in an excursion in Unit 1 primary water sulfate concentration. The second involves inadequate corrective actions for a repair to Unit 1 Reactor Vessel Head Penetration Nozzle 56 during Refueling Outage 1R16. Although neither finding presents an immediate safety concern because the conditions identified no longer exist, additional review is required by the NRC staff to assess the significance of these findings.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch D
Division of Reactor Projects

Dockets: 50-313
50-368
Licenses: DPR-51
NPF-6

Enclosure:
NRC Inspection Report
50-313/02-05; 50-368/02-05

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313, 50-368
Licenses:	DPR-51, NPF-6
Report No:	50-313/02-05, 50-368/02-05
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	September 22 through December 28, 2002
Inspectors:	R. Bywater, P.E., Senior Resident Inspector J. Clark, Senior Project Engineer, Project Branch D E. Crowe, Project Engineer, Project Branch D J. Keeton, Project Engineer, Project Branch A C. J. Paulk, Senior Project Engineer, Project Branch A K. Weaver, Resident Inspector
Accompanying Personnel:	S. Doctor, Ph.D., Consultant NDE Specialist
Approved By:	Linda Joy Smith, Chief, Project Branch D Division of Reactor Projects
Attachment:	Supplemental Information

SUMMARY OF FINDINGS

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/02-05; 50-368/02-05

IR 05000313-02-05, IR 05000368-02-05; Entergy Operations, Inc.; 9/22/02 - 12/28/02; Arkansas Nuclear One, Units 1 and 2; Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles, Problem Identification and Resolution.

The inspection was conducted by two resident inspectors, two senior project engineers, and two project engineers. The inspection identified two unresolved findings. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified Findings

Cornerstone: Barrier Integrity

- TBD. On October 8, 2002, the inspectors identified an unresolved item associated with repeat reactor coolant system (RCS) boundary leakage. In the ANO U-1 Spring 2001 Outage (1R16), CRDM Nozzle 56 was identified as leaking. Repairs were made to the nozzle weld, and the unit was returned to operation for another cycle. Upon shutdown for Refueling Outage 1R17, repeat leakage of the nozzle was self revealed during visual examination of the reactor vessel head. In 2001, the licensee performed an embedded flaw repair in accordance with Section XI of the ASME Code. However, the licensee recently concluded that this repair method was inadequate to prevent recurrence of the original primary water stress corrosion cracking (PWSCC). They stated that the partial arc of the excavation and overlay did not adequately seal the termination points of the weld. Appendix B, Criterion XVI, of 10 CFR 50, states that in the case of significant conditions adverse to quality, the licensee shall assure that corrective action taken precludes repetition. Although the licensee determined the reactor coolant system boundary leakage from vessel head penetration (VHP) Nozzle 56 was a significant condition adverse to quality, they failed to take adequate corrective actions to preclude repetition. Although the ASME Code repair was intended to last for at least 30 years, the inadequate design of the weld repair for this application of the ASME Code resulted in the repeat failure of the pressure boundary in less than one operating cycle. This leak was repaired in this outage with a more comprehensive strategy, and is therefore not an ongoing safety concern.

  Due to the fact that actual reactor coolant system pressure boundary leakage occurred, the issue is greater than minor and required a Phase 2 significance determination in accordance with NRC Manual Chapter 0609. However, due to NRC management interest in maintaining consistency throughout the Agency for similar leakage, the issue was referred to the Office of Nuclear Reactor Regulation for a Phase 3 determination process. Therefore, this issue is being treated as an unresolved item (URI) until a final characterization of the risk is determined.

- TBD. On November 8, 2002, the inspectors identified an unresolved item associated with a 1988 Unit 1 primary water chemistry sulfate excursion that was not documented in the licensee's response to NRC Generic Letter 97-01. During heatup following the 1988 Unit 1 1R8 Refueling Outage, primary water chemistry sulfate levels exceeded guidelines of the Electric Power Research Institute because of an earlier unintended intrusion of demineralizer resin into the reactor coolant system. NRC Generic Letter 97-01 requested that licensee's provide information regarding occurrences of resin intrusion into the reactor coolant system and effects on reactor coolant system chemistry. A Babcock and Wilcox Owners Group Report which documented licensee inputs in response to Generic Letter 97-01 stated that no noticeable change in reactor coolant system chemistry was noted following the subject event. This statement was not correct and there was no evaluation of the effects of the sulfate excursion that did occur. 10 CFR 50.9 requires that information provided by a licensee to the Commission be complete and accurate in all material respects. This issue is considered unresolved, however, pending NRC review of the licensee's corrected response to Generic Letter 97-01, in order to allow an evaluation of the potential significance of the event and determination of the degree to which the NRC relied upon this information from licensees for regulatory decision making during review of licensee responses to Generic Letter 97-01; as described in the NRC Enforcement Manual for violations involving inaccurate or incomplete information.

  This issue is greater than minor because an actual reactor coolant system sulfate excursion occurred. However, this issue did not pose an immediate safety concern because the identified sulfate excursion was low in magnitude, short in duration, and happened approximately 14 years ago. All Unit 1 reactor vessel head penetration nozzles were inspected during Refueling Outage 1R17 and repairs were made as necessary. Additional review is needed to determine the final characterization of this finding.

Report Details

Summary of Plant Status

Unit 1 began the inspection period at approximately 83.5 percent power and was in a power coastdown of approximately 1.2 percent per day in anticipation of the 1R17 Refueling Outage. On October 4, 2002, Unit 1 operators commenced a power reduction at approximately 25 percent per hour in preparation for Refueling Outage 1R17. On October 4, during this power reduction, Unit 1 operations personnel commenced planned testing of the main turbine overspeed trip functions. On October 4, 2002, with Unit 1 at approximately 42 percent reactor power, the main turbine tripped unexpectedly during the turbine overspeed trip testing activities which resulted in an automatic reactor trip. Unit 1 then began Refueling Outage 1R17 following the reactor trip.

On November 12, 2002, following completion of the Unit 1 1R17 Refueling Outage activities, Unit 1 operators made the reactor critical. On November 13, Unit 1 operators commenced a power escalation and closed the main turbine generator output breakers and ended Refueling Outage 1R17. Unit 1 achieved 100 percent reactor power on November 15, and remained at or near 100 percent reactor power for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power. On December 19, 2002, Unit 2 experienced an automatic reactor trip from 100 percent power following a main generator lockout and turbine trip. The generator lockout was caused by failure of a generator anti-motoring protective relay. On December 19, following completion of repairs, post-transient review and recovery activities, Unit 2 operators made the reactor critical. On December 20, Unit 2 operators closed the main generator output breakers and commenced a power escalation. With the unit at approximately 13 percent power, an automatic turbine trip occurred due to high turbine bearing vibration caused by a turbine packing rub (no reactor trip required). Following a turbine stabilization period that day with the turbine on the turning gear, Unit 2 operators rolled the turbine to rated speed, placed the unit online with the reactor at approximately 15 percent power, and continued power escalation. Turbine vibration levels remained normal during the power escalation. On December 21, Unit 2 achieved 100 percent power and remained at or near 100 percent power throughout the remainder of the inspection period.

REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [REACTOR-R]

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During the week of November 4, 2002, the inspectors reviewed both Units 1 and 2 preparations for freeze protection of plant equipment. Freeze protection preparations were documented in Unit 1 Procedure 1307.037, "Unit 1 Plant Freeze Protection Testing," Revision 13, and Unit 2 Procedure 2106.032, "Unit Two Freeze Protection Guide," Revision 9. The inspectors walked down the Unit 1 and 2 service water intake structures to verify that the freeze protection measures were performed and in place in order to protect safety-related equipment from being affected by freezing weather conditions. The inspectors also reviewed the cold weather requirements for the emergency cooling pond. The inspectors discussed the status of freeze protection preparations with the appropriate system engineer.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Complete Walkdown (71111.04S)

a. Inspection Scope

On November 26, 2002, the inspectors performed a complete system walkdown of accessible portions of the Unit 2 low pressure safety injection system, Train B. During this walkdown, the inspectors verified correct valve alignment, electric power availability, and no adverse material condition of system components. Positions of valves and electrical power breakers were compared to Procedure 2104.040, "LPSI System Operations," Revision 34. Drawing No. M-2232, "Safety Injection System," Revision 110, was referenced during this inspection. Minor discrepancies identified by the inspectors were documented by the licensee in Condition Report ANO-2-2002-2080.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Inspection (71111.05Q)

a. Inspection Scope

During the week of December 2, 2002, the inspectors performed inspections of the following fire zones to ensure that conditions were consistent with the requirements of the licensee's fire protection program for fire protection systems design, control of transient combustibles and ignition sources, fire detection and suppression capability, fire barriers, and any related compensatory measures. The Fire Hazards Analysis Report, Revision 7, was used as a reference during these inspections.

- Unit 1, Emergency Diesel Generator No. 1, Fire Zone 87-H
- Unit 1, Emergency Diesel Generator No. 2 Room, Fire Zone 86-G
- Unit 2, Battery No. 2D-11, Fire Zone 2103-V

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination (NDE) Activities

a. Inspection Scope

The inspector reviewed the completed documentation for the ASME Code Section XI specified examinations listed below:

System	Component/Weld Identification	Examination Method
Reactor Coolant	Pressurizer Relief Nozzle Between W-Y; Component ID 05-014R	Ultrasonic Examination
Reactor Coolant	Pressurizer Relief Nozzle Between Z-W; Component ID 05-015R	Ultrasonic Examination
Low Pressure Injection	Pipe to Elbow Circumferential Weld; Conponent ID 36-017	Liquid Penetrant Examination; Ultrasonic Examination
Main Steam	Pipe to Elbow Circumferential Weld; Component ID 31-054	Magnetic Particle Examination; Ultrasonic Examination
High Pressure Injection	Elbow to Pipe Circumferential Weld; Component ID 21-297	Liquid Penetrant Examination
High Pressure Injection	Pipe to Elbow Circumferential Weld; Component ID 21-296	Liquid Penetrant Examination
Main Feedwater	Pipe to Elbow Circumferential Weld; Component ID A-10015	Ultrasonic Examination
Main Feedwater	Elbow to Pipe Circumferential Weld; Component ID A-10016	Ultrasonic Examination
Main Steam	Pipe to Sweep-O-Let Branch Connection; Component ID 31-003	Magnetic Particle

System	Component/Weld Identification	Examination Method
High Pressure Injection	Elbow to Pipe Circumferential Weld; Component ID 21-185	Liquid Penetrant Examination; Ultrasonic Examination
High Pressure Injection	Pipe to Pipe Circumferential Weld; Component ID 20-008A	Liquid Penetrant Examination; Ultrasonic Examination
High Pressure Injection	Elbow to Pipe Circumferential Weld; Component ID 20-008A	Liquid Penetrant Examination; Ultrasonic Examination
High Pressure Injection	Elbow to Pipe Circumferential Weld; Component ID 21-232	Liquid Penetrant Examination; Ultrasonic Examination
High Pressure Injection	Elbow to Pipe Circumferential Weld; Component ID 21-287	Liquid Penetrant Examination

During the review of each examination, the inspector verified that the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspector also verified that indications revealed by the examinations were compared against the previous outage examination reports and the ASME code-specified acceptance standards and appropriately dispositioned. The review was performed on all completed examinations.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the two condition reports issued during the past year on inservice inspection activities. The inspector verified that the licensee identified, evaluated, corrected, and trended problems.

b. Findings

No findings of significance were identified.

.3 Steam Generator Tube Integrity

a. Inspection Scope

The inspector reviewed the in-situ screening criteria to verify that the criteria was in accordance with industry guidelines. The estimated size and number of tube flaws identified up to the date of the inspection were compared, by the inspector, to the operational assessment predictions from the previous outage. The inspector also reviewed the eddy current examination scope and expansion criteria to determine if the Technical Specifications, industry guidelines, and commitments to the NRC were being met.

The inspector reviewed the areas of potential degradation (based on site-specific and industry experience) to verify that such areas were being inspected. The inspector also reviewed the leakage history for the steam generators to verify that the leakage was less than 3 gallons per day during operations. The eddy current probes and equipment were reviewed to ascertain if they were properly qualified for the expected types of tube degradation.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Requalification Activities Review by Resident Staff (71111.11Q)

a. Inspection Scope

On December 5, 2002, the inspectors observed the Unit 2 licensed operator simulator qualification training Scenario SPG-2-03-02-04, "Shifty's Choice," conducted for Training Cycle 2003-02-05. The inspectors compared their observations to the applicable abnormal and emergency operating procedures, the emergency plan procedures, and applicable Technical Specifications.

