APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-313/92-10 Licenses: DPR-51
50-368/92-10 NPF-6

Dockets: 50-313
50-368

Licensee: Entergy Operations, Inc.
Route 3, Box 137G
Russellville, Arkansas 72801

Facility Name: Arkansas Nuclear One (ANO), Units 1 and 2

Inspection At: ANO Site, Russellville, Arkansas

Inspection Conducted: June 21 through August 1, 1992

Inspectors: L. J. Smith, Senior Resident Inspector
Project Section A, Division of Reactor Projects

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Project Section A, Division of Reactor Projects

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Approved: W D Johnson 8/31/92
William D. Johnson, Chief, Project Section A, Division of Reactor Projects Date

Inspection Summary

Inspection Conducted June 21 through August 1, 1992 (Report 50-313/92-10; 50-368/92-10)

Areas Inspected: This routine resident inspection addressed items of regional interest, onsite followup of licensee event reports, followup on previous inspection findings, followup on corrective actions for violations, monthly maintenance observation, bimonthly surveillance observation, operational safety verification, licensee evaluation of proposed changes to the environs around licensed reactor facilities, and preparation for refueling activities.

Results: No violations or deviations were identified.

The decision to measure the leakage across Low Pressure Injection Crossover Check Valve DH-17 rather than just venting when the valve was determined to be off its seat was appropriate. It showed a safety first consciousness. The decision to declare Valve DH-17 inoperable as a result of the calculated leak rate exceeding the Technical Specification limit was correct. The licensee effectively developed a safety evaluation and compensatory measures in support of their request for a temporary waiver of compliance from Technical Specification 3.1.6.9. The special work plan to crack open Valve CV-1401 was well written and very carefully implemented. (Section 6.6)

The licensee carefully evaluated options available for measuring specific activity in the reactor coolant system as required by Technical Specification 4.4.8 when the normal test methods were unavailable due to equipment malfunction. The best alternative was selected. The inspector reviewed the procedures and found the licensee's approach to be technically reasonable. The measurements used were calibrated in the range of interest. (Section 6.7)

Provisions to inspect nonsafety-related spring cans for accumulation of debris and provisions to remove the debris were included in both Units 1 and 2 spring can inspection procedures. Program implementation was verified by randomly selecting accessible safety-related and nonsafety-related spring cans located in both the auxiliary and turbine buildings. Very minimal amounts of debris were still found in the spring cans but not of significant quantity to impact the operability of the spring cans. (Section 3.1.2)

Except for several radiological practice weaknesses, several maintenance and test activities were performed well.

The two operator rounds observed were performed well on both units. (Sections 6.2 and 6.3)

The new program to maintain foreign material exclusion and cleanliness of the emergency diesel generator fuel oil system was implemented. (Section 4.1)

Fuel was successfully transferred to the spent fuel pool in a well coordinated and expeditious fashion. Health physics coverage and Level 1 housekeeping was observed to be in accordance with plant policy. (Section 7)

The effect of 161 Kilovolt capacitor bank availability was evaluated when determining operability of the offsite power sources. (Section 6.1)

Bases information was used correctly to arrive at a conservative implementation of Technical Specifications during repair of Unit 1 pressurizer heaters. (Section 6.9)

The reactor coolant system hot leg temperature reduction was carefully evaluated prior to performance. The evolution was well planned. The crew

briefing was excellent. The test was performed successfully. All desired information was obtained and an error in the computer model of the secondary system was uncovered. (Section 6.4)

During the Unit 2 monthly main turbine control valve stroke test, Turbine Bypass Valve 2CV-0303 failed shut and ultimately led to the release of reactor coolant to the auxiliary building. Followup inspection of corrective actions taken in response to this event will tracked as Inspector Followup Item 50-368/9010-01. Apparent work control weaknesses in a nonsafety-related system will be included in this review.

Several weak radiological practices were observed during surveillance testing and during routine operator rounds.

DETAILS

1. PERSONS CONTACTED

B. Fenech, Director, Nuclear Operations
*S. Boncheff, Licensing Specialist
*J. Taylor-Brown, Manager, Quality Engineering & Control
S. Cecil, Unit 2 Shift Superintendent
*M. Chisum, Unit 2 Assistant Operations Supervisor
M. Cooper, Licensing Specialist
*S. Cotton, Manager, Radiation Protection/Radiation Waste
R. Douet, Unit 1 Maintenance Manager
R. Edington, Unit 2 Plant Manager
*J. Fisicaro, Licensing Director
*R. King, Plant Licensing Supervisor
*R. Sessoms, Central Plant Manager
*J. Vandergrift, Unit 1 Plant Manager
*C. Warren, Unit 2 Maintenance Manager
C. Zimmerman, Unit 1 Operations Manager

*Present at exit interview conducted on August 4, 1992.

The inspectors also contacted other plant personnel, including operators, engineers, technicians, and administrative personnel.

2. PLANT STATUS

2.1 Unit 1

Unit 1 began the inspection period at 100 percent power.

On July 10, power was decreased to 98 percent to perform turbine generator throttle valve and governor valve testing. The unit returned to 100 percent power the same day.

On July 24, electric power was decreased by 200 MW at the system dispatcher's request and in preparation for a condenser water box tube leak inspection. Reactor power was reduced to 76 percent. On July 25, the unit increased power to 90 percent for turbine generator throttle valve and governor valve testing. On July 26, the unit returned to 100 percent power and remained at full power throughout the remainder of the inspection period.

2.2 Unit 2

Unit 2 began the inspection period at 100 percent power.

On July 24, during the monthly turbine control valve testing, Steam Dump Bypass Control Valve 2CV-0303 failed closed, causing a small reactor power transient. (See Section 6.8) Reactor power was subsequently stabilized at

95 percent. The unit returned to 100 percent power the same day and remained at 100 percent throughout the remainder of the inspection period.

3. FOLLOWUP (92701, 92700 and 92702)

3.1 Items of Regional Interest (92701)

3.1.1 Unit 2 - Refueling Water Tank Minimum Solution Temperatures

Technical Specification 3.1.2.8 was revised in Amendment 82 to increase the minimum boron concentration from 1731 parts per million (ppm) to 2500 ppm with an upper limit of 3000 ppm. The minimum solution temperature was not raised from 40°F. Because Technical Specifications for other vendor types typically specify minimum solution temperatures of 55°F, the boric acid solubility curve was reviewed. Boric acid at a concentration less than 3000 ppm would not plate out at temperatures 40°F or higher. Therefore, raising the boron concentration without raising the minimum solution temperature was acceptable.

3.1.2 Units 1 and 2 - Debris in Spring Cans Observed in NRC Inspection Report 50-313;368/91-08

In NRC Inspection Report 50-313/91-08; 50-368/91-08 accumulation of debris in spring cans located in the turbine building was noted. Spring cans not included in the inservice inspection program were being inspected periodically in accordance with Procedure 2306.022, "Unit 2 Spring Can Pipe Hanger Surveillance," Revision 0. However, the procedure did not specifically address the identification or removal of debris inside spring cans.

