ENCLOSURE 3

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-313 50-368
License Nos.:	DPR-51 NPF-6
Report No.:	50-313/97-13 50-368/97-13
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	May 12-June 5, 1997
Inspectors:	Lawrence E. Ellershaw, Reactor Inspector, Maintenance Branch William M. McNeill, Reactor Inspector, Maintenance Branch Yun-Seng Huang, Senior Mechanical Engineer, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation
Approved By:	Dr. Dale A. Powers, Chief, Maintenance Branch Division of Reactor Safety

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/97-13; 50-368/97-13

This inspection consisted of a review of the licensee's implementation of its inservice inspection program, and followup to unresolved items regarding inservice testing issues. The inspection report covers a 2-week period onsite, with followup in the office by a region-based inspector.

Maintenance

- The installation of eddy current testing robotics for steam generator tubing and the inservice inspection were performed very well (Section M1).

- The inspectors identified a deviation wherein the licensee failed to meet commitments regarding testing of high pressure injection/makeup nozzles (Section M8.1).

- A weakness in the condition reporting and corrective action procedure was identified, in that, it did not require positive verification of completion from the responsible personnel prior to closing a condition report (Section M8.3).

- The inspectors identified a violation of 10 CFR 50.55a and the ASME Code regarding a failure to include required valves in the inservice test program, and a failure to test or exercise valves that were included in the inservice test program to verify their ability to fulfill their intended safety functions (Section M8.5).

Report Details

Summary of Plant Status

Unit 1 was operating in Mode 1, and Unit 2 was in a refueling outage for the entire inspection period.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Inservice Inspection (73753)

a. Inspection Scope

The inspectors observed nondestructive examinations on the following welds and supports.

- Liquid Penetrant Examination - Exam 79-068W (4 lugs) Integrally Welded Attachments for Support 2HCB-27-H5 on Containment Spray Line 2HCB-27-20.
- Automated Ultrasonic Examination - Exam 17-001 Circumferential Pipe Weld of Feedwater Loop Line 12DBB-1-24 to the Transition Piece of the Steam Generator Feedwater Nozzle.
- Magnetic Particle Examination - Exam 19-040W (4 lugs) Integrally Welded Attachments for Support 2DBB-2-H14 on Feedwater Line 2DBB-2-2.

The inspectors observed the installation of the "Genesis Manipulators" robotics used for the eddy current testing of the steam generator tubes. This included verification of manipulator arm position.

b. Observations and Findings

The inspectors found the observed examinations were performed in accordance with the applicable procedures and ASME Code requirements. The examination personnel noted a limitation during the liquid penetrant examination in which the last 1/4-inch of the welds could not be inspected because of interference from a pipe clamp. The examiners also noted a curvilinear indication during the liquid penetrant examination of Lug 1 which was appropriately dispositioned in accordance with the procedure.

The inspectors witnessed the calibration of the ultrasonic equipment and verified the linearity checks. The inspectors verified the use of proper search units, calibration block, and testing materials. The inspectors also verified that proper examination coverage was accomplished during the ultrasonic examination.

The inspectors observed that contractor installation of the eddy current testing robotics for Steam Generator A cold and hot legs was in accordance with the procedures. The inspectors witnessed the position verification activities.

c. Conclusions

The inservice inspections and installation of the eddy current robotics for steam generator tube testing were performed in accordance with the applicable procedures.

M3 Maintenance Procedures and Documentation (73753)

The inservice inspection records were in accordance with licensee program, procedure, and ASME Code requirements. The inspectors observed, however, that the baseline liquid penetrant examination report for Support 2HCB-27-H5 on Containment Spray Line 2HCB-27-20. did not identify the limitation or the curvilinear indication that was observed during this examination. The inspectors considered the failure to identify the limitation during the baseline examination to be a lack of attention to detail. This was discussed with the licensee nondestructive examination supervisor who considered this to be a minor and isolated condition. Without further observations to the contrary, the inspectors agreed with the supervisor's position.