On December 5, 2002, the inspectors observed the Unit 2 licensed operator simulator qualification training Scenario SPG 2-03-02-03, "Unannounced Casualties," conducted for Training Cycle 2003-02-05. The inspectors compared their observation to the applicable abnormal and emergency operating procedures, the emergency plan procedures, and applicable Technical Specifications.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

.1 Routine Maintenance Effectiveness Inspection (71111.12Q)

a. Inspection Scope

The inspectors independently verified that licensee personnel properly implemented 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspectors used the reactor oversite process Inspection Procedure 71111.12, "Maintenance Effectiveness," to perform the inspection. The following equipment performance problems were reviewed:

- Unit 2 intake structure ventilation system Fans 2VEF-25A and 2VEF-25B failure to start on an Engineered Safeguards Signal (CR ANO-2-2001-0092)

- Seal leakage on Unit 1 decay heat removal Pump P-34B (CR ANO-1-2002-1783, CR ANO-1-2000-0167, CR ANO 1-2001-0367, and CR ANO-1-2001-1489)

The inspectors focused the review on whether the structures, systems, or components (SSCs) that experienced problems were properly characterized in the scope of the program. They also reviewed whether the SSC failure or performance problem was properly characterized. The inspectors assessed the adequacy of the licensee's significance classification for the SSC. This included the appropriateness of the performance criteria established for the SSC (if applicable) and the adequacy of corrective actions for SSCs classified in accordance with 10 CFR 50.65 a(1) as applicable.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors evaluated and discussed with the licensee the risk assessments listed below to verify that assessments were performed when required and appropriate compensatory actions were taken. The inspectors reviewed these assessed risk configurations against actual plant conditions and any in-progress evolutions or external events to verify that the assessments were accurate, complete, and appropriate for the conditions. In addition, the inspectors walked down the control room and plant areas to verify that compensatory measures identified by the risk assessments were appropriately performed.

- Unit 1 Train B decay heat room component risk assessment associated with Room Cooler VUC-1D leak on October 15, 2002

- Unit 1 No. 2 emergency diesel generator risk assessment associated with surveillance testing conducted on November 18, 2002

- Unit 2 No. 1 emergency diesel generator risk assessment associated with surveillance testing conducted on December 5, 2002

- Daily review of risk assessments and score cards during Refueling Outage 1R17 completed in accordance with Arkansas Nuclear One, Shutdown Operations Protection Plan, dated September 9, 2002, and comparison to actual plant conditions to ensure that the licensee implemented acceptable defense-in-depth strategies for critical safety functions

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Nonroutine Plant Evolutions and Events (71111.14, 71153)

.1 Unit 1 Refueling Outage 1R17

a. Inspection Scope

The inspectors reviewed and observed operator performance and response during portions of the Unit 1 shutdown, cooldown, transfer to shutdown cooling system operation, and draining the RCS to reduced inventory for steam generator maintenance during Refueling Outage 1R17. These activities were conducted in accordance with Procedure 1102.016, "Power Reduction and Plant Cooldown," Revision 3; Procedure 1104.004, "Decay Heat Removal Operating Procedure," Revision 69; and, Procedure 1103.011, "Draining and Nitrogen Blanketing the RCS," Revision 27.

During the power reduction in preparation for starting 1R17, Unit 1 experienced a reactor trip from approximately 42 percent power while turbine trip testing was in progress. The inspectors observed operator posttrip response in accordance with Procedure 1202.001, "Reactor Trip," Revision 28, to verify that plant systems responded as expected and that the plant was maintained in a safe condition. This event was reported to the NRC in Licensee Event Report 50-313/2002-002-00. The inspectors will review the significance of this event during a future inspection for closeout of the licensee event report.

During the Mode 3 walkdown of the reactor building on October 5, 2002, the licensee discovered that Valve PSV-1800, pressurizer sample line thermal relief, was leaking past its seat at a flow rate into the reactor building of approximately 0.22 gpm. The purpose of this relief valve was to provide overpressure protection of the associated reactor building penetration following isolation during an accident. The inspectors reviewed the licensee's operability evaluation, documented in Condition Report ANO-1-2002-1146, to verify that the as-found condition did not violate overall reactor building leakage limits.

Also during the Mode 3 reactor building walkdown on October 5, 2002, the licensee discovered that a weld on a drain connection to the 'C' High Pressure Injection Line was leaking at a flow rate into the reactor building of approximately 0.2 gpm. The leak was located upstream of 'C' High Pressure Injection Line Check Valve MU-34C and Manual Isolation Valve MU-45C. The inspectors reviewed and discussed with operations and engineering personnel the Technical Specification requirements for RCS leakage and high pressure injection system operability for this as-found condition. The condition was documented and evaluated in Condition Report ANO-1-2002-1147. The licensee concluded that the leakage was not RCS pressure boundary leakage as defined in the Technical Specifications and clarified in the Technical Specification bases because the leak was located upstream of the check valve and manual isolation valve. Additionally, with engineering input, the licensee concluded that the high pressure injection system was operable because it remained capable of performing its safety function. Following additional discussions with NRC management, the inspectors agreed that the leak was not RCS pressure boundary leakage as defined in the Technical Specifications and no action statement entry was required. However, the inspectors informed the licensee that the leak location was part of the RCS pressure boundary as defined in 10 CFR 50.2. On October 8, 2002, the licensee reported this condition as being a serious degradation of one of the plant's principal safety barriers in accordance with 10 CFR 50.72. This report was followed by submittal of Licensee Event Report 50-313/2002-004. The inspectors will review the significance of this event during a future inspection for closeout of the licensee event report.

The inspector's review of RCS pressure boundary leakage associated with a leaking vessel head penetration nozzle is documented in Section 4OA5 of this inspection report.

On October 24, 2002, a worker was injured while working under the Unit 1 reactor vessel head. The worker was transported to the local hospital for treatment, accompanied by licensee radiation protection personnel, and was found to be slightly contaminated. Radiation protection personnel performed decontamination activities at the hospital and returned all contaminated material to the plant for disposal. The issue was documented in Condition Report ANO-1-2002-1538 and the inspectors verified that appropriate industrial safety actions were taken to review the event and prevent future injuries. The inspectors and a radiation protection specialist from the regional office interviewed radiation protection personnel to verify appropriate means were taken to ensure no offsite spread of contamination occurred. The licensee reported this occurrence to the NRC in accordance with 10 CFR 50.72.

On November 10-12, 2002, the inspectors observed portions of the Unit 1 plant heatup and startup activities at the conclusion of Refueling Outage 1R17. The inspectors verified that Unit 1 operators performed the startup activities in a safe manner and in accordance with Procedure 1102.002, "Plant Startup," Revision 69.

b. Findings

No findings of significance were identified.