The licensee issued Licensing Information Request 92-0120 as a result of the observations noted in NRC Inspection Report 50-313/91-08; 50-368/91-08. The request noted that consideration should be given to the revision of Procedure 1300.013, "Unit 1 Spring Pipe Hanger Preventive Maintenance," and Procedure 2306.022, "Unit 2 Spring Can Pipe Hanger Surveillance," to add a step to check for and clean out debris in spring cans as necessary. Step 8.1.1(a) of both procedures incorporated provisions to inspect each pipe hanger for the presence of debris on the inside of the spring can. The step further instructed individuals to document satisfactory or unsatisfactory spring cans on the attachment sheet. Step 8.1.1(b) instructed the initiation of a condition report and job request to correct unsatisfactory spring cans.

Program implementation was verified by randomly selecting accessible safety-related and nonsafety-related spring cans located in both the auxiliary and turbine buildings. Very minimal amounts of debris were still found in the spring cans but not of significant quantity to impact the operability of the spring cans. The revision to the procedures and subsequent programmatic implementation was determined to be effective.

3.2 Onsite Followup of Licensee Event Reports (LERs) (92700)

3.2.1 (Closed) LER 50-313/90-018: "Procedural Deficiencies Which Resulted in Failure to Perform Adequate Local Leak Rate Test of Containment Airlocks"

During a review of Procedure 1304.020, "Unit 1 Reactor Building Access and Ventilation Leak Rate Testing," which was used to perform local leak rate testing of the Unit 1 containment personnel airlock, the licensee identified two apparent discrepancies regarding performance of the test: (1) a local leak rate test of two pressure gage penetrations was not being performed following gage removal and protective cap installation; and (2) a test of the leak tightness of the equalizing valve was not being performed following removal of a protective cap installed during pressurization of the air lock barrel.

The licensee initiated Condition Report 1-90-767 and performed an evaluation of these discrepancies. The licensee concluded that the local leak rate testing of the airlock barrel was being performed incorrectly and that the test results did not accurately reflect the actual as-left leak tightness of the component. Based on this, the Unit 1 airlock was declared to be inoperable. Unit 1 was in cold shutdown condition.

As a result of the discovery of the discrepancies with the Unit 1 containment airlock local leak rate testing procedure, the licensee performed a review of Procedure 2304.022, "Reactor Building Access & Ventilation Leak Rate Testing," which was used to perform local leak rate testing of the Unit 2 containment personnel airlock and emergency escape airlock. A similar discrepancy regarding capping of the inner door equalizing valve prior to performance of the local leak rate testing on the Unit 2 airlocks was identified. The licensee determined that both of these airlocks were also being tested incorrectly. The licensee performed a visual inspection of the airlocks, and there were no pressure gages found that were similar to those installed on Unit 1.

The licensee determined that the root cause of the event was previous organizational weaknesses, which resulted in reliance on personnel who were not adequately trained or knowledgeable in areas such as local leak rate testing methods or the 10 CFR Part 50, Appendix J, requirements for the development and implementation of the testing procedures. As a result, the procedures were approved and implemented without recognition of the deficiency.

The licensee developed and implemented a Unit 1 plant modification which removed the pressure gages and capped the penetrations through the airlock bulkheads. Four pressure gauge penetrations were capped and seal welded. Procedure 1304.020 was revised to delete reference to removal of the pressure gages during testing and to require a separate test of the equalizing valve

after completion of the airlock barrel local leak rate testing. The equalizing valve and airlock barrel were tested satisfactorily prior to establishing containment integrity.

For Unit 2, the licensee declared the inner doors to both the personnel and emergency escape airlocks to be inoperable. The equalizing valve penetrations on both airlock inner doors were capped, and local leak rate tests of the airlock barrels were performed with acceptable results. The caps were left installed on the equalizing valve penetrations, and the licensee declared the inner doors operable. Procedure 2304.022 was revised to include provisions for adequately testing the equalizing valves.

Subsequent to the licensee's corrective actions to address this LER for Unit 1, the licensee implemented Plant Change 91-7039, which replaced the personnel air lock sliding mechanical pressure equalization configuration with a ball valve on both the inner and the outer door. Procedure 2304.022 was revised to delete reference to the equalizing valves.

Based on a review of Condition Report 1-90-767, Condition Report 2-90-543, Procedure 1304.020, Procedure 2304.022, and the review documented in previous NRC Inspection Report 50-313/92-05; 50-368/92-05, this LER was closed.

3.3 Followup on Previous Inspection Findings (92701)

3.3.1 (Closed) Inspector Followup Item 50-368/9002-03: Actions to Resolve the 14,000 gallon per minute (gpm) Flow Limit Imposed on the Service Water Pumps

This inspector followup item involved a review by the licensee to resolve the upper flow limit of 14,000 gpm for the service water pumps.

The licensee completed a review of the design basis of the Unit 2 service water pumps. The licensee determined that the limitation of 14,000 gpm identified in the ANO Unit 2 Safety Analysis Report was correct for worst-case accident operating conditions for the service water pumps while aligned to the emergency cooling pond. The licensee stated the pump should be tested at less than 14,000 gpm because that was the analyzed range; however, the vendor stated pump run out would not occur until approximately 20,000 gpm. The inspector had previously observed the licensee adhere to the 14,000 gpm upper test limit as specified in Procedure 2305.019, "Service Water Pumps Flow Test."

Based on the review of Procedure 2305.019, Revision 3, Plant Change 2, the Unit 2 Safety Analysis Report, and the licensee's engineering response to Reference Document No. 0-NRCM-90-02-06, this item was closed.

3.3.2 (Closed) Inspector Followup Item 50-313/9209-02: Hydraulic Performance Degradation of Service Water Pump P-4A

This inspector followup item identified possible generic implications associated with degraded pump performance as a result of repetitive applications of liquid metal to the pump bowls over consecutive overhauls. The surveillance test of Service Water Pump P-4A performed at ANO indicated that the pump performed approximately 5 to 10 percent below the baseline curve. The licensee speculated that the geometrical configuration of the pump bowls changed as a result of repetitive applications of liquid metal in the bowl for corrosion control. Two selected spare pumps were bench tested. Both spare pumps had applications of liquid metal in the bowl.

The licensee stated that both spare pumps performed better during bench testing than when installed at ANO. One pump performed on the baseline curve and the other performed between the baseline value and the acceptable normal range (5 to 6 percent below the baseline). Based on the data collected during the bench test, no direct correlation of degraded pump performance and the application of liquid metal was identified. The licensee, however, concluded that the pump degradation was probably caused by a synergism of parameters relating to: (1) flow measurement instrumentation errors at ANO; (2) degraded impeller wear rings; (3) the lack of standardization of manufacturer quality control during pump casting for carbon steel pumps; and (4) application of liquid metal to the pump bowls. The licensee added that the application of liquid metal to the pump bowl was probably a minor contributor to degraded pump performance.

