UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos. 50-269/80-07, 50-270/80-05 and 50-287/80-05

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242

Facility: Oconee Nuclear Station

Docket Nos. 50-269, 50-270 and 50-287

License Nos. DPR-38, DPR-47 and DPR-55

Dates of Inspection: February 1-29, 1980

Inspection at Oconee Nuclear Station, Units 1, 2, and 3

Inspectors: Frank Jape — 3/31/80
F. Jape — Date Signed

for Frank Jape — 3/31/80
W. T. Orders — Date Signed

Approved by: R D Martin — 4/2/80
R. D. Martin, Section Chief, RONS Branch — Date Signed

SUMMARY

Inspection on February 1-29, 1980

Areas Inspected

This routine inspection involved 150 inspector-hours on site in the areas of plant operations, outage activities, witnessing testing, compliance with the January 2 and February 7, 1980 Show Cause Orders, plant tours, followup on previous inspection findings, verifying completion of selected fire protection commitments, and meeting with local civil defense personnel.

Results

Of the eight areas inspected, no apparent items of noncompliance or deviations were identified in six areas; two apparent items of noncompliance were found in two areas (Infraction: failure to follow a procedure resulting in damage to steam driven emergency feedwater turbine bearings , paragraph 6; and Infraction: failure to follow procedures resulting in an incorrect primary gasket to be installed on a steam generator manway, paragraph 7.)

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DETAILS

1. Persons Contacted

Licensee Employees

*J. E. Smith, Station Manager
*J. M. Davis, Superintendent of Maintenance
*J. N. Pope, Superintendent of Operations
*T. B. Owen, Superintendent of Technical Services
*R. T. Bond, Licensing and Projects Engineer
J. Brackett, Senior QA Engineer

Other licensee employees contacted included 20 operations supervisors, 8 technicians, 25 operators, 6 mechanics, 2 security force members, and 3 office personnel.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on February 15 and 27, 1980, with those persons indicated in Paragraph 1 above. The inspector discussed the items of noncompliance at the meetings. Licensee management acknowledged these findings and indicated that the event related to emergency feedwater turbine bearing failure and use of incorrect primary manway gasket on A OTSG were under investigation. Status of previous inspection findings was discussed and licensee representatives indicated efforts to resolve those remaining open would be continued in a timely fashion. Other comments and announcements were made without significant comments from the licensee.

3. Licensee Action on Previous Inspection Findings

a. (Closed) Unresolved Item (269/78-25-02) Steam Generator Tube Plugging. The licensee's actions to preclude reccurrance of steam generator tube plugging errors were examined by the inspector. Operating Procedure, OP/O/A/2005/01, "Identification of Tubes in OTSG", was revised to entail expanded tube identification and reverification techniques. Maintenance procedure, MP/O/A/1130/01, "OTSG Tube Plugging", was revised to incorporate OP/O/A/2005/01, and Station Directive 3.3.16, "Control of OTSG MAintenance", which establishes firm control over steam generator maintenance.

b. (Open) Deviation (269/79-23-01, and 270/79-21-01) Fire Protection. Items 1, 3, and 5 in the licensee's October 24, 1979 response were verified as complete. These items are closed. However, items 2 and 4 remain open pending further action by the licensee.

c. (Closed) Infractions (269/270/287/79-26-01 and 269/79-26-02) Reactor Vessel Head O-Ring. Licensee's response, dated November 15, 1979, was verified as complete. Material requisition program and maintenance procedure, MP/O/A/1150/09 were revised to insure proper procurement and installation of reactor vessel head O-rings.

d. (Open) Infraction (269/79-15-01) Snubber Operability. Licensee's response, dated August 17, 1979, states that a formal training program to instruct maintenance personnel on snubber inspection would be established. Followup on this matter reveals that the program is in preparation and has not been formalized. Additional followup will be conducted. During the interim, informal training is conducted by maintenance foreman prior to each inspection.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Plant Operations

The inspector reviewed plant operations, throughout the report period, to ascertain conformance with regulatory requirements, technical specifications and administrative directives. The control room logs, shift supervisors' logs, shift turnover records, and the removal and restoration record books for all three units were reviewed. Interviews with plant operations, maintenance, chemists, health physicists and performance personnel were held on the day and night shifts.