.2 Unit 2 Entry into Abnormal Operating Procedure for Reactor Coolant System Leakage

On November 7, 2002, Unit 2 operators entered Abnormal Operating Procedure 2203.016, Excess RCS Leakage, Revision 9. Chemical and volume control system letdown backpressure Control Valve 2CV-4811 failed closed, resulting in letdown pressure exceeding the 600 psig setpoint of Relief Valve 2PSV-4822, which discharged reactor coolant to the boron management system. Operators terminated the loss of inventory by isolating letdown and securing charging flow, allowing Valve 2PSV-4822 to reseat and pressurizer level to be maintained. The inspectors verified that Unit 2 operators responded appropriately in accordance with the abnormal operating procedure. Letdown and charging flow were restored later that day and maintenance personnel performed troubleshooting and repair activities on Valve 2CV-4811. Condition Reports ANO-2-2002-2020 and ANO-2-2002-2021 were initiated and reviewed by the inspectors for this event.

.3 Unit 2 Reactor Trip from 100 Percent Power and Recovery Activities

a. Inspection Scope

On December 19, 2002, Unit 2 experienced an automatic reactor trip from 100 percent power. The inspectors responded to the control room and observed and reviewed posttrip recovery activities performed by Unit 2 operators to verify their appropriate response, that plant-safety systems performed as designed, and that the plant was maintained in a safe condition. Procedure 2201.001, "Standard Post Trip Actions," Revision 6 and Procedure 1015.037, "Post Transient Review," Revision 3, were referenced during this review. The cause of the reactor trip was determined to be the result of the failure of a main generator anti-motoring relay, which caused a generator lockout, main turbine trip, and automatic reactor trip on high RCS pressure.

The inspectors observed and reviewed Unit 2 restart activities, including resolution of turbine vibration problems which caused an automatic turbine trip due to high vibration on December 20, 2002.

The inspectors will review the significance of this event during review of the associated licensee event report, which is due to be submitted to the NRC in February 2003.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the correctness of evaluations, the use of compensatory measures, if needed, and compliance with the Technical Specifications. The inspectors' review included a verification that operability determinations were made as specified by the licensee's Procedure LI-102, "Corrective

Action Process," Revision 2, and Procedure 1000.104, "Condition Reporting and Immediate Reportability Determinations," Revision 17. The technical adequacy of the determinations was reviewed and compared to the Technical Specifications, Technical Requirements Manual, Updated Final Safety Analysis Report, associated licensing-basis documentation, as appropriate. The operability determinations that were reviewed were documented in the following condition reports:

ANO-1-2001-0549	Evaluation of body to bonnet leakage from Unit 1 makeup and purification system Valve MU-1235-2
ANO-1-2002-1130 and ANO-1-2002-1233	Evaluation of boric acid on Unit 1 incore instrumentation nozzles
ANO-1-2002-1147	Evaluation of Unit 1 high pressure injection system operability following identification of drain stack weld leak
ANO-1-2002-1371	Operability assessment of Unit 1 'B' decay heat room components required for Mode 6 operation following VUC-1D Cooler leak
ANO-1-2002-1576	Evaluation of Unit 1 borated water storage tank suction standpipe drawing discrepancy
ANO-2-2002-1971	Evaluation of Environmental qualification of Unit 2 hot leg injection Valve 2CV-5102-2
ANO-2-2002-02082	Evaluation of postmaintenance testing of Unit 2 low pressure safety injection Valve 2CV-5037-1

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

During the week of November 4, 2002, the inspectors reviewed the Unit 1, Unit 2, and common operator workaround lists to determine if the functional capability of involved systems, or the human reliability in responding to an initiating event had been affected by the associated workarounds. The inspectors assessed the cumulative effect of the workarounds for system misoperation, affect on multiple mitigating systems, and ability of operators to respond to plant transients or accidents in a correct and timely manner. The inspectors interviewed operations department managers to verify that the workarounds had been appropriately evaluated and corrective actions were being taken or planned in a timely manner.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

.1 Annual Review (71111.17A)

a. Inspection Scope

The inspectors reviewed the implementation of the following permanent plant modification to verify that the modification was designed and installed in accordance with applicable station procedures controlling installation of modifications, including Procedure DC-115, "Engineering Request Response Development," Revision 1, and Procedure LI-101, "10CFR50.59 Review Program," Revision 2:

Engineering Request 981055N101, "Borated Water Storage Tank Vortex Breaker"

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

For the maintenance activities identified below, the inspectors observed the postmaintenance testing activities in the control room or locally and/or review the test data obtained from the field. The inspectors observed whether the tests were performed in accordance with procedures, that the procedures' acceptance criteria were consistent with the Technical Specifications and the supporting license change application, and the results recorded met the test acceptance criteria. In addition, the inspectors verified that startup test deficiencies were recorded and resolved. These activities included:

- Unit 1 Decay Heat Removal Pump B testing in accordance with Procedure 1104.004, "Decay heat Removal Operating Procedure," Revision 69, conducted on October 23, 2002

- Unit 1 Decay Heat Removal Pump B testing in accordance with Procedure 1104.004, "Decay Heat Removal Operating Procedure," Revision 69, conducted on October 24, 2002

- Unit 2 Control Element Assembly (CEA) No. 65 postmaintenance and monitoring activities following repairs of a failed sensor in accordance with Maintenance Action Item 72310, conducted on December 18, 2002

- Unit 2 Low Pressure Safety Injection Valve 2CV-5037-1 testing in accordance with Procedure 2305.005, "Valve Stroke and Position Verification," Revision 21, Procedure 2104.040, "LPSI System Operations," Revision 34, Procedure 1015.015, "Unit 2 Operations Forms," Revision 24, and Maintenance Action Item 39293, conducted on November 26, 2002. Engineering Request 010911R201 associated with this test was also referenced during the inspection

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

Throughout Unit 1 Refueling Outage 1R17, the inspectors reviewed weekly and daily work schedules to identify risk-significant evolutions and maintenance activities. The inspectors reviewed the Unit 1 shutdown operations protection plan prior to the outage to ensure that the licensee had considered risk, had developed mitigation strategies for losses of key safety functions, and had adhered to operating license and Technical Specification requirements. This included verification that the licensee had appropriately incorporated applicable portions of the Unit 1 Improved Standard Technical Specifications, which were effective for the first time during Refueling Outage 1R17. The inspectors observed portions of the plant cooldown and RCS draindown to reduced inventory for steam generator nozzle dam installation. The inspectors also reviewed implementation of overtime guidelines or limitations as required by Technical Specifications.

The inspectors toured areas of the reactor building on multiple occasions during the refueling outage to inspect refueling and maintenance activities and assess the material condition of equipment not normally accessible for visual inspection. The inspectors also reviewed implementation of Procedure 1032.037, "Inspection and Evaluation of Boric Acid Leaks," Revision 0, and Procedure 5000.005, "Boric Acid Corrosion Prevention Program Administration," Revision 1.