M4 Maintenance Staff Knowledge and Performance (73753)

The inspectors reviewed the qualification records of the personnel observed performing the examinations and found them to be appropriate. The inspectors concluded that the inservice inspection personnel were knowledgeable and that their performance was good.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 Followup on Industry Event at Oconee Unit 2: Unisolable pressure boundary leak.

On April 21, 1997, Duke Power Company, the licensee for Oconee 2, identified leakage in a high pressure injection nozzle. The leakage appeared to be the result of high-cycle, low-stress, thermal fatigue with flow-induced vibration as a likely contributor. The unisolable pressure boundary leak was a precursor to a small break loss-of-coolant-accident. This failure mechanism was identified in 1982 at Crystal River, Unit 3, which experienced an unexplained loss-of-coolant on January 24, 1982. Subsequently, it was revealed that the high pressure injection/makeup nozzles were cracked. The NRC issued Information Notice 82-05. The licensees who were users of Babcock & Wilcox nuclear steam supply systems formed a "B&W Owners' Group Safe-End Task Force," that established a root cause and made recommendations to address the problem. This problem became Generic Issue 69 and the NRC issued Generic Letter 85-20, which endorsed the Owners' Group recommendations.

The licensee committed to perform Recommendation 3, which was then added to the inservice inspection program augmented testing requirements for the injection nozzles. By letter dated April 22, 1985, the licensee informed the NRC of its agreement to implement the recommendations of the B&W Owners' Group. The licensee had, in fact, already initiated implementation of the augmented testing during the Unit 1 forced outage in 1982.

Recommendation 3 addressed the following nozzle conditions and the associated examination schedule:

> Unrepaired nozzles were to be examined by radiography and ultrasonics during each of the next five refueling outages, then every fifth refueling outage thereafter;
>
> Nozzles with the new sleeve design were to be similarly examined during the first, third, and fifth refueling outages, then every fifth refueling outage thereafter; and
>
> Nozzles that were re-rolled were to be examined by radiography during each of the next five refueling outages, then every fifth refueling outage thereafter.

The radiographic testing was to ensure no gap existed between the thermal sleeve and the safe end. The ultrasonic testing was to detect cracking of the safe-end and the adjacent pipe.

Unit 1 has four high pressure injection nozzles (Nozzles A through D) with one nozzle also providing makeup flows (Nozzle D). Each nozzle has a thermal sleeve within the nozzle and safe-end. The original Babcock & Wilcox design was for a safe-end to be welded to the injection pipe. A thermal sleeve was "lightly" rolled into the inside diameter of the safe-end to minimize thermal transients. The thermal sleeve extended beyond the nozzle into the reactor coolant loop. The safe-end was welded to the nozzle. During the Unit 1 forced outage in 1982, Nozzles A and D were replaced with the new sleeve design, which had a "hard" roll, and had pins installed to secure the thermal sleeve. Nozzle B was re-rolled, and Nozzle C was unrepaired.

During this inspection, the inspectors verified that the licensee had included the high pressure injection/makeup nozzles in its augmented inservice inspection program at the examination frequency stated above. The inspectors learned that in 1989, the licensee had identified that certain planned examinations had been missed. This was documented in Condition Report 1-89-0508. The examinations were identified in the inservice inspection program as augmented examinations, and not identified as being associated with commitments to the NRC. As a result of the categorization of the examinations, subsequent work schedule or ALARA demands led to the examinations being cancelled.

The inspection history of the nozzles is as follows. During the forced outage of April 1982, all four nozzles were examined by radiography. At that time, the new sleeve design was installed on Nozzles A and D, and Nozzle B was re-rolled. Approximately one year later, during Refueling Outage 5 (February 1983), Nozzles A, B, and D were ultrasonically examined. Nozzle C was not examined. The ultrasonic examinations were of the safe-end to pipe welds only, and did not include the safe-end to nozzle welds. These two efforts were considered to be the baseline or first test. During Refueling Outage 6 (January 1985), the thermal sleeve of Nozzle C was radiographically examined. During Refueling Outage 7 (November 1986), the safe-end-to-nozzle weld in Nozzle A was radiographically examined.