The licensee stated that actions to control the parameters would be implemented after each subsequent overhaul. The licensee noted that if the task to control the parameters becomes too labor intensive, other options would be pursued. Applications of liquid metal did not have significant generic implications. This item was closed.

3.4 Followup on Corrective Actions for Violations

3.4.1 (Closed) Violation 50-313;368/9030-02: Failure to Perform an Adequate Safety Review for Conducting Resin Transfer Cask Dewatering Activities

This violation involved the failure to perform an adequate safety review for conducting resin transfer cask dewatering activities. The safety review did not evaluate the consequences of performing the activity in a nonradiologically controlled area (train bay). As a result, contaminated resin fines were released via the turbine building heating, ventilation, and air conditioning system. Areas in the Unit 1 auxiliary and turbine buildings were contaminated with various levels of loose resin fines.

As part of their corrective actions, the licensee initiated Condition Report 1-90-429. The licensee determined the root cause to be a failure to follow operating procedures during performance of the resin drying evolution.

The licensee determined that the system was operated in an abnormal lineup, by securing the dewatering pump intermittently with Valve DW-10 shut. This caused pressurization of the resin container. Pressure in the container vented around the gaskets on the access doors to the upper fill head area. As the moisture content in the resin dropped due to drying, small resin fines were expelled into the surrounding area.

In the letter of response to the violation dated April 7, 1992, the licensee agreed that an inadequate review of the safety analysis for conducting the resin transfer cask dewatering activities had been performed. As part of their corrective actions, the licensee has included in Procedure 1000.131, Revision 0, "10 CFR Part 50.59 Review Program," an action by the 10 CFR Part 50.59 reviewer to recognize, when conducting activities involving processing of radioactive material outside the controlled access area, that a review was required which adequately addresses the potential radiological consequences of performing the activity. Procedure 1000.131 also directs the 10 CFR Part 50.59 reviewer to the proper department in the organization for conducting the evaluation.

Also as part of the licensee's corrective action, Procedure 1012.015, Revision 1, "Radiological Safety Evaluations," was formalized and incorporated. Procedure 1012.015 required that activities involving the processing of radioactive materials outside of the controlled access area would receive a documented review for impact at the site exclusion area boundary. The licensee has also revised Procedure 1612.003, "Radiological Work Permits," to ensure that radiological safety evaluations are performed for the processing of radioactive materials in areas outside of the controlled access areas. The licensee also reviewed the existing work plans and procedures to determine if a radiological safety evaluation was required.

Based on the review of Procedure 1000.131, Procedure 1012.015, Procedure 1612.003, and Condition Report 1-90-429, this violation was closed.

3.4.2 (Closed) Violation 50-368/9130-02: Fire Door Breached Rendering the Fire Barrier Inoperable, Without the Shift Superintendent Being Informed

This violation involved the fire door separating the Emergency Diesel Generator 2K-4A and 2K-4B emergency diesel generator rooms. The fire door was found obstructed by test cables rendering the barrier inoperable. The shift superintendent was not informed, which prevented him from taking compensatory action.

As part of their corrective actions, the licensee initiated Condition Report 1-90-429. The licensee stated that a review of previous condition reports, LERs, and violations did not identify any programmatic problems with fire door breaches. The licensee determined that the root cause of the violation was personnel error and an inadequate prejob briefing stressing the importance of station fire barriers. The violation was reviewed by Unit 2

systems engineering personnel for discussion of lessons learned and system engineering responsibilities as contract coordinators during prejob briefings.

The licensee's general employee training program has been revised to heighten worker awareness of the Fire Barrier Watch Program. The licensee has also enhanced the current labels on all Technical Specification required fire doors to read: "KEEP DOOR CLOSED AT ALL TIMES OR POST A FIRE WATCH, PER PROCEDURE #1000.120."

Based on the review of Condition Report 2-91-0588, Reference Training Document GET-1/1A/1B, Revision 8/26/91, and Job Order (JO) No. 00799737, this item was closed.

4. MONTHLY MAINTENANCE OBSERVATION (62703)

Station maintenance activities for the safety-related systems and components listed below were observed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, and industry codes or standards, and in conformance with the Technical Specifications.

4.1 Unit 1 - Replace Fuel Oil Check Valve FO-82B and Relief Valve on Filter F-45B on Emergency Diesel Generator K-4B (JO No. 873198 and 875254)

On July 6, the licensee entered 7-day Technical Specification 3.7.2.C, for the fuel oil supply system maintenance on Emergency Diesel Generator K-4B. The licensee performed a surveillance test on Emergency Diesel Generator K-4A prior to performing maintenance repairs on Emergency Diesel Generator K-4B to verify operability of at least one diesel generator in accordance with Technical Specification 3.7.2.C. Emergency Diesel Generator K-4A was verified as operable following successful completion of the surveillance test.

The Emergency Diesel Generator K-4B fuel oil system was hold carded for maintenance in accordance with Procedure 1000.27, Revision 16, "Hold and Caution Card Control." The instructions to replace leaking Check Valve FO-82B and the leaking relief valve on Filter F-45B were incorporated on the corrective maintenance JO No. 873198. The existing check valve was bronze, and the replacement check valve was stainless steel. The licensee provided Maintenance Engineering Request 1-92-06-0006-M, which justified the use of a non-"like-for-like" replacement item.

Quality control personnel inspected the replacement items to verify cleanliness. A quality control inspector was present during the installation of the items. The system was restored in accordance with the corrective maintenance instructions. The licensee exited Technical Specification 3.7.2.C the same day. The maintenance activity performed on Emergency Diesel Generator K-4B was efficient and clean. The mechanics implemented the new requirements to ensure foreign material exclusion from the fuel oil system.

The licensee stated that a fitting leak still existed on a fuel oil line which connected the fuel oil header to the injector. Operations established a compensatory measure to prime the fuel oil system every 4 hours to maintain an acceptable level in the sight glass until repairs could be made. On July 28, Emergency Diesel Generator K-4B was removed from service for 13 hours to clean injectors and repair the fitting leak in accordance with JO No. 875254. The repair of the leak was successful.

4.2 Unit 2 - Boric Acid Makeup Pump 2P-39B Disassembly and Maintenance (JO No. 00809657)

On July 7, the inspector observed portions of the disassembly and repair of Boric Acid Makeup Pump 2P-39B. This pump had an excessive amount of boron buildup and had experienced mechanical seal leakage. Vibration analysis had also indicated degradation of the pump's bearings. All components associated with the isolation boundary were hold carded and placed in their proper positions.

The craft was knowledgeable of Procedure 2404.025, Revision 6, "Unit 2 Boric Acid Make-up Pumps (2P-39A & 2P-39B) Maintenance." The craft knew that no Technical Specification limiting conditions for operation had been entered due to the pump being taken out of service. The as low as reasonably achievable (ALARA) practices were appropriate and there were no problems identified.