The inspectors observed portions of the Unit 1 reactor refueling activities in accordance with Procedure 1502.004, "Control of Unit 1 Refueling," Revision 33, during core reload on October 20, 2002, to verify that fuel handling activities were accomplished in accordance with the procedure and that Technical Specification requirements were met.

Other refueling outage inspection activities are discussed in Section 1R14 of this report.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed from either the control room or locally the performance of and/or reviewed the documentation for the following surveillance tests. This was done to verify that the surveillance tests were performed in accordance with approved licensee procedures and met Technical Specification requirements. In addition, the applicable test data was also reviewed to verify whether they met Technical Specifications, Updated Final Safety Analysis Report, and licensee procedure requirements.

- Procedure 1102.001, Supplement 5, "RCS Leak Test," Revision 61 and Procedure 5120.242, "Unit 1 Post Outage Pressure Test," Revision 6, performed on November 11, 2002

- Procedure 1305.006, "Integrated ES System Test," Revision 19, performed on October 25, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

During the month of October 2002, the inspectors reviewed the implementation of Attachment E, "Installation of Temporary Fire Pump," of Procedure 1104.032, "Fire Protection Systems," Revision 55. This procedurally controlled temporary alteration was installed during Refueling Outage 1R17 to supply cooling loads from the fire water system to preclude start and run cycles on the permanently installed fire water pumps. The inspectors confirmed that this temporary alteration was properly installed as authorized by the procedure.

b. Findings

No findings of significance was identified.

4. Other Activities [OA]

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspector reviewed Units 1 and 2 operations stations logs and monthly operating logs for the third quarter of 2002 to verify the accuracy and completeness of the data used to calculate and report the following performance indicators in accordance with Procedure LI-107, "NRC Performance Indicator Process," Revision 2. Interviews were also conducted with system engineers.

- Unit 1Emergency AC unavailability
- Unit 2 Emergency AC unavailability
- Unit 1 HPSI unavailability
- Unit 2 HPSI unavailability
- Unit 1 DH unavailability
- Unit 2 LPSI unavailability
- Unit 1 EFW unavailability
- Unit 2 EFW unavailability
- Unit 1 Safety system functional failures
- Unit 2 Safety system functional failures

The inspectors evaluated licensee performance indicator collection and reporting practices against the standards of NEI 99-02, “Regulatory Assessment Performance Indicator Guideline.”

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Adverse Trend of Unit 1 Reactor Coolant System Leak Rate

a. Inspection Scope

The inspectors reviewed the licensee's evaluations of an adverse trend in Unit 1 RCS unidentified leakage that had been documented in Condition Reports ANO-1-2002-0552 and ANO-1-2002-0958. The review included a verification that the licensee had accurately documented the identified problem, assessed the significance of the problem, and identified corrective actions to identify and eliminate the cause of the adverse trend in RCS leakage. Principal sources of this unidentified RCS leakage found during Refueling Outage 1R17 are discussed in Section 1R14 of this report.

b. Findings

No findings of significance were identified.

.2 Review of Boric Acid Accumulation on Unit 1 Incore Instrumentation Nozzles

a. Inspection Scope

In response to a condition identified at another utility involving the observed accumulation of boric acid on incore instrumentation nozzles located under the reactor vessel, the licensee initiated Condition Report ANO-1-2002-1130 to evaluate the issue and perform inspections during Refueling Outage 1R17 to identify whether a similar condition existed on Unit 1. During the licensee's visual inspection of the Unit 1 incore instrumentation nozzles, some boric acid accumulation was noted on four of the 52 incore instrumentation nozzles, as well as the walls of the reactor pedestal, the floor of the reactor cavity, and some incore instrumentation tubing in the reactor cavity and instrumentation tunnel. A boroscopic examination was performed of the reactor vessel surface in the vicinity of the four nozzles of interest, including the annulus between the reactor vessel and nozzle to identify signs of boric acid deposits that could be indicative of a RCS pressure boundary leak. The identified conditions were documented and evaluated in Condition Report ANO-1-2002-1233. In addition to the inspectors' review of the licensee's videotaped inspections, this issue was the subject of discussions between the licensee and NRC staff from the Region IV Office and the Office of Nuclear Reactor Regulation. In response to NRC staff questions, the licensee provided a response to a request for additional information on November 21, 2002, documenting the as-found condition and assessment of the source of the identified boric acid traces. The traces of boric were determined to not be RCS pressure boundary leakage; rather, they were determined to have resulted from reactor refueling cavity seal plate leakage that had occurred during previous refueling outages. No indications of RCS leakage from the incore instrumentation nozzles was identified during the postoutage pressure test.

b. Findings

No findings of significance were identified.

.3 Review of Past Event Involving Intrusion of Resin into the Unit 1 Reactor Coolant System

a. Inspection Scope

During review of the licensee's inspection of Unit 1 reactor vessel head penetration nozzles for indications of cracking during Refueling Outage 1R17, the inspectors performed a historical review of the licensee's response to NRC Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," (including Babcock and Wilcox Owner's Group Report 51-1257700-00), and the licensee's Industry Experience Evaluations for NRC Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," and Westinghouse Nuclear Safety Advisory Letter (NSAL) 94-028, "Sensitized Alloy 600 Material and RCS Sulfur Intrusion."

b. Findings

Intergranular attack of Alloy 600 components had been identified as a potential degradation mechanism resulting from the high temperature degradation of demineralizer resin into chemical products (particularly sulfates and chlorides) in the RCS. Generic Letter 97-01, in particular, requested that licensee's provide information to the NRC staff regarding any instances of resin intrusion into the RCS, whether any primary water chemistry guidelines established by the Electric Power Research Institute had been exceeded, and an assessment of the potential for intergranular attack. The licensee’s evaluations associated with these documents concluded that the identified industry issue of intergranular attack of Alloy 600 components was not applicable at ANO Unit 1 because the plant had not operated with resin present in the RCS. However, the licensee’s evaluations did identify that a resin intrusion event occurred during Refueling Outage 1R8 in 1988, documented in Condition Report ANO-1-1988-0285.