The corrective actions identified in Condition Report 1-89-0508 resulted in the performance of radiographic and ultrasonic examinations during Refueling Outage 9. Licensee personnel also performed an ultrasonic examination of Nozzle D during Refueling Outage 12 in February 1995. The licensee has established plans to perform radiographic and ultrasonic examinations of all nozzles during the spring of 1998 (Refueling Outage 14).

However, between conduct of the initial committed radiographic and ultrasonic examinations during Refueling Outage 5 in February 1982, and Refueling Outage 9 in November 1990, 12 of the 14 scheduled examinations were not performed. The licensee failed to meet the commitments made to the NRC in the April 22, 1985, letter (1CAN048501) and the NRC was not informed of a change to the commitment. This failure to implement a commitment was identified as a Deviation (50-313/9713-02).

M8.2 <u>(Closed) Inspection Followup Item 50-313/9511-01</u>: monitoring and quantifying of leakage through the refueling cavity liner plate because of weld cracks.

During this inspection, the inspectors verified that the licensee had established procedural requirements to identify if water was leaking from the fuel transfer canal leak detection system. Those new requirements were contained in Procedure 1102.015, "Filling and Draining the Fuel Transfer Canal," Revision 17, and Procedure 2102.015, "Filling and Draining the Fuel Transfer Canal," Revision 9. No leakage had been identified and no problems were identified.

M8.3 (Closed) Unresolved Item 50-313;-368/9604-01: Possible premature or inappropriate closure of a condition report and certain of its action items.

This item, which was identified during review of licensee actions associated with inservice testing and backleakage issues, was comprised of four examples in which it appeared that all actions may not have been completed prior to closure of the condition report and its action items.

The inspectors reviewed each of the identified action items and determined that they had been appropriately closed. In each case, all specified actions had been completed. The condition report had also been appropriately closed in accordance with Procedure 1000.104, "Condition Reporting and Corrective Actions," Revision 11.

The inspectors did identify a weakness in the procedure, in that condition report final closure verification was a passive process. Rather than requiring positive affirmation from cognizant individuals that a condition report could be properly closed, a passive process (i.e., not responding to or acknowledging a closure verification request within a specified time) was being used. Thus, no response or acknowledgement could be taken to mean that the identified condition report could be appropriately closed. The passive system did not take into consideration the possibility of a request being lost or misplaced. Licensee personnel agreed to review the closure verification process for possible enhancement to the procedure.

M8.4 (Closed) Unresolved Item 50-313;-368/9604-02: Engineered safety features system leakage surveillance procedures did not require assessment of total leakage to assure that Final Safety Analysis Report allowable values were not exceeded.

Engineered safety feature system leakage surveillance procedures did, however, have either train or component acceptance criteria which were set conservatively low to minimize the possibility of exceeding Final Safety Analysis Report total system leakage allowable values. Further, the procedures required initiation of a condition report if an individual component leak rate criterion was exceeded.

Licensee personnel evaluated the condition and initiated action items to revise system leakage surveillance procedures to incorporate the total leakage criteria specified in the Final Safety Analysis Report. This is designed to identify and prevent the total leak rate from exceeding any total limit specified in the Final Safety Analysis Report. System engineering personnel were now responsible for assuring that total system leakage assessments are performed.

M8.5 (Closed) Unresolved Item 50-313;-368/9604-03: Engineered safety feature/ emergency core cooling system recirculation isolation valves were not being leak rate tested.

In 1991, ABB-Combustion Engineering, Inc., issued Info-Bulletin 91-03, "Unanalyzed Potential Release Path Through Safety Injection Refueling Water Tank," and NRC issued Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere." These documents were issued to inform utilities that leakage through engineered safety feature/emergency core cooling system recirculation boundary valves could result in a potential unmonitored and unfiltered radioactive materials release path from the containment sump to the atmosphere during the recirculation phase following a loss-of-coolant accident. Further, the information notice indicated that the recirculation boundary valves may not be identified in licensee inservice testing programs as ASME Code, Section XI, Category A valves requiring leak rate testing.