4.3 Unit 2 - Containment Spray Pump 2P-35B Oil Sample (JO No. 872727)

On July 13, the licensee entered a 72-hour action statement, per Technical Specification 3.6.2.1, to perform maintenance activities on Containment Spray Pump 2P-35B. The inspector witnessed the oil sampling of the pump during the maintenance activities. A health physics technician was present to survey the area 8 feet into the overhead. The mechanics were also observed donning cotton liners and rubber gloves as a precautionary measure. Five hundred (500) milliliters of the oil were drawn from the pump and 500 milliliters of oil were replaced. The replacement oil was the type specified in the JO. The analysis of the oil was performed by the hot chemistry lab, and the results of the testing confirmed that the oil sample was satisfactory.

Upon exiting the overhead area, one mechanic appropriately discarded the protective rubber gloves and the cotton liners. The other mechanic proceeded to the door carrying a ladder. The inspector asked the mechanic if he should exit the room wearing the rubber gloves. The mechanic decided to appropriately discard the protective rubber gloves and the cotton liners. The inspector discussed the observation with the health physics supervisor. The health physics supervisor stated that the area where the work was being performed was surveyed clean and that the mechanics were not required to wear protective rubber gloves. The licensee also stated that, due to previous personnel contamination event trends, shop supervisors were requiring

personnel to wear protective clothing in overhead areas that were surveyed clean. The mechanic did not violate a plant requirement, but he did exhibit a poor work practice.

Personnel involved with the job demonstrated appropriate safety precautions when climbing to the pump motor. During the job, personnel demonstrated appropriate ALARA practices. The oil sample was treated as potentially contaminated. Surveys were performed on the sample and care was taken to ensure that the oil was not spilled. The oil sampling activity was performed well.

4.4 Unit 1 - Preventive Maintenance on H_2/O_2 Waste Gas Analyzer (JO No. 874052)

On July 20, the licensee isolated the H_2/O_2 Waste Gas Analyzer C-119A to perform preventive maintenance on the oxygen sensor. The oxygen sensor was removed, and the existing O-rings and diaphragms were replaced. The sensor was reinstalled, and a surveillance evolution was performed on the H_2/O_2 analyzer. The inspector did not observe the test but was informed that the H_2/O_2 analyzer passed the surveillance.

4.5 Summary of Findings

The oil sampling of Containment Spray Pump 2P-35B was performed well with the exception of one worker that demonstrated a poor work practice. After completing activities in the overhead he failed to remove the gloves he had donned as a precautionary measure. The overhead area had been surveyed clean, but shop foremen were requiring the workers to wear gloves when working in the overhead because of prior personnel contamination event trend data.

No problems were identified during the maintenance and disassembly of the Unit 2 boric acid management pump.

The replacement of O-rings and diaphragms for the Unit 1 H_2/O_2 oxygen sensor was successful. Waste Gas Analyzer C-110A passed the surveillance.

The replacement of the Unit 1 emergency diesel generator fuel oil check valve and relief valve was excellent. Mechanics invoked the plant's new policy for foreign material exclusion and cleanliness of the fuel oil system.

5. BIMONTHLY SURVEILLANCE OBSERVATION (61726)

The inspectors observed the Technical Specification required surveillance testing on the systems and components listed below and verified that testing was performed in accordance with Technical Specifications and the licensee's implementing procedures.

5.1 Unit 1 - Emergency Diesel Generator K-4A Surveillance (JO No. 874045)

On July 6, the inspector observed a portion of the surveillance performed on Emergency Diesel Generator K-4A in accordance with Procedure 1104.36, Revision 30, Supplement 1, "Emergency Diesel Generator Operations." The surveillance was performed to verify operability of Emergency Diesel Generator K-4A as required by Technical Specification 3.7.2.C prior to performing maintenance on Emergency Diesel Generator K-4B. Vibration tests were performed on Emergency Diesel Generator K-4A and the operating parameters were accurately recorded in the operator logs. Appropriate entries were made in the station log when the diesel was declared inoperable. The licensee completed the test and informed the inspector that Emergency Diesel Generator K-4A passed the surveillance and, therefore, was verified as being operable.

5.2 Unit 2 - Surveillance on High Pressure Safety Injection Pump 2P-89C (JO No. 873822)

The inspector observed the performance of a portion of Procedure 2104.039, Revision 28, Supplement 3, "High Pressure Safety Injection System Operation," for High Pressure Safety Injection Pump 2P-89C surveillance on July 10. The surveillance was performed following corrective maintenance to Pump Casing Drain Valve 2ABS-22C and Seal Coolers 2E-53C and 2E-53F. Vibration data was also gathered by predictive maintenance during the surveillance. The instrumentation used to collect the vibration data was calibrated. High Pressure Safety Injection Valve 2CV-5035 was satisfactorily stroke tested and the pump passed the surveillance.

5.3 Unit 1 - Surveillance of the Turbine Driven Emergency Feedwater Pump P-7A (JO No. 874468)

On July 13, the licensee was unable to complete surveillance testing of Emergency Feedwater Pump P-7A, in accordance with Procedure 1106.006, Revision 45, Supplement 2, "Emergency Feedwater Pump Operations," because Steam Bypass Valve SV-2663 failed to close. The licensee stated that the root cause for failure of Steam Bypass Valve SV-2663 to close was still not determined. Because the licensee could not complete the surveillance testing in accordance with the procedure, Turbine Driven Emergency Feedwater Pump P-7A was declared inoperable, and the licensee entered 24-hour Technical Specification 3.4.4 at 4:45 a.m. The licensee initiated Condition Report 1-92-0443 to identify the problem and JO No. 875378 to repair the valve.

Plant Change 3 was requested to change Procedure 1106.006 to preclude the use of Steam Bypass Valve SV-2663 and Steam Admission Valve CV-2663 in the surveillance test if both Valve SV-2663 and Valve CV-2663 were inoperable. This change was based on Turbine Driven Emergency Feedwater Pump P-7A being fully capable of automatic actuation through redundant Steam Bypass Valve SV-2613 and Steam Admission Valve CV-2613. Additionally, both Valves SV-2613 and CV-2613 were powered from the preferred green train. Valves CV-2663 and SV-2663 were powered from the red train. Table 10.5-1 of the Safety Analysis

Report only credits the green train to supply power to Valves SV-2613 and CV-2613 in the event of an accident. Incorporating Plant Change 3 into the procedure was acceptable.

The surveillance on Turbine Driven Emergency Feedwater Pump P-7A was started at 6 p.m., and was successfully completed at 8:23 p.m. Plant Change 3 was adequately incorporated into the procedure and the procedure was properly approved. The control room operators adhered to the steps in the procedure. The Technical Specification was exited at 8:23 p.m. Repair of Steam Bypass Valve SV-2663 was completed at 3 a.m., July 14.