The inspectors reviewed Condition Report ANO-1-1988-0285. During Refueling Outage 1R8 on September 30, 1988, an unanticipated transfer of mixed-bed demineralizer resin from the borated water storage tank to the RCS occurred. Resin accumulation in the borated water storage tank was determined to have been caused by previous operational problems with the spent fuel demineralizer system that had caused unexpected backflushing of resin to tank. The total amount of resin transferred to the RCS was estimated at approximately 849 liters. Cleanup activities were performed that reduced the amount of unrecoverable resin in the RCS to an estimated 3 liters. Prior to plant heatup at the conclusion of Refueling Outage 1R8, the licensee completed an evaluation that determined that continued operation was acceptable. The basis for this acceptability was that a primary water chemistry would be monitored during the heatup and impurities would be removed prior to increasing system temperature above predetermined temperature plateaus. Work Plan 1409.153, "After 1R8 Resin Intrusion Monitoring," Revision 0, was written to provide instructions for this activity. Work Plan 1409.153 provided the following limits of interest:

- If sulfate concentration was between 100 and 250 ppb, reduce concentration to less than 100 ppb within 24 hours or reduce RCS temperature

- If sulfate concentration was greater than 250 ppb, then immediately reduce RCS temperature

- Sulfate concentration must be less than 50 ppb for power operation

The inspectors reviewed the results of Work Plan 1409.153 and compared results obtained to contemporary primary water chemistry limits contained in Procedure 1000.106, "Primary Chemistry Monitoring Program," Revision 5. The inspectors identified that during heatup following Refueling Outage 1R8, primary water sulfate concentration exceeded 50 ppb for approximately 12 days from November 29 through December 11, 1988, with a maximum recorded concentration of 230 ppb on November 30, 1988. During discussions with the licensee, the inspectors were informed that one sample provided a result in excess of 250 ppb (not recorded in the work plan or station log), requiring an immediate temperature reduction, but the next sample obtained approximately 30 minutes later produced a result less than 250 ppb, therefore no temperature reduction was performed.

The inspectors noted that the results obtained in Work Plan 1409.153 were inconsistent with the licensee's industry events evaluations and Babcock and Wilcox Owners Group Report 51-1257700-00, which stated that no noticeable change was detected in RCS chemistry. The licensee agreed that the information contained in these documents was inaccurate with respect to the failure to identify that resin decomposition products were produced during the heatup following Refueling Outage 1R8. The licensee initiated Condition Report ANO-1-2002-1687 documenting this finding and stated that they intended to submit a supplement to their response to Generic Letter 97-01. The inspectors considered this a potential violation of 10CFR50.9 for providing inaccurate or incomplete information to the NRC. This issue is considered unresolved, however, pending NRC review of the licensee's corrected response in order to allow an evaluation of the potential significance of the event and determination of the degree to which the NRC relied upon this information from licensees for regulatory decision making during review of licensee responses to Generic Letter 97-01; as described in the NRC Enforcement Manual for violations involving inaccurate or incomplete information (URI 50-313/2002-05-01).

The inspectors concluded that this issue did not pose an immediate safety concern because the identified sulfate excursion was low in magnitude, short in duration, and happened approximately 14 years ago. Additionally, all Unit 1 reactor vessel head penetration nozzles were inspected during Refueling Outage 1R17 and those with rejectable indications were repaired.

4OA3 Event Followup (71153)

The following licensee event reports were issued during this inspection period. Evaluation of their significance will be performed and documented in a future inspection report:

(Open) Licensee Event Report 50-313/2002-001-00: Main steam safety valve as-found lift settings were not within Technical Specification requirements

(Open) Licensee Event Report 50-313/2002-002-00: Main turbine trip due to binding of the mechanical trip spool valve resulted in an automatic actuation of the reactor protection system

(Open) Licensee Event Report 50-313/2002-003-00: Reactor coolant system pressure boundary leakage from a crack in a control rod drive mechanism nozzle reactor vessel head penetration weld due to previous weld repair method

(Open) Licensee Event Report 50-313/2002-004-00: Reactor coolant system pressure boundary leakage from a fatigue stress crack of a weld connecting a drain line to a high pressure injection line

As discussed in Section 1R14 of this report, a Unit 2 reactor trip occurred on December 19, 2002. The licensee event report for this event had not been issued at the conclusion of the inspection period.

4OA5 Other Activities

A. Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (TI 2515/145 and TI 2515/150)

.1 Bulletin 2001-01/2002-01 Responses and Inspection Overview

On October 7 through November 1, 2002, the inspectors performed NRC Inspection Manual Temporary Instruction (TI) 2515/145 for Unit 1 during Refueling Outage 1R17. They reviewed the licensee's inspection plan, and the NRR assessment, in response to NRC Bulletins 2001-01 and 2002-01 regarding PWSCC of VHP nozzles. The inspectors noted that ANO Unit 1 was considered a highly-susceptible plant (Bin 1) according to Bulletin 2001-01. The inspectors noted that the bulletin recommended a 100 percent volumetric examination of each of the 69 reactor VHP nozzles. The licensee conducted a bare metal examination of the top of the reactor vessel head using a robot crawler and camera, between the top of the head and the permanent insulation. The inspectors reviewed the licensee's procedures and methods for this examination, and observed video tapes and still images of the inspection. The licensee's methodology also employed both ultrasonic (UT) and eddy current (ET) examination of the inside diameter (ID) of each VHP nozzle. Through discussions with the licensee, and conferences with NRR, the inspectors assessed the validity of the licensee's methods to meet the intent of NRC Bulletin 2001-01.

On October 18, 2002, the NRC issued Inspection Manual TI 2515/150 which gave additional scope and clarification for inspections, based upon previous TI 2515/145 results and the responses to NRC Bulletin 2002-02. Based upon ANO Unit 1 susceptibility and the preliminary inspection findings from this outage, NRC management directed the inspectors to conduct Inspection Manual TI 2515/150 during this outage.

The inspectors reviewed the NDE procedures, the welding repair procedures, and the qualification of personnel to these methods for the two contractors utilized by ANO during 1R17. Due to the proprietary nature of both the Westinghouse and Framatome methods, these documents are not specifically referenced or detailed in this report.

Due to the NRC's need to gather and document information regarding VHP nozzle cracking and inspections, TI 2515/145 and 2515/150 called for additional documentation beyond that normally addressed by Manual Chapter 0612. Additional information is provided in this section for documentation and tracking, rather than enforcement or inspection follow up.

.2 Bare Metal Visual Examinations

a. Inspection Scope

On October 7, 2002, the inspectors conducted an exterior visual examination of the reactor vessel head, prior to its removal from the vessel. The inspectors observed the flange bolt area, service structure support skirt, and areas of the head that were visible through the service structure support skirt access mouse holes. The inspectors also examined the structure for evidence of significant boric acid spillage or accumulation.