The inspectors reviewed the applicable sections of the Operating Licenses and Final Safety Analysis Reports for Units 1 and 2 to determine the bases for not considering the recirculation boundary valves as Category A valves. For Unit 1, external emergency core cooling system leakage was considered to be a part of the design/licensing basis, and was addressed in Final Safety Analysis Report, Section 6.4, "Engineered Safeguards and Radiation Leakage Considerations." Section 6.4.3 states, "With the exception of the boundary valve discs, all of the potential leakage paths are examined during periodic tests or normal operations." The potential external leakage sources are identified as valves, flanges, and pump seals, with boundary valve disc leakage assumed to be retained in other closed systems and not released to the auxiliary building. Section 6.4.3 also stated that for those paths from the emergency core cooling system that contain dual isolation valves, the closed system definition is met since the first isolation valve serves as the interfacing system isolation valve and the second isolation valve provides closure. Even though the Unit 1 Final Safety Analysis Report considered boundary valve seat leakage to remain in the interfacing system, the total leakage estimate shown in Final Safety Analysis Report Table 6.11 included boundary valve seats. The assumed values were an estimate of leakage, and were not intended to provide operational or testing requirements.

With respect to Unit 2, engineered safety feature leakage was discussed in Chapter 15 of the Final Safety Analysis Report. As with Unit 1, valves, flanges, and pump seals were considered for contribution to the total engineered safety feature leakage, and were identified in Table 15.1.13-5. The offsite dose for engineered safety feature leakage assumed the total leakage to be released into the engineered safety feature pump rooms. There was no discussion or consideration regarding valve seat leakage to other systems. The Final Safety Analysis Report did not address any of the various types of leakage paths listed in Table 15.1.13-5 on an individual basis. Since all leakage was assumed to contribute to the offsite dose, the Unit 2 analysis was considered to be bounding.

The inspectors concluded that the licensing and design basis for both units appeared to have dismissed containment sump water leakage from associated boundary valves as a potential leakage path. Therefore, there was no requirement for classifying the emergency core cooling system/engineered safety feature boundary valves as ASME Code, Section XI, Category A valves, for which testing and acceptance criteria would be necessary.

On May 19, 1997, the inspectors noted that Check Valves BW-4A and BW-4B (Unit 1 Borated Water Storage Tank Outlet Check Valves) were included in the inservice test program; however, they were identified as having an open safety function only. Check Valves 2BS-1A and 2BS-1B (Unit 2 Refueling Water Tank Outlet Check Valves) were similarly identified. These valves were the first isolation valves in paths from the emergency core cooling system/engineered safety feature that contained second isolation valves; therefore, the licensee considered this arrangement as meeting the closed system definition (i.e., dual isolation valves). However, the valves were not identified as having a closed safety function and were not being tested in the closed position. The inspectors considered that closure of the valves could not be taken credit for, therefore, the configuration did not meet the definition of a closed system.

The licensee responded by initiating Condition Report CR-1-97-0145 on May 19, 1997. The condition report noted that Valves BW-4A and BW-4B had been tested in the closed position until 1993, when that safety function was removed. The condition report recommended that closure testing be reestablished for these valves, and that other valves in the emergency core cooling system be evaluated to determine if similar conditions exist. On May 19 and 20, 1997, Valves BW-4A and BW-4B were successfully tested to demonstrate their ability to close, as required. Further review by licensee personnel revealed four additional check valves that were identified as not having a closed safety function, yet were considered part of a dual isolation configuration (CA-61, CA-62 - sodium hydroxide tank outlet check valves, and BW-2, BW-3 - high pressure injection pump suction check valves). Licensee personnel conducted an operability assessment on these valves and determined them to be operable based on recent surveillance test information and periodic maintenance. The inspectors reviewed and agreed with the assessment.