5.4 Emergency Diesel Generator 2K-4B Surveillance (JO No. 874567)

On July 15, the licensee performed a monthly surveillance on Emergency Diesel Generator 2K-4B as required by the surveillance requirements set forth in Technical Specification 3.8.1.1. The inspector observed a portion of the test. The auxiliary operator conscientiously identified the correct valves to be manipulated in accordance with Procedure 2104.036, Revision 34, "Emergency Diesel Generator Operations," Supplement 2. The operating parameters recorded by the auxiliary operator in the surveillance logs were comparable to the values the inspector observed. The licensee appropriately entered the Technical Specification action statement when the diesel generator was declared inoperable. These entries were recorded in the station log and on the status board. The Emergency Diesel Generator 2K-4B surveillance was completed, and the licensee informed the inspector that the diesel satisfactorily passed the surveillance.

5.5 Unit 1 - Surveillance on High Pressure Injection/Makeup Pump P-36A (JO No. 876209)

The licensee performed a surveillance on High Pressure Injection Pump P-36A following repairs to the pump's head gasket on July 23. The surveillance was performed in accordance with Procedure 1104.002, Revision 41, "Makeup Purification System Operation."

Prior to starting the surveillance, a waste control operator checked the pump's oil levels and discovered the bubbler on the inboard bearing oil level was empty. The bubbler was cracked. The waste control operator removed the bubbler and exited High Pressure Injection Pump Room A. The waste control operator reached inside the health physics clothing disposal receptacle and removed a potentially contaminated coverall. As a result, a cloth bootie fell from the receptacle to outside the C-zone. The waste control operator laid the coveralls on the floor inside the C-zone. A waste control operator, wearing rubber gloves outside the C-zone, picked up the bootie and disposed of it in the receptacle. The inspector did not observe health physics notification or a swipe survey of the clean area. The waste control operator in the clean area began pouring oil into the bubbler while the waste control operator in the C-zone held the bubbler over the coveralls. The waste control operator stated that the purpose of the coveralls was to absorb any oil that might spill into the C-zone while filling the bubbler. After the inspector

noted the cloth bootie that fell outside the C-zone, the waste control operator stated that oil absorbing cloths should have been used. The inspector notified the health physics supervisor about the observation of poor work practices. The health physics supervisor agreed that reaching inside a contaminated receptacle and mixing oil with contaminated dirty laundry were bad practices. The supervisor noted that the potentially contaminated bootie falling into the clean area was unacceptable. The inspector concluded that individual radiation worker practices needed improvement.

The inspector observed that the hand held frisker located in the penetration emergency ventilation room on Elevation 335' was not accessible due to a spent resin transfer evolution taking place between the High Pressure Injection Pump Room A and the emergency penetration ventilation room. The area between the two rooms was roped off to prevent individuals from entering the area during the spent resin transfer. The inspector reviewed the radiological work permits associated with the individuals performing the High Pressure Injection Pump P-36A surveillance on Elevation 335. The radiological work permits directed the individuals exiting from a contaminated area to proceed to the nearest frisking station and perform a hand and foot frisk prior to leaving the elevation.

The health physics supervisor stated that not having a frisker available for individuals performing the High Pressure Injection Pump P-36A surveillance was the result of inadequate planning. The health physics supervisor added that a revision to the radiological work permits was incorporated to allow individuals to perform a hand and foot frisk at the nearest available frisker, even if it was on the next elevation.

The high pressure injection pump was started and the temperature was stabilized. Predictive maintenance collected vibration data during pump operation. The vibration instrumentation was properly calibrated. The inspector checked for oil leaks on the new head gasket and there were no leaks identified. The inspector observed the inboard bearing sight glass and noted a layer of foam approximately 1/4-inch thick on the surface of the oil. The licensee stated that the operability of the pump was not impacted because the foam remained on the oil surface. The licensee noted that the addition of foam inhibitor to the oil would aid in diminishing the foam. The licensee noted that the inhibitor would be added to the oil when the inhibitor arrived on site. The surveillance was successfully completed.

5.6 Summary of Findings

The licensee satisfied Technical Specification 3.7.2.C to demonstrate that Emergency Diesel Generator K-4A was operable prior to performing maintenance on the Emergency Diesel Generator K-4B. The data acquisition for the operating parameters was accurately completed in the operator logs during the surveillance. The Emergency Diesel Generator K-4A surveillance test was performed well.

The auxiliary operator performed the Emergency Diesel Generator 2K-4B surveillance in a conscientious fashion and accurately recorded operating parameters. The licensee entered the appropriate Technical Specification when Emergency Diesel Generator 2K-4B was declared inoperable. Emergency Diesel Generator 2K-4B passed the surveillance, and the surveillance test was performed in an excellent manner.

The corrective maintenance on High Pressure Safety Injection Pump 2P-89C for the high pressure safety injection coolers and the casing drain valve was followed by a successful surveillance performance on the pump.

The licensee adequately resolved technicalities in the surveillance procedure to restore Turbine Driven Emergency Feedwater Pump P-7A to operable status in a timely manner. The subsequent surveillance test of Turbine Driven Emergency Feedwater Pump P-7A was performed very well.

The inspector noted weak radiological practices during performance of the High Pressure Injection Pump P-36A surveillance. A waste control operator pulled a potentially contaminated coverall from the hamper for use as an oil absorber and, in the process, he inadvertently knocked a cloth bootie into the clean area. The inspector also observed the existence of foam in the pump's bearing oil supply. The pump successfully passed the surveillance, and the evolution was acceptable.

6. OPERATIONAL SAFETY VERIFICATION (71707)

The inspectors routinely toured the facility during normal and backshift hours to assess general plant and equipment conditions, housekeeping, and adherence to fire protection, security, and radiological control measures. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures and that proper communications with the control room staff had been established.

During tours of the control room, the inspectors verified proper staffing, access control, and operator attentiveness. Technical Specification limiting conditions for operation were evaluated. The inspectors examined status of control room annunciators, various control room logs, and other available licensee documentation.

6.1 Units 1 and 2 - 161 Kilovolt (KV) Switchyard Capacitor Banks

On June 30, at 9:05 p.m., the AP&L system dispatcher notified the licensee that Feeder Breaker B1286 for the 161 KV switchyard capacitor banks had tripped. The licensee initiated Condition Report C-92-0055 to document the operability evaluation for Start Up Transformer No. 2. At the time of the trip, neither unit was aligned to Start Up Transformer No. 2. The 500/161/22 KV autotransformer was available. Based on expected loading, the licensee determined that Start Up Transformer No. 2 would be operable until 11:00 a.m., July 1. Blown fuses were replaced in the capacitors, and the

capacitor bank was returned to service at approximately 2:05 a.m. on July 1. Therefore, no limiting condition for operation was exceeded.

The inspector reviewed the Units 1 and 2 control room logs and Condition Report C-92-0055 to verify proper documentation of the event. The licensee's operability determination was correct.