Activities within the control rooms were observed and monitored during day and night shifts and at shift changes. The actions and activities were conducted as prescribed in Section 3.08 of the Station Directives. The number of licensed personnel on each shift met or exceeded the minimum required by IEB 79-05C. Operators were assigned special duty as required by the modification to the January 2, 1980 Show Cause Order, issued on February 7, 1980. The modified Show Cause Order permitted Units 2 and 3 to remain in operation provided dedicated, qualified persons were assigned specified duties. This was implemented upon receipt by issuance of OP/2/A/1112/18 and OP/3/A/1102/18, "Diverse Containment Isolation and Safety Valves Positions". Operators assigned the special duty were trained. The operating procedures and the training program were examined by the inspector and were found adequate and satisfactory. In addition, an operator was assigned the "KI" inverter watch throughout the report period as committed in IE Inspection Report 50-270/79-34. Plant tours were taken during the inspection period as follows:

a. Turbine Building

b. Auxiliary Building

c. Unit 1 Reactor Building

d. Unit 1, 2, and 3 Electrical Equipment Rooms

e. Unit 1, 2, and 3 Cable Spreading Rooms

f. Station yard areas within the protected area

During the plant tours, observations were made of ongoing activities housekeeping, security, equipment status and radiation control practices. Special emphasis was given to the final cleanup of Unit 1 reactor building prior to power escalation on February 26, 1980. The inspector along with plant personnel verified that housekeeping within the reactor building was adequate for power operation.

The inspector noted that the contaminated zone, at the personnel entrance to Unit one Reactor Building, was promptly cleared of excess materials and decontaminated following completion of the outage maintenance activities. The housekeeping efforts within this radiation zone were above satisfactory.

Within the areas inspected, no items of noncompliance or deviations were identified.

6. Emergency Feedwater Turbine Maintenance

On February 5, 1980, operating procedure OP/1/A/1106/06 was employed to verify operability of the emergency feedwater turbine following extensive maintenance. Step 4.2.12 of the procedure requires, in part, that bearing oil pressure be verified to be approximately 6 psig prior to turbine operation.

Oil pump discharge pressure, approximately 35 psig, was verified but bearing oil pressure was apparently overlooked. The turbine was operated without bearing oil which resulted in the destruction of the turbine bearings.

Investigation during the maintenance to replace the destroyed bearings revealed that the pressure regulator between auxiliary oil pump and the turbine bearings was misadjusted allowing no oil flow. Failure to verify bearing oil pressure as detailed in the procedure is considered to be noncompliance with Technical Specification 6.4.1a which requires procedural compliance. This matter was discussed with licensee management on February 15, and 27, 1980. Licensee management assigned responsibility to investigate the matter and acknowledged the inspector's findings.

This item was identified as an infraction and applies to Unit 1 (269/80-07-01).

7. Steam Generator Primary Gasket Leakage; Unit 1

The primary manway gaskets on Unit 1 steam generators (OTSG) were replaced following maintenance. When raising primary system pressure and temperature

in preparation for reactor startup on February 9, a leak was discovered on the lower primary manway on "A" OTSG. The primary system was at 450 psig and 230F at the time. Investigation, by the licensee, indicated an incorrect gasket might have been installed.

Followup by the inspector revealed noncompliance in that instructions and procedures were not followed when completing the work request and performing maintenance for this activity. Details are discussed below:

a. Work requests (WR), Nos. 54292 and 57139, issued to remove and replace the primary manways did not include the replacement gasket stock number. Station Directive 3.35, "Maintenance Work Request", states that WRs of this priority be completed by a planner prior to implementation. Section IV of an WR includes ten items of information. The Station Directive describes each item and states who should fill in these data. WR's are not always completely filled in, therefore, activities affecting quality are not satisfactorily accomplished. In this instance, it appears that the craftsman or the storekeeper selected an incorrect part, which was installed.

b. Maintenance procedure, MP/O/A/1130/2, "Primary Side Manway and Handholes Installation and Removal," States that stud threads be inspected for cleanliness prior to installation. The degree of cleanliness is not specified and the installation apparently was accomplished with studs that were not clean enough in that leakage occurred at 450 psi. The stud nuts were torqued to the specified valve of 1925 foot-pounds, as recorded within the procedure. However, thickness measurements, taken by the licensee, of the gasket that leaked, revealed that it was not compressed to the value stated in the OTSG Instruction Manual OM-1201-1083. The gasket is moninally 0.175 inches thick when purchased. With proper compression, the gasket should measure 0.120 to 0.140 inches. The failed gasket measured 0.153 to 0.157 inches after removal. It appears that improper compression of the gasket was due to improper cleanliness of the studs.