Licensee personnel also initiated Condition Report CR-2-97-0229 on May 21, 1997, to perform a similar evaluation of Unit 2 engineered safety feature system boundary valves. Since Unit 2 was in a refueling outage, completion of the Unit 2 evaluation was identified by the licensee as a startup restraint. On May 23, 1997, licensee personnel performed nonintrusive testing on Valves 2BS-1A and 2BS-1B. The results showed that both valves stroked full open and full closed, thus, demonstrating their ability to meet all safety functions.

Article IWV-3000 in Section XI of the ASME Code specifies the type of tests to be performed on each category of valve, and Subarticle IWV-3412(a) states that valves are to be exercised to the position required to fulfill their function (i.e., open or closed).

The licensee's failure to test or exercise the above eight valves to verify their ability to fulfill all safety functions, constitutes a violation of 10 CFR 50.55a(f)(4) and Section XI of the ASME Boiler and Pressure Vessel Code (50-313;-368/9713-01).

Further licensee review identified an additional seven Unit 2, ASME Code, safety-related, normally closed valves that have an open safety function, but were not in the inservice test program. The valves were identified as 2FP-31, 2FP-46, 2SW-138, 2SW-56, 2SW-57, 2SW-62, and 2SW-67, all Category B valves in the service water piping which provides makeup water to the spent fuel pool. Licensee personnel performed an operability assessment on these valves and determined that they were operable based on recently performed surveillance tests on other equipment which required opening of the seven valves.

Article IWV-1100 of the ASME Code provides the rules and requirements for inservice testing to assess operational readiness of certain ASME Code Class 1, 2, and 3, valves which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of an accident. The licensee's failure to include ASME Code, safety-related valves in the inservice test program is an additional example of a violation of 10 CFR 50.55a(f)(4), and Section XI of the ASME Boiler and Pressure Vessel Code (50-313;-368/9713-01).

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.2 Review of Final Safety Analysis Report Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Final Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the Final Safety Analysis Report description. While performing the inspections discussed in this report, the inspectors reviewed the applicable sections of the Final Safety Analysis Report that related to the areas inspected.

During discussions with licensee personnel regarding emergency core cooling system/engineered safety feature leakage paths and potential contributing sources to offsite dose limits, the inspectors were informed of the existence of Action Items 25 and 26 in Condition Report C-96-0135. These action items (for Unit 1 and 2, respectively), dated January 9, 1997, were initiated to address discrepancies identified between the Final Safety Analysis Reports and actual plant configurations.

The action items stated that Table 6-11 of the Unit 1 Final Safety Analysis Report and Table 15.1.13-5 of the Unit 2 Final Safety Analysis Report, respectively, contained a listing of the components that were assumed to provide a leakage path to the auxiliary building during the recirculation mode following a loss-of-coolant accident. It was identified that the accuracy of the listed boundary valves listed in each of the tables had come into question and the basis could not be found. Tables 6-11 and 15.1.13-5 showed a total of 78 and 71 boundary valves, respectively, as opposed to the actual configurations of 126 and 114 boundary valves, respectively.

Licensee personnel documented their evaluation of the differences between the Unit 2 configuration and the Final Safety Analysis Report in Engineering Report 97-R-2002-01 dated April 15, 1997. A 10 CFR 50.59 review was completed, reviewed, and accepted by the Plant Safety Committee on May 12, 1997. The review concluded that a change to Table 15.1.13-5 was required in terms of identifying the correct number of boundary valves, but the new analysis did not affect the total leakage to the auxiliary building. No plant changes or input changes to existing dose calculations were required, and the probability of an accident previously evaluated in the Final Safety Analysis Report was not increased. At the same time, a licensing document change request was initiated to submit the Final Safety Analysis Report change to the NRC.

A similar evaluation for Unit 1 was documented in Engineering Report 97-R-1002-01, dated February 21, 1997. The 10 CFR 50.59 review had not been completed at the close of this inspection, as it was awaiting finalization of additional supporting information.

The inspectors identified review of the pending 10 CFR 50.59 evaluation as an inspection followup item (50-313/9713-03).

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the onsite portion of the inspection on June 5, 1997. The licensee personnel acknowledged the findings presented.