6.2 Unit 1 - Waste Control Operator Tour of Auxiliary Building

On July 2, the inspector accompanied the waste control operator and his trainee, an auxiliary operator, for a portion of the routine rounds in the auxiliary building. The waste control operator was very knowledgeable and was effectively training the auxiliary operator. The log entries taken by the waste control operator were verified as accurate. No system abnormalities were identified during the tour.

6.3 Unit 2 - Auxiliary Operator Tour for Inside and Outside Rounds

The licensee performed an inside and outside tour of Unit 2 facility for the purpose of auxiliary operator log taking during the reactor coolant system hot leg temperature reduction on July 7 (see Section 6.4). The inspector accompanied the auxiliary operator during this evolution. The major areas toured were the main steam isolation valve room, turbine deck, emergency cooling pond, cooling tower, and service water bay. There were no system abnormalities noted.

6.4 Unit 2 - Reactor Coolant System Hot Leg Temperature Reduction

Special Work Plan 2409.351, Revision 0, "Unit 2 Hot Leg Temperature Reduction," was written to provide instructions for the reduction of Unit 2 reactor coolant system hot leg temperature to less than 600°F, while maintaining reactor power at 100 percent. Normally reactor coolant system hot leg temperature averaged 607°F. The licensee had determined from industry data that intragranular stress corrosion cracking in Inconel 600 steam generator tubes was more severe in plants operating above 600°F than in plants operating with reactor coolant system hot leg temperatures below 600°F. The temporary hot leg temperature reduction was planned to gather data on emergency feedwater pump performance, on control valve testing, and to validate the computer modeling of system performance within the new operating parameters prior to making the necessary procedure changes to permanently operate at the reduced temperature. The licensee stated that operation at the reduced temperatures should reduce or preclude secondary side stress cracking corrosion and, thereby, reduce the probability of steam generator tube failures.

The inspector reviewed the 10 CFR Part 50.59 evaluation associated with Special Work Plan 2409.351, Revision 0. The licensee determined that operating at a lower reactor coolant system temperature differed from operation described in the Safety Analysis Report. Several tables and figures in the Safety Analysis Report contained full power normal operating

temperature design parameters which would not be true during operation at reduced reactor coolant temperatures. Therefore, a 10 CFR Part 50.59 evaluation was required to ensure the proposed change would not involve an unreviewed safety question.

Special Work Plan 2409.351 directed plant operation within reactor coolant system temperature limits specified in the Technical Specifications. The licensee further stated that the reactor coolant system temperature limits used in the reload analysis were conservative with respect to the temperature limits used in the safety analysis which were conservative with respect to the limits in the Technical Specifications. Other factors were determined to either be within original design margins or conservative. The licensee concluded that performance of Special Work Plan 2409.351 would not introduce any unreviewed safety questions. The inspector did not identify any deficiencies in the licensee's evaulation.

On July 7, the inspector observed the crew briefing prior to the performance of Special Work Plan 2409.351. The briefing was given by the Operations Manager and supplemented by the author of the test instruction. The briefing was excellent and clearly addressed the purpose of the test, the function of the test coordinator, operations' retention of command and control, the authority for aborting the test if necessary, the importance of slow steady maneuvering, and important limits and precautions. Expected changes for components and parameters which could be affected by a reduction in reactor coulant system temperature or secondary steam pressure were reviewed. The effect of the new operating parameters on the relevant Abnormal Operating Procedures was addressed in the special work plan and discussed during the briefing.

The test was performed successfully. As a result of the test performance, an error was detected in the computer program which was used to model secondary plant performance. Turbine Driven Emergency Feedwater Pump 2P-7A capability was confirmed at the lower steam generator saturation temperature. The licensee subsequently submitted a Technical Specification change request to the NRC Office of Nuclear Reactor Regulation. This change would lower the specified test pressure for 2P-7A. The capability of one valve in the steam bypass system to release enough energy for the main turbine control valves to stroke for Technical Specification testing without reducing reactor power was also demonstrated.

6.5 Unit 2 - Inspector Walkdown of Balance of Plant and Engineered Safeguards Feature Systems

On July 7 and 8, the inspector toured the turbine and the auxiliary buildings to inspect the balance of plant and the engineered safeguards feature system. Plant system status was normal and there were no deficiencies noted. A tag for Job Request 858556 was attached to the Emergency Feedwater Heat Tracing Panel 2C-323 and was dated December 29, 1990. The inspector questioned the licensee about the prioritization of the completion of jobs on safety systems. The licensee stated that the job was completed and closed out on JO No. 832759

in late 1990 and that the job request tag was inadvertently left on the emergency feedwater heat tracing panel. The repair of the emergency feedwater heat tracing panel was timely.

6.6 Unit 1 - Excess Back Leakage Discovered from Low Pressure Injection Crossover Check Valve DH-17

On July 8, at approximately 12:10 a.m., Technical Specification 3.1.6.9 was entered. This Technical Specification related to leakage criteria for reactor coolant system pressure isolation valves. Low Pressure Injection Crossover Check Valve DH-17 was the second check valve in series which provided reactor coolant system pressure isolation.

The measured leakage of 9.3 gallons per minute from Check Valve DH-17 was greater than the allowed cumulative leakage limit of 5 gpm from Check Valves DH-17 and DH-13. Check Valve DH-13 was subsequently found not to be leaking. The source of the leakage through Check Valve DH-17 was determined to be from Core Flood Tank T-2B. Low Pressure Injection Valve CV-1401 was de-energized and locked closed in accordance with Technical Specification 3.1.6.9. Valve CV-1401 was an outboard containment motor-operated isolation valve. Locking Valve CV-1401 satisfied Technical Specification 3.1.6.9. However, it rendered one train of low pressure injection inoperable. As a result, the licensee entered the allowed outage time of Technical Specification 3.3.6 at 2:05 a.m., July 8, 1992.

The licensee suspected Check Valve DH-17 had lifted from its seat because a small amount of undetected backleakage had reduced the differential pressure across the check valve. The vent path which was normally used to vent the backleakage was not large enough once the check valve was fully unseated.

The backleakage had been undetected because the instrument isolation valves for Pressure Transmitter PT-1401 were shut instead of open as required by procedure. As a result, Pressure Transmitter PT-1401 in conjunction with Pressure Transmitter PT-1009, indicated as though a differential pressure drop existed across the check valve when, in fact, a pressure drop sufficient to fully seat the check valve was not maintained.

After conversations with Region IV staff and members of the Office of Nuclear Reactor Regulation, the licensee requested, and was subsequently granted, a 4-hour temporary waiver of compliance from Technical Specification 3.1.6.9. The waiver was granted at 10:40 p.m. on July 8, 1992. This temporary waiver permitted the licensee to crack open Valve CV-1401 for the purpose of reseating Check Valve DH-17. The licensee used Special Work Plan 1409.420 to control the evolution. The waiver was entered at 12:46 a.m. on July 9, 1992. Valve CV-1401 was cracked open for approximately two seconds. Pressure upstream of Valve CV-1401 did not exceed 40 pounds per square inch gage (psig) during the bumping evolution. The temporary waiver was exited at 12:57 a.m., July 9, 1992, after Check Valve DH-17 was successfully reseated and zero backleakage verified. The licensee initiated Significant Condition

Report 1-92-0436 to document the failure to maintain instrument isolation Valves DH-1013A and DH-1013B in the open position as required by procedure.