On October 8-10, 2002, the inspectors monitored the licensee's top of the vessel head bare metal examination. The inspectors reviewed the tapes and still images of the inspection of all 69 of the VHP nozzles.

b. Findings

The inspectors saw no visual indications of significant boric acid accumulation or spillage on the lower portions of the reactor vessel head. The inspectors identified one apparent boric acid stain trail coming from under the insulation, to the vessel flange area. This was subsequently determined to be from Nozzle 56. The inspectors were able to visually examine eight of the peripheral nozzles, through the access mouse holes. The inspectors saw no accumulations of boric acid in this peripheral area.

On October 7, 2002, the licensee identified RCS boundary leakage at VHP Nozzle 56, through the bare metal visual examination. This leakage was evidenced by boric acid accumulation at the downhill side of the nozzle, kernels of boric acid in approximately 180 degrees of the circumference of the nozzle annulus, and a boric acid stain trailing down the reactor vessel head from this nozzle. The licensee and the inspectors noted that this was the same nozzle that had been repaired after leakage was identified in the previous refueling outage. The licensee concluded that Nozzle 56 was the only VHP nozzle with evidence of leakage, and entered that finding in their corrective action program.

Through review of the tapes and photographs, the inspectors identified additional nozzles that they considered had potential evidence of leakage. The inspectors identified seven additional nozzles to the licensee as having minor boric acid crystal accumulation in the annulus of the nozzle. In addition, the inspectors identified another 23 nozzles as having items such as boric acid crystals in the vicinity of the nozzle and/or stains running down the slope of the head from these nozzles. The inspectors told ANO management and engineers that while what they observed was not conclusive visual evidence of nozzle leakage, they could not understand how the licensee concluded these were nonleakers, without further review or evaluation.

Subsequently, the licensee provided the inspectors with additional information about the suspect nozzles. The inspectors also conducted additional interviews with the engineering personnel that conducted the licensee's review. The inspectors found that the licensee apparently questioned these nozzles, but had included a review and analysis process with the initial observations. Engineering personnel had performed additional visual examinations, including reviews of tapes from previous outages, to determine that the visual evidence was not from active leakage. However, the inspectors were concerned that this process was not consistent with other ANO visual examination processes, such as the boric acid control program. The head examination did not involve the inclusion of the suspect nozzles into the corrective action program, and the preliminary suspect status of these nozzles was never raised such that management could review them. Upon discussion of these concerns with ANO management, they decided that ANO would adopt a better process to ensure that suspect findings, from subsequent VHP inspections, would be addressed in a more consistent manner with other visual findings. Based upon the followup discussions, and subsequent volumetric examination data, the inspectors concluded that no additional actively leaking nozzles were overlooked.

.3 Volumetric Examinations

a. Inspection Scope

The inspectors verified that the licensee's volumetric inspection plan and critical performance objectives were incorporated into site procedures. They also interviewed plant inspection personnel and contractors performing the inspections, to determine their understanding of NRC Bulletins 2001-01, 2002-01, and 2002-02, as well as the specific inspection plan. The inspectors observed the data collection and analysis for seven nozzles each, of both the Westinghouse method and the Framatome method.

Due to the complex data gathering and technical analyses involved in the UT/ET examinations of the VHP nozzles, the inspectors were assisted in this inspection by an NDE specialist from Pacific Northwest National Laboratory.

b. Findings

Specific details of the data gathering and analysis methods, used by Westinghouse and Framatome, were considered proprietary. Therefore, details of techniques, data, and analyses are not discussed in this report.

The inspectors determined that the contractor personnel were able to gather and analyze UT/ET data with considerable confidence. However, the inspectors noted that the examinations were limited to the ID of the nozzle, the nozzle base material, and the interface area of the nozzle outside diameter (OD) and j-grove weld. The inspectors were informed that, at present, no technique was available to accurately interrogate and analyze UT/ET data for the volume of the j-groove welds themselves on a Babcock and Wilcox vessel head. This meant that flaws, and potential leak paths, could not be determined inside the weld material of this vessel head. Therefore, flaws initiating from the wetted surface of the weld can not presently be detected by volumetric examination on Unit 1 until they impact the nozzle material, or go through-wall and become visually evident by boundary leakage up through the VHP nozzle annulus. The inspectors and NRC management discussed this vulnerability with ANO personnel. They also informed ANO management that subsequent Bulletin 2002-02 and TI 2515/150 inspections would require a 100 percent volumetric or dye penetrant examination of the j-grove weld wetted surfaces.

Based upon the VHP nozzle examinations, eight nozzles were identified as having indications that required repair during this outage. These indications varied in number per nozzle, origination points, depth, and length. Specific characterizations of the flaws were reported to the NRC in the licensee's 30-day postoutage written response (ADAMS

Document ML023510241). The inspectors reviewed this document and concluded that the findings were accurately described.

.4 Nozzle Repairs

a. Inspection Scope

The inspectors reviewed the repair techniques for each of the eight VHP nozzles repaired during this outage. Weld repair techniques were also presented to NRC staff at a meeting in Rockville, Maryland on October 16, 2002. The inspectors verified that each of the repairs were planned and implemented in accordance with ASME Code requirements, or previously approved NRC exemption requests. The inspectors also observed postmaintenance UT/ET examinations of the repaired nozzles.

b. Findings

Nozzles 54 and 68 were repaired by the Westinghouse method which performed a complete overlay of the j-grove weld, from the stainless steel cladding to the OD of the nozzle. Nozzles 3, 6, 15, 33, and 56 were repaired using the Framatome repair process. This process severed the nozzle at about mid-bore of the reactor vessel head, and removed that lower portion of the nozzle. A temper-bead weld was then applied to the bottom of the nozzle and the reactor vessel head. The inspectors noted that this left a portion of uncladded carbon steel reactor vessel head in contact with the RCS. The NRC and industry analyses have shown that this is acceptable due to the very low boric acid corrosion rates without exposure to an oxygen environment.

Nozzle 56 was visually identified in April 2001, during Refueling Outage 1R16, as having through-wall RCS pressure boundary leakage. At that time, the licensee also conducted UT/ET examination of the nozzle. A circumferential crack, on the OD of the nozzle was identified below the j-grove weld. The specific details of the flaw and repair method are described in LER 50-313/2001-002-00 (ADAMS Document ML011350195) and ANO's 30-day response to Bulletin 2001-01 (ADAMS Document ML012550242).

