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

Report Nos. 50-269/82-41, 50-270/82-41, and 50-287/82-41

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

Facility Name: Oconee Nuclear Station

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

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

Inspection at Oconee site near Seneca, South Carolina

Inspectors: A. J. Signatowicz for — 12-03-82
W. Orders — Date Signed

A. J. Signatowicz for — 12-03-82
D. Falconer — Date Signed

Approved by: J. C. Bryant — 12/3/82
J. C. Bryant, Section Chief, Division of Project and Resident Programs — Date Signed

SUMMARY

Inspection on October 10 - November 10, 1982

Areas Inspected

This routine, announced inspection involved 232 inspector-hours on site in the areas of surveillance testing, surveillance test program, maintenance activities, operations and review of licensee event reports.

Results

Of the five areas inspected, no violations or deviations were identified.

DETAILS

1. Persons Contacted

   Licensee Employees

   *J. E. Smith, Station Manager
   J. N. Pope, Supervisor Operations
   T. Owen, Supervisor Technical Services
   J. Davis, Supervisor Mechanical Maintenance
   R. Rogers, Licensing Engineer
   *T. Matthews, Licensing Engineer

   Other licensee employees contacted included technicians, operators, and staff engineers.

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on November 5, 1982, with those persons indicated in paragraph 1 above. The licensee acknowledged the findings.

3. Licensee Action on Previous Enforcement Matters

   (Closed) Violation (50-287/82-23-02): Inadequate maintenance procedure. Procedure MP/0/A/1200/27 was reviewed by the resident inspection staff. The procedural changes noted in the violation response were verified. This item is closed.

   The following unresolved items have been closed based upon the evaluation of corrective measures taken, review of appropriate documentation, and observations and discussions with licensee personnel:

   (Closed) Unresolved Item (50-269/79-05-01)

   (Closed) Unresolved Item (50-287/79-05-02)

   (Closed) Unresolved Item (50-287/79-05-03)

   (Closed) Unresolved Item (50-287/79-23-01)

   (Closed) Unresolved Item (50-269, 270, 287/81-14-01)

   The last item of above pertained to the containment purge valve issue. The containment purge valve concerns were summarized in NUREG-0737 items II.E.4.2.6 and II.E.4.2.7. For Item II.E.4.2.6, the containment purge valves are maintained closed except in the cold shutdown mode and are verified closed at least once every 31 days. For Item II.E.4.2.7, the pneumatically

operated containment purge isolation valves close on a high radiation signal from the unit vent gaseous monitor.

In a letter dated October 28, 1982, the NRC Division of Licensing concluded that the requirements of these items have been met pending approval of the License Amendment to the technical specifications requested by letter dated September 30, 1982.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Plant Operations

The inspectors reviewed plant operations throughout the report period, October 10 - November 10, 1982, to verify conformance with regulatory requirements, technical specifications and administrative controls. Control room logs, shift supervisors' logs, shift turnover records and equipment removal and restoration records for the three units were routinely perused. Interviews were conducted with plant operations, maintenance, chemistry, health physics, and performance personnel on day and night shifts.

Activities within the control rooms were monitored during all shifts and at shift changes. Actions and/or activities observed were conducted as prescribed in Section 3.08 of the Station Directives. The complement of licensed personnel on each shift met or exceeded the minimum required by technical specifications. Operators were responsive to plant annunciator alarms and appeared to be cognizant of plant conditions.

Plant tours were taken throughout the reporting period on a routine basis. The areas toured include but are not limited to the following:

Turbine Building

Auxiliary Building

Units 1, 2, and 3 Electrical Equipment Rooms

Units 1, 2, and 3 Cable Spreading Rooms

Station Yard Zone within the protected area

During the plant tours, ongoing activities, housekeeping, security, equipment status and radiation control practices were observed.

Unit 1 began the report period operating at 100 percent power. On October 22, the unit was shutdown to replace the pressurizer code safety valves. Details of this outage are documented elsewhere in this report. Criticality was reestablished on October 30, and the unit was placed on-line

the following day. At the close of the report period, Unit 1 was operating at full power with no discernible problems.

Unit 2 began the report period operating at 100 percent power. On October 14, the unit was shutdown to replace the pressurizer code safety valves. Details of this outage are documented elsewhere in this report. Criticality was reestablished on October 20, and the unit placed on-line. Power operation continued until the unit tripped on November 4. The reactor tripped on high RCS pressure caused by an ICS runback in response to a 2A feedwater pump trip on a spurious low vacuum signal. During the trip, the main steam safety valves lifted as expected. Reactor coolant pressure remained below the setpoint of the PORV and pressurizer code safety valves, primary and secondary levels remained on scale, and no ES setpoints were reached.

Criticality was reestablished on November 5, and the unit placed on-line. At the close of the report period, Unit 2 was operating at full power with no discernible problems.