The above two items are considered to be in noncompliance with Technical Specification 6.4.1.e, which requires adherence to procedures for activities affecting quality.

This is an infraction and applies to Unit 1.

8. Fire Protection

On August 11, 1978, the NRC issued amendment no. 64 to Facility Operating License No. DPR-38 for Oconee Unit 1. The amendment required Duke Power Company to install certain modifications to improve fire protection systems. Followup by the Resident Inspector was completed on items located within

the Reactor Building that were required to be completed prior to beginning cycle 6 on Unit 1. The results and findings are discussed below:

a. Smoke Detectors

Pyralarm ionization smoke detectors have been installed at eleven locations within the reactor building. Two detectors are at each location. The inspector verified the installation at each location. A functional test of each detector has been successfully completed by performing IP/0/A/250/5B.

Detectors are located as follows:

(1) above personnel hatch

(2) above each reactor coolant pump motor

(3) near the incore instrumentation area

(4) near the core flood tank

(5) near stairwell

(6) above the three reactor cooling fans

b. Fire Hose Stations

Six fire hose stations within the reactor building have been installed. These are connected to the low pressure service water system. The resident inspector verified the installation.

c. Oil Collection System

The reactor coolant pump motor oildrain and overflow system has been modified to include housing and leak collection provisions around the upper and lower bearing level devices and the upper bearing oil cooler. Drains to a collection barrel from the housings have been provided.

These modifications for all four reactor coolant pumps were verified by the resident inspector.

Within the areas inspected, there were no items of noncompliance or deviation from the committment identified.

9. Test Witnessing - Unit 1

The following test activities were witnessed by the inspector to ascertain crew performance, conformance with license and procedural requirements and

adequacy of test results. Within the area inspected, no items of noncompliance or deviation were identified.

a. Control Rod Drive Patch Verification

The control rod drive patch verification test, IP/O/A/330/2D, is designed to functionally check all power, instrumentation, and thermocoople cabling and patching. It also establishes a basis and data to allow verification of the patching, by cross-checking the hard wired computer analog signals against the patched group and position indicator signals. The inspector reviewed procedure IP/O/A/330/2D, witnessed test execution and examined test results.

The inspector observed that the test was performed by adhering to the approved procedure. Test personnel actions appeared to be correct and timely during the performance of the test and the test results indicated that acceptance criteria had been met.

b. Reactor Coolant System Leak Test

The reactor coolant system leak test, PT/O/A/200/46 is performed to assure reactor coolant system integrity prior to return to operation following opening, modification or repair of the reactor coolant system.

The reactor coolant system pressure is increased to 2285 psig and a visual inspection is made to verify that no reactor coolant leakage exists. Any other leakage to the reactor building atmosphere is evaluated prior to resumption of reactor operation.

The inspector reviewed procedure PT/O/A/200/46 which appeared to be technically adequate and in compliance with Technical Specification requirements.

The inspector witnessed the performance of PT/O/A/200/46 and observed that the test appeared to be performed as required by the approved procedure. The test personnel appeared to perform correctly and timely and the test results apparently met acceptance criteria.

c. PORV and Code Safety Valve Position Indication

Reactor coolant system relief (PORV) and code safety valves have been provided with a positive position indication in the control room. The system was installed as described in Nuclear Station Modification 1391.

Position of the three relief valves, RC-66, RC-67 and RC-68, is monitored by the TEC valve monitoring system. Acoustical accelerations proportional to valve position are generated as flow is established through the discharge piping of the valve. These signals are detected by an accelerometer strapped to the piping and are converted to a

voltage signal. The voltage is proportional to position and is displayed in the control room. An alarm module initiates a Statalarm whenever flow rate indicates the valve to be open greater than 25%.
Test procedure, TT/1/B/1391/0, "Pressurizer Valve Monitor Calibration" was provided for initial checkout and calibration of the system.