The decision to measure the leakage across Low Pressure Injection Crossover Check Valve DH-17, when it was determined to be off its seat was appropriate. The decision to declare Valve DH-17 inoperable as a result of the calculated leak rate exceeding the Technical Specification limit was correct. The licensee effectively developed a safety evaluation and compensatory measures in support of their request for a temporary waiver of compliance from Technical Specification 3.1.6.9. The special work plan to crack open Valve CV-1401 was well written and very carefully implemented.

6.7 Unit 2 - Specific Activity of the Reactor Coolant System

On July 20, the inlet and outlet valves, on the sampling bomb used to determine the amount of gas entrained in the reactor coolant system, were determined to be leaking. This led to spurious data when performing surveillance testing required by Technical Specification 4.4.8. The determination of the amount of gas entrained in the reactor coolant system was normally used to determine gross activity and to verify that specific activity of the reactor coolant system was less than the Technical Specification 3.4.8 limit.

The licensee replaced the leaking inlet and outlet valves prior to exceeding the time allowed to conduct the testing required by Technical Specification 4.4.8. However, due to admitting air into the line during the maintenance activity, the results were not believed to be reliable until after several purges had been conducted. In order to perform the testing required by the Technical Specification within the required time frame, the licensee revised the sampling procedures. The temporarily revised procedures determined the amount of gas entrained in the reactor coolant system by extrapolation from the amount of hydrogen in the reactor coolant system. A calibrated measurement of the amount of hydrogen in the reactor coolant system was available. A calibrated measurement of the percent of gas in the reactor coolant system that was hydrogen was also available.

The licensee carefully evaluated options available for measuring specific activity in the reactor coolant system as required by Technical Specification 4.4.8 when the normal test methods were unavailable due to equipment malfunction. The best alternative was selected. The inspector reviewed the procedures and found the licensee's approach to be technically reasonable. The measurements used were calibrated in the range of interest.

6.8 Unit 2 - Failure of Steam Dump Bypass Control Valve 2CV-0303

On July 24, during the monthly turbine control valve testing, per Procedure 2106.009, Revision 19, Supplement 2, "Turbine Generator Operations," Steam Dump Bypass Control Valve 2CV-0303 failed closed, causing a mismatch

between reactor and secondary power. The reactor coolant system cold leg temperature increased above the Technical Specification 3.2.6 limiting condition for operation of 554.7°F.

To correct the high reactor coolant system temperature condition within the Technical Specification allowed 2 hours, the licensee increased turbine load using the load limit set feature of the turbine's electrohydraulic control system. The reactor coolant system temperature was subsequently reduced below the Technical Specification limit. The licensee aborted the control valve stroke test and attempted to return turbine control to the load limit potentiometer. While the operator was returning the main turbine generator control to the load limit potentiometer, the load limiting light failed to illuminate. The operator continued to adjust the potentiometer down, causing Turbine Control Valve 2CV-252 to close. The letdown flow increased, and a relief valve on the letdown line lifted. The reactor coolant system fluid discharged to Waste Gas Surge Tank Relief Valve 2PSV-2402.

The licensee had previously removed Valve 2PSV-2402 due to ongoing maintenance activities. The tail pipe for Valve 2PSV-2402 was not blanked off, and the reactor coolant system fluid discharged to the auxiliary building. The licensee initiated Condition Report 2-92-0186. Operators and health physics technicians were dispatched to the waste gas surge tank area. As part of their corrective actions, the licensee opened Letdown Globe Valve 2CVC-139 to reduce pressure in the letdown line, and the letdown relief valve seated. Valve 2PSV-2402 was blanked, and the reactor coolant system fluid leakage was subsequently stopped. Whole body counts were performed on the personnel in the Unit 2 controlled access area during the event, and the results were satisfactory.

Reactor power was subsequently stabilized at 95 percent. The licensee successfully completed turbine control valve testing and returned reactor power to 100 percent. The fact that the turbine control board operator was unaware that he had inadvertently decreased turbine load was viewed as a weakness because indications of turbine load other than the failed load limiting light were available. Further review of the root causes and corrective actions proposed in response to this transient and subsequent reactor coolant system spill will be tracked as Inspector Followup Item 50-368/9210-01. This review will include apparent nonsafety-related system work control weaknesses.

6.9 Unit 1 - Failure of the Pressurizer Non-Vital (Backup) and Vital (Swing) Heaters

On July 30, the midshift operator noted a 10 psig drop in reactor coolant system pressure. Low Level Cut-off Relay LSX-1001 had failed, causing a loss of power to backup heaters (Groups 1-12) and a loss of control power to the swing heater (RUB-14). The green and red train vital heaters were powered from Motor Control Centers MCC-51 and MCC-61, respectively, and automatically fired, thus restoring reactor coolant system pressure to 2155 psig. Swing

Heater RUB-14 was powered from Motor Control Center MCC-56. Relay LSX-1001 was common to the backup and swing heaters, and the relay was to actuate when pressurizer level reached a low level of 55 inches.

At 1:55 a.m., the licensee entered a 72-hour action statement of Technical Specification 3.1.3.6. The action statement required two of the three emergency (vital) groups of heaters to be operable while the reactor was critical. The licensee conservatively assumed that a loss of either the red or the green train heaters in conjunction with an inoperable swing heater group may render the plant vulnerable for inadequate reactor coolant system pressure control in the event of a loss of offsite power. This was consistent with the bases for the Technical Specification requirement of greater than or equal to 126 kilowatt (KW) heating capacity for reactor coolant system pressure control during a loss of offsite power. The swing heater was needed in conjunction with the available red or green powered heater to provide the 126 KW heating capacity. The licensee's use of bases information to ensure plant safety rather than just literal compliance with the Technical Specifications was viewed as a strength.

As a compensatory measure, the licensee initiated JO No. 020113 and Temporary Modifications No. 92-1-036 to Jumper Relay LSX-1001 with a hard wire and a toggle switch at Motor Control Center MCC-56. An operator was stationed at the temporary modification to operate the toggle switch in the event the swing heaters were needed.

Operations and health physics personnel expeditiously convened and coordinated plans to enter the reactor building to replace Relay LSX-1001. The licensee stated that maintenance activities for Unit 1 were scrutinized for possible impact during the relay replacement. The relay was replaced and tested satisfactorily, and the licensee exited the Technical Specification at 11:49 a.m.

6.10 Unit 2 - Planned Removal of the Plant Computer From Service While Operating at Full Power

The inspector interviewed licensee personnel regarding plant computer modifications planned to begin on August 4 and to complete prior to the end of Refueling Outage 2R9. The modifications were planned under a design change package which received a 10 CFR Part 50.59 review that addressed both the final configuration and operation at power with the plant computer removed.