Unit 3 began the report period in a shutdown condition to locate and repair a tube leak in the 3A steam generator. Nitrogen bubble, hydro-drip and eddy current checks failed to detect the leak which had apparently self-sealed during the shutdown and cooldown of the unit. Criticality was restablished on October 21, and the unit was placed on-line the following day. During power escalation, the 3A steam generator tube leakage recurred. Subsequently, power was oscillated between 50 and 95 percent for leak evaluation. Indicated leakage has remained less than 0.5 gpm.

An ICS runback to 65 percent power occurred on November 3. The unit was operating at 80 percent power when the 3B main feedwater pump (MFWP) tripped at 9:22 a.m. Details of the MFWP trip/ICS runback are documented elsewhere in this report.

At the close of the report period, Unit 3 was operating at 70 percent power with an indicated 3A steam generator tube leak of approximately 0.45 gpm.

6. Feedwater Pump Trip Transient

On November 3, at 9:22 a.m., Unit 3 was operating at 80% power. A nuclear equipment operator (NEO) was tasked with bypassing the thrust bearing wear trip device on the 'B' main feedwater pump turbine to allow I&E technicians to affect necessary maintenance. During maintenance the NEO did not use the applicable procedure, PT-0-A-290-05. Step 12.16.2 of PT-0-A-290-05 requires that the keyswitch in the thrust bearing wear trip circuit be repositioned from the NORMAL to the TEST position and that the Thrust Wear Test Complete lamp illuminates.

The NEO did not observe the Thrust Wear Test Complete indication, yet opened the dump valves and tripped the pump. This resulted in a transient runback to 65% power. All systems responded as designed during the runback.

The task being performed is quite simple and may be considered not to require a procedure; however, a procedure was available. Since main feedwater pumps are not considered to be safety-related equipment, failure to use a procedure during the test is not a violation of NRC requirements.

7. Surveillance Testing

The surveillance tests detailed below were analyzed and/or witnessed by the inspector to ascertain procedural and performance adequacy.

The completed test procedures examined were analyzed for embodiment of the necessary test prerequisites, preparations, instructions, acceptance criteria and sufficiency of technical content.

The selected tests witnessed were examined to ascertain that current written approved procedures were available and in use, that test equipment in use was calibrated, that test prerequisites were met, system restoration was completed and test results were adequate.

The selected procedures perused and identified below attested conformance with applicable Technical Specifications; they had received the required administrative review and they were performed within the surveillance frequency prescribed.

Procedure	Title
IP-0-A-0310-03C	ES Analog Channel-1 Reactor Building Pressure Narrow Range
IP-0-A-0310-03D	ESS Analog Channel-A Reactor Building Pressure Switch Calibration and Pressure Switch Contact Buffer Test
IP-0-A-0310-04C	ES Analog Channel-2 Reactor Building Pressure Narrow Range
IP-0-A-0310-04D	ESS Analog Channel-B Reactor Building Pressure Switch Calibration and Pressure Switch Contact Buffer Test
IP-0-A-0310-05B	ES Analog Channel-3 Reactor Coolant Pressure
IP-0-A-0310-05C	ES Analog Channel-3 Reactor Building Pressure Narrow Range
IP-0-A-0310-05D	ESS Analog Channel-C Reactor Building Pressure Switch Contact Buffer Test

IP-O-A-0275-05P	SG Startup and Full Range Level Instrumentation Calibration
IP-O-A-0275-05Q	SG Temperature Compensated Range Level Instrumentation Calibration
IP-O-A-0275-05W	Emergency Steam Generator Level Control System Calibration and Functional Test
IP-3-A-0305-09	RPS Channel-A FDW Pump and Main Turbine Trip Calibration

The inspector employed one or more of the following acceptance criteria for evaluating the above items:

10 CFR

ANSI N 18.7

Oconee Technical Specifications

Oconee Station Directive

Duke Administrative Policy Manual

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

Missed ES Surveillance Tests (RO-269/82-16)

On October 18, 1982, a licensee Q.A. audit detected that during January 1982 certain Engineered Safeguards (E.S.) surveillance tests were not performed. Details of the event are as follows:

Unit 1 was shutdown for refueling during the latter part of 1981. In preparation for unit start-up, licensee I&E personnel began E.S. surveillance testing on November 23, 1981. These tests were completed by December 17, 1981. They were next scheduled to be performed on January 5, 1982, which would have placed testing in the allowable time period of 45 days. However, due to the large work load on January 5, 1982, operations could not support the E.S. surveillance tests and thus they were rescheduled for January 6, 1982. The reactor was shutdown that day, therefore the tests were not required. These tests should have been rescheduled and performed prior to the unit's restart, but I&E failed to reschedule and, due to a flaw in the system, the computer did not reflect the missed tests.