The inspector asked the licensee personnel whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Ashley, Licensing Supervisor
T. Brown, Outage Manager, Unit 1
T. Chilcoat, Senior Oversight Specialist, Corporate
A. Clinkingbeard, Shift Supervisor, Operations, Unit 1
M. Cooper, Licensing Specialist
D. Denton, Director, Support
R. Edington, General Manager
D. Graham, Engineering Programs Supervisor
R. Harris, Nuclear Engineering Supervisor
J. Howell, Design Engineer
R. Lane, Director, Design Engineering
R. McWilliams, Inservice Test Engineer
S. Pyle, Licensing Specialist
J. Souto, System Engineer, Unit 1
G. Woerner, Design Engineering Supervisor

NRC

K. Kennedy, Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 73753	Inservice Inspection
IP 92902	Followup - Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-313,-368/9713-01	VIO	failure to include certain ASME Code, safety-related valves in inservice test program, and failure to appropriately test certain valves in their safety function position (Section M8.5)
50-313/9713-02	DEV	failure to meet commitments regarding inservice inspection frequency with no subsequent notification made to NRC (Section M8.1)

50-313/9713-03	IFI	review and assessment of 10 CFR 50.59 evaluation, including supporting documentation regarding discrepancy between Final Safety Analysis Report and plant configuration (Section E2.2)

<u>Closed</u>

50-313/9511-01	IFI	monitoring and quantifying of leakage through the refueling cavity liner plate because of weld cracks (Section M8.2)
50-313;368/9604-01	URI	possible premature or inappropriate closure of a condition report and certain of its action items (Section M8.3)
50-313;368/9604-02	URI	engineered safety features system leakage surveillance procedures did not require assessment of total leakage to assure that Final Safety Analysis Report (allowable values were not exceeded (Section M8.4)
50-313;368/9604-03	URI	engineered safety feature/emergency core cooling system recirculation isolation valves were not being leak rate tested (Section M8.5)

LIST OF DOCUMENTS REVIEWED

Inservice Inspection Plan Arkansas Nuclear One Unit 1, Revision 31

Inservice Inspection Plan Arkansas Nuclear One Unit 2, Revision 4

Procedure 1415.004, "Liquid Penetrant Examination - ASME Section XI," Revision 3

Procedure 1415.012, "Magnetic Particle Examination - ASME Section XI," Revision 5

Procedure 1415.045, "Automated Ultrasonic P-Scan Examination of Piping," Revision 1

Procedure STD-NSS-074, "Remote Installation and Removal of ABB/Combustion Engineering Genesis Manipulators," Revision 7

Procedure STD-NSS-078, "Setup, Checkout, and Operation of ABB/Combustion Engineering Genesis Manipulators," Revision 7

Work Plan 2409.551, "Steam Generator Eddy Current Testing," Revision 0

Procedure 1102.015, "Filling and Draining the Fuel Transfer Canal," Revision 17

Procedure 2102.015, "Filling and Draining the Fuel Transfer Canal," Revision 9

Procedure 1000.003, "Station Commitment Tracking," Revision 11

Procedure 1062.009 "Commitment Management System (CMS)," Revision 3

Procedure 1000.104, "Condition Reporting and Corrective Actions," Revision 11

Engineering Standard HES-17, "ANO-1 IST Program Bases Document," Revision 1

Engineering Standard HES-18, "ANO-2 IST Program Bases Document," Revision 2

ANO-1 Operating License

ANO-2 Operating License

ANO-1 Final Safety Analysis Report

ANO-2 Final Safety Analysis Report

Engineering Report 97-R-1002-01, "ECCS Leakage Quantities to the Auxiliary Building," Revision 0

Engineering Report 97-R-0001-01, "ECCS Leakage SAR Clarification," Revision 0

Engineering Report 97-R-2002-01, "ESF Leakage Quantities to the Auxiliary Building," Revision 0

10 CFR 50.59 Safety Evaluation, "Leakage Quantities to Auxiliary Building," dated May 12, 1997