Performance of this test was witnessed by the inspector. The test was conducted at hot shutdown conditions. The modification performed adequately and met acceptance criteria. In addition, the following procedures were revised to incorporate the modifications:

EP/0/A/1800/1, "Load Rejection"
EP/0/A/1800/2, "Turbine Trip"
EP/0/A/1800/3, "Reactor Trip"

These were reviewed by the inspector and found to be adequate.

d. High Pressure Injection System Modification (NSM 1080)

Previous small break loss of coolant analyses considered the reactor coolant pump (RCP) suction line as the limiting break location for small breaks. Assuming only one of two trains of the high pressure injection (HPI) system were available, the installed system was adequate to provide the necessary core cooling. However, it has been determined that the limiting break location for small breaks is the pump discharge of the reactor coolant system cold legs and not the RCP suction. A revised safety analysis performed by the NSSS vendor, for breaks at this location, revealed that one train of HPI flow was insufficient to maintain the core covered. Interim measures have been in effect that require operator actions outside the control room. Permanent plant modifications have been installed on Oconee Unit 1. The modifications consist of a cross-connect line between the A and B HPI discharge lines down stream of the Engineered Safeguards (DS) valves and another tie-line connecting this cross-connect line and the B and C HPI pumps discharge header. The newly installed valves are manually controlled, electrically operated valves, capable of being manipulated from the control room. Thus, operator actions outside the control room will be eliminated.

The inspector reviewed and witnessed the performance of the functional test, TT/1/A/203/11, verifying operability of the station modification.

The acceptance criteria requires a flow rate via the cross-connect lines of greater than or equal to 450 gpm with RCS pressure at 600 psig. The system performed adequately as expected and was declared operable.

e. High Pressure Injection System Performance Test

The unit one high pressure injection system performance test, PT/1/A/0202/11, verifies that the HPI pumps are operable and that

various valves necessary to HPI Engineered Safeguards actuation and normal operation function properly.

The inspector reviewed procedure PT/1/A/0202/11 which appeared to be technically adequate and in compliance with technical Specification requirements.

The inspector witnessed the performance of PT/1/A/0202/11 and observed that the test appeared to be performed as required by the approved procedure. The test personnel appeared to perform correctly and timely and the results of the test appear to meet acceptance criteria.

f. Steam Generator Automatic Level Control

An automatic steam generator level control system has been installed. Details are described in Nuclear Station Modification 1275, Part K.

The system consists of two redundant trains of level control for each steam generator. Whenever there is an emergency start of the motor driven emergency feedwater pumps, the level control system will throttle emergency header valves, FDW 315 and FDW 316, to control the level at 25 inches. If there is a loss of all four reactor coolant pumps, the level will be controlled at 240 inches.

Test procedure, TT/1/A/275/5W, "Steam Generator Auxiliary Level Control System Turning Procedure", was performed with the plant at hot shutdown and at 15% power.

The inspector witnessed the test performance at 15% power and reviewed the test results of both tests. The system is considered operational and the testing successfully completed. The level control system is totally independent of the Integrated Control System, and controlled level as expected.

10. Control Rod Drop Time Test

The unit one control rod drop time test, IP/0/A/0330/03/A, is designed to functionally check the Control Rod Drive System total trip time from the Manual Trip Button to three-fourths insertion of each Control Rod. Acceptance criteria for the test dictates that the maximum control rod trip insertion time from the fully withdrawn position to three-fourths insertion (104 inches travel) must not exceed 1.66 seconds at RC full flow conditions or 1.40 seconds for no flow conditions.

The inspector reviewed procedure IP/0/A/0330/03/A which appeared to be technically adequate and examined the test results which appear to meet established acceptance criteria.

11. Meeting With Local Civil Defense Personnel

The inspector visited civil defense facilities at Oconee County and Pickens County. The visit was part of the emergency response preparedness and evaluations workshop conducted by NRC and FEMA on February 20-22, 1980. The facilities were toured at both locations and discussions were held with responsible personnel. There were no questions or comments by the inspector regarding implementation of the currently approved emergency plan as described in Station Directive 3.8.5.