On January 29, 1982 the unit achieved criticality; the tests should have been performed at this point. Since the required surveillance testing had

not been performed, the associated systems were then technically inoperable. The systems affected and the test procedures were:

System	Procedure
HPI and RB Isoation	IP/0/A/310/12A
LPI Logic Ch. 3	IP/0/A/310/12B
RB Isolation and Cooling	IP/0/A/310/12C
RB Spray Logic Ch. 7	IP/0/A/310/12D
HPI and RB Isolation	IP/0/A/310/13A
LPI Logic Ch. 4	IP/0/A/310/13B
RB Isolation and Cooling	IP/0/A/310/13C
RB Spray Logic Ch. 8	IP/0/A/310/13D
ES Analog Ch. A	IP/0/A/310/14A
ES Analog Ch. B	IP/0/A/310/14B
ES Analog Ch. C	IP/0/A/310/14C

On February 4, 1982, all the above tests were completed, the results show that the associated systems would have functioned as designed if called upon.

Analysis of the event reveals that the administrative controls for surveillance testing were inadequate to prevent this occurrence. The applicable portion of these controls is defined in Station Directive 3.3.6, Preventative Maintenance Program. I&E surveillance testing is initiated by issuance of a standing work request by the planning section. If the work request is not returned within the time specified, a notice is issued to the appropriate I&E Coordinator. For this occurrence, the standing work request was not performed and was returned to the planning section. The work request was processed as a completed work request by the planning surveillance group which stopped the issuance of the previous mentioned notification. The return to planning of the work request was the accepted method for handling work not performed due to unit conditions. When the incomplete work request was utilized to update the surveillance schedule, it negated the mechanism employed to ensure test completion.

The licensee has implemented changes to the administrative controls which should preclude recurrence of this event.

On October 4, 1982 with the Unit at 40% power, the licensee discovered that the Unit 3 weekly Borated Water Storage Tank (BWST) boron sample had not

been taken as was required per Technical Specification 4.1.3. The last boron sample of the BWST, prior to the Technical Specification violation, was taken on September 22, 1982 (unit at cold shutdown). Per Technical Specification Table 4.1.3, Item 2, the next sample was due on September 29, 1982.

Once it was determined that the BWST Technical Specification surveillance period had expired and the unit was under a limiting condition for operation per Technical Specification 3.2.2, operators began to reduce power at 10 percent/hour. After it was determined that the Unit 3 BWST boron concentration was greater than the required concentration, the unit shutdown was terminated.

The station chemist and the chemistry technicians involved were counseled, and administrative changes are being implemented within the chemistry section to make clear the exact status/frequency requirements of the BWST boron sample tests. The primary chemistry first line supervisor will verify that Technical Specification surveillances are met using the existing sample verification program. These actions should preclude recurrence.

The events described above violate technical specification surveillance requirements; however, since the licensee received a similar violation in inspection report 50-269/82-36 for which a response and corrective action have not been completed, these items are considered as additional examples of the earlier violation.

8. Surveillance Test Program Adequacy

During the current report period, the resident inspection staff evaluated the licensee's program for ensuring performance of surveillance tests required by Technical Specifications. The results of the inspection are categorized below:

a. Administrative Controls

On October 26, 1982, Oconee Nuclear Station Directive 3.2.2 was revised to provide assurance that Technical Specification Amendments will be incorporated into plant procedures, instructions, or drawings as appropriate. The revised version of Station Directive 3.2.2 should provide implementation verification.

b. Implementation Verification

The three most recent license amendments, 113-113-110, 112-112-109, and 111-111-108 for the three Oconee units respectively, were evaluated to verify that they had resulted in acceptable revisions to appropriate documents. Inspection revealed that the requirements had been incorporated into the appropriate documents.

In addition, older license amendments were evaluated. The results of the latter evaluation are delineated in report 50-269/82-36. In

summary, the most recent license amendments appear to have been adequately implemented, and/in view of the revision to Station Directive 3.2.2, future amendments should be promptly and effectively implemented.

c. Licensee Implementation Program

Station Directive 3.2.2 as revised of October 26, 1982 appears to provide an acceptable systematic method for assuring that the Technical Specification revisions are implemented by a document or instruction.

d. Surveillance Test Verification

The licensee's program for assuring that surveillance tests are scheduled and completed was evaluated to determine its effectiveness. The resident inspection staff randomly selected 50 surveillance tests and verified that the surveillances were included on the facility schedule, an adequate procedure exists for each, and that surveillance frequency was adhered to.

e. QA Surveillance Audits

The resident inspection staff and the Region II Management Programs Section personnel evaluated the licensee's QA surveillance test audit program and found that it appears acceptable pursuant to current regulatory requirements.

9. Maintenance Activities

Maintenance activites were observed and/or reviewed throughout the report period to ascertain that the work was being performed by qualified personnel, that activities were accomplished employing approved procedures or the activity was within the skill of the trade. Limiting conditions for operation were examined to ensure that technical specification requirements were satisfied. Activities, procedures, and work requests were examined to ensure adequate fire protection, cleanliness control and radiation protection measures were observed and that equipment was properly returned to service.

Acceptance criteria employed