Through December 20, 2002, the inspectors conducted several followup interviews with engineering personnel to understand the failure method and root cause of the repeat leakage of Nozzle 56. The licensee provided written response to the NRC via LER 50-313/2002-003-00 (ADAMS Document ML023400549) and the 30-day poststartup response for Bulletin 2002-02 (ADAMS Document ML023510241). In summary, the licensee performed an embedded flaw repair in accordance with Section XI of the ASME Code. However, the licensee concluded that this repair method was inadequate to prevent recurrence of the original PWSCC. They stated that the partial arc of the excavation and overlay did not adequately seal the termination points of the weld. The licensee also stated that additional recommendations of the ASME Code that were not implemented, such as water conditioning and heat treatment, as they were initially considered unnecessary, may have assisted in preventing recurrence. The inspectors concluded that, during 1R16, the licensee was not adequately sensitive to the fact that RCS pressure boundary leakage needed to be precluded and not just minimized. The inspectors also reviewed the root cause and written response and concluded that they were appropriate.

Appendix B, Criterion XVI, of 10 CFR 50, states that in the case of significant conditions adverse to quality, the licensee shall assure that corrective action taken precludes repetition. Although the licensee determined the RCS pressure boundary leakage from VHP Nozzle 56 was a significant condition adverse to quality, they failed to take adequate corrective actions to preclude repetition. Although the ASME Code repair was intended to last for at least 30 years, the inadequate design of the weld repair for this application of the ASME Code resulted in the repeat failure of the pressure boundary in less than one operating cycle.

The inspectors considered the significance of this issue using the Significance Determination Process of NRC Manual Chapter 0609. The issue was considered greater than minor because the RCS pressure boundary was actually affected. The inspectors also considered, from the visual evidence, that the leak had been ongoing for more than 30 days. Using the guidance of MC 0609, the inspectors concluded that a Phase 2 analysis was required. Due to the complex nature of VHP nozzle cracking, and to have consistency with other PWSCC findings, NRC management directed that a Phase 3 analysis be conducted for this issue. Therefore, until the final characterization of the risk associated with this issue is available, this issue is considered an unresolved item (URI 50-313/2002-05-02).

.5 Review of Institute of Nuclear Power Operations Evaluation Report

The inspectors reviewed for information only the Interim Report issued by the Institute of Nuclear Power Operations for the evaluation of the ANO facility performed in July 2002.

4OA6 Meetings, Including Exit

The inspectors presented the inservice inspection results to Mr. R. Bement, General Manager Plant Operations, and other members of the licensee's management staff at the conclusion of the inspection on October 17, 2002. The licensee acknowledged the findings presented.

The resident inspectors presented the overall results of the inspections to Mr. C. Anderson, Vice President, Operations, and other members of the licensee's management staff on January 3, 2003. The licensee acknowledged the findings presented.

The inspectors noted that while proprietary information was reviewed during the inspections, no proprietary information would be included in this inspection report.

ATTACHMENT
SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

E. Addison, Technical Specialist
C. Anderson, Vice President, Operations
G. Ashley, Licensing Manager
D. Bauman, Supervisor, Project Management
R. Bement, General Manager Plant Operations
M. Cooper, Licensing Specialist
S. Cotton, Director, Nuclear Safety Assurance
B. Daiber, Supervisor, Systems Engineering
B. Eichenberger, Manager, Unit 1 Operations
N. Eggemeyer, Manager, Technical Support
C. Eubanks, General Manager Plant Operations
M. Farmer, Assistant Manager, Unit 1 Operations
F. Forrest, Acting Manager, Maintenance
D. Fouts, Supervisor, Safety Analysis
R. Fuller, Senior Specialist, Corporate Assessments
B. Greeson, Supervisor, Code Programs
R. Hathaway, Assistant Manager, Unit 2 Operations
D. Hawkins, Licensing Specialist
A. Heflin, Acting Manager, Planning and Scheduling and Outages
R. Henry, Manager, Information Technology
J. Hoffpauir, Plant Manager, Operations
D. James, Manager, Engineering Programs and Components
K. Jones, Unit 1 Control Room Supervisor
J. Keys, Unit 1 Shift Manager
J. Kowalewski, Director, Engineering
J. McWilliams, Manager, Project Management
T. Mitchell, Manager, Unit 2 Operations
K. Nichols, Manager, Design Engineering
M. Paterak, Quality Implementation
G. Parks, Supervisor, Quality
L. Schwartz, Manager, Nuclear Engineering
W. Sims, Design Engineering
C. Tyrone, Manager, Quality

ITEMS OPENED

50-313/2002-05-01	URI	Failure to provide accurate and complete information in response to an NRC Generic Letter regarding a Unit 1 primary water chemistry sulfate excursion (Section 4OA2)
50-313/2002-05-02	URI	Failure to prevent recurrence of RCS pressure boundary leakage (Section 4OA5)

DOCUMENTS REVIEWED

In addition to the documents called out in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Condition Reports

CR-ANO-1-2001-00938	CR-ANO-1-2002-01359
CR-ANO-2-2002-00923	CR-ANO-1-2002-01413

Procedures

NUMBER	TITLE	REVISION
2311.009	ANO Unit 1 and Unit 2 Alloy 600 Inspection	001-00-0
5120.500	Steam Generator Integrity Program Implementation	009-05-0
5120.509	Steam Generator Inservice Inspection Implementation Program	001-02-0
5120.518	ANO Steam Generator Testing and Repair	000-05-0
HES-27	ANO-1 Steam Generator ECT Examination Guidelines	08
MRS-SSP-1267	Reactor Vessel Head Penetration Repair at Waterford 3, ANO 2, and ANO 1	2
MRS-SSP-1282	Reactor Vessel Head Penetration Inspection Tool Operation for Waterford 3/ANO 2/ANO 1	2
WCAL-002	Pulser/Receiver Linearity Procedure	1
WDI-ER-004	Intraspect Eddy Current Analysis Guidelines Inspection of Reactor Vessel Head Penetrations	1, FC 01
WDI-ET-008	Intraspect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations with Gap Scanner	0, FC 01
WDI-UT-010	Intraspect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic & Longitudinal Wave & Shear Wave	3, FC 01
WDI-UT-013	CRDM/ICI UT Analysis Guidelines	1, FC 01

Procedures

NUMBER	TITLE	REVISION
NDE 9.04	Ultrasonic Examination of Ferritic Piping Welds (ASME Section XI)	2
NDE 9.23	Ultrasonic Examination of Austenitic Piping Welds (ASME Section XI)	2
NDE 9.30	Magnetic Particle Examination (MT)	1
NDE 9.31	Magnetic Particle Examination (MT) for ASME Section XI	1
1415.022	Ultrasonic Examination of Dissimilar Metal Welds	005-00-0
1415.038	Manual Ultrasonic Examination of Pressure Vessel Welds	007-00-0