The 30 most important computer points on the plant computer were selected by personnel from the operations, systems engineering, and design organizations. The critical points were paralleled to the critical applications process system computer, which would remain in service during the modification. As a result, this key information would not be lost to the operators when the plant computer was removed from service.

The safety parameter display system and the radiological dose assessment calculator were operated on separate computers which would remain in service

during the modification and would be available for use during emergencies. Some parameters were hard wired from the field on one circuit and then paralleled to the plant computer, critical applications process system, and safety parameter display system. The modification involved planned wire lifts that would cause brief unavailability of some information and short entry into Technical Specification action statements related to the core operating limit supervisory system and the core protection calculators. The planned unavailability would be restricted to affect one train or channel at a time.

The licensee stated that all parameters used in the emergency operating procedures which were read from the plant computer were paralleled to the critical applications process system computer and would be available.

Based on the information provided by the licensee, the inspector, following discussions with personnel in the NRC Office of Nuclear Reactor Regulation, determined that the planned removal of the plant computer from service while operating at full power would not pose any undue risk or limit emergency response capabilities.

6.11 Routine Tours

During routine tours the inspector identified weak health physics practices. A survey sheet on the 404-foot elevation was outdated. A health physics technician reached across a C-zone barrier as a reflex reaction to catch his hard hat. The hat had fallen from his head and was bouncing into the C-zone. He did wipe the hat clean with a masslin and frisked the hat. The pancake probes on the friskers were routinely left face down on the tables after use.

6.12 Summary of Findings

The inspector accompanied waste control operators and auxiliary operators on rounds. The operators were very knowledgeable and accurately recorded data. No system abnormalities were observed.

Reportability and operability determinations were made effectively with plant safety considerations given high priority. The licensee used bases information during the failure of some Unit 1 pressurizer heaters to correctly implement the Technical Specifications. The capacitor bank outage was evaluated for its impact on availability of offsite power.

The reactor coolant system hot leg temperature reduction was carefully evaluated prior to performance. The evolution was well planned. The crew briefing was excellent. The test was performed successfully. All desired information was obtained, and an error in the computer model of the secondary system was uncovered.

The decision to measure the leakage across Low Pressure Injection Crossover Check Valve DH-17 rather than just venting when the valve was determined to be off its seat was appropriate. It showed a safety first consciousness. The decision to declare Valve DH-17 inoperable as a result of the calculated leak

rate exceeding the Technical Specification limit was correct. The licensee effectively developed a safety evaluation and compensatory measures in support of their request for a temporary waiver of compliance from Technical Specification 3.1.6.9. The special work plan to crack open Valve CV-1401 was well written and very carefully implemented.

The licensee carefully evaluated options available for measuring specific activity in the reactor coolant system as required by Technical Specification 4.4.8 when the normal test methods were unavailable due to equipment malfunction. The best alternative was selected. The inspector reviewed the procedures and found the licensee's approach to be technically reasonable. The measurements used were calibrated in the range of interest.

Based on the information provided by the licensee, the inspector determined the planned removal of the plant computer from service while operating at full power would not pose any undue risk or limit emergency response capabilities.

7. PREPARATION FOR REFUELING ACTIVITIES (60705)

7.1 Unit 2 - Storage of Fresh Fuel for 2R9 Preparation (JO No. 873203)

On July 1, the licensee unloaded 12 bundles of fuel in preparation for Refueling Outage 2R9. The fuel was lifted in accordance with approved Procedure 1005.002, Revision 9, "Control of Heavy Loads," and approved Procedure 2506.03, Revision 1, "Fresh Fuel Shipping Container Operations." The bundles were surveyed by health physics and no detectable amounts of radiation and contamination were present. The bundles were inspected by the ANO reactor engineering group. A representative from the fuel contractor was present to verify any damage that might have been incurred due to shipment. The fuel bundles were stored in the spent fuel pool in locations specified in Procedure 1022.12B, Revision 15, "Nuclear Fuel Transfer Report." The inspector noted full health physics coverage during the fuel transfer. Level 1 housekeeping was invoked around the spent fuel pool area. The fuel transfer was expeditious and well coordinated.

7.2 Summary of Findings

Fuel was successfully transferred to the spent fuel pool in a well coordinated and expeditious fashion. Health physics coverage and Level 1 housekeeping were observed to be in accordance with plant policy.

8. UNITS 1 AND 2 - LICENSEE EVALUATIONS OF PROPOSED CHANGES TO THE ENVIRONS OF LICENSED REACTOR FACILITIES (TI2515/112)

Through interviews with licensee personnel, the inspector determined that a procedurally controlled periodic program to evaluate change to the environs of the reactor did not exist.

A program existed to ensure that screening reviews for environmental effects were conducted for all modifications to the facility. Further, the emergency

preparedness evacuation study was under review to ensure 1990 Census results were consistent with the original growth assumptions. In response to Generic Letter 88-20, Supplement 4, the licensee was preparing an evaluation of external events on both units. However, the licensee stated that no significant changes were known to have occurred since the plant was licensed.

9. SUMMARY OF OPEN ITEMS

LER 50-313/90-018, "Procedural Deficiencies which Resulted in Failure to Perform Adequate Local Leak Rate Test of Containment Airlocks," was closed.

Inspector Followup Item 50-368/9002-03, "Actions to Resolve the 14,000 gpm Flow Limit Imposed on the Service Water Pumps," was closed.

Inspector Followup Item 50-313/9209-02, "Hydraulic Performance Degradation of Service Water Pump P-4A," was closed.

Violation 50-313;368/9030-02, "Failure to Perform an Adequate Safety Review for Conducting Resin Transfer Cask Dewatering Activities," was closed.

Violation 50-368/9130-02, "Fire door Breached Rendering the Barrier Inoperable, Without the Shift Superintendent Being Informed," was closed.

Inspector Followup Item 50-368/9210-01, "Followup on Corrective Actions Proposed in response to Reactor Power Transient and Subsequent Reactor Coolant Spill (Condition Report 2-92-0186)" was opened.

10. EXIT INTERVIEW

The inspectors met with members of the Entergy Operations staff on August 4, 1992. The list of attendees was provided in paragraph 1 of this inspection report. At this meeting, the inspectors summarized the scope of the inspection and the findings.

ATTACHMENT

Acronyms and Initialisms

ANO	Arkansas Nuclear One
ALARA	as low as reasonably achievable
gpm	gallons per minute
JO	job order
KV	kilovolt
KW	kilowatt
LER	licensee event report
ppm	parts per million
psig	pounds per square inch gage
10 CFR Part 2.790	Section 790, Part 2, Title 10, Code of Federal Regulations
10 CFR Part 50, Appendix J	Appendix J, Part 50, Title 10, Code of Federal Regulations
10 CFR Part 50.59	Section 59, Part 50, Title 10, Code of Federal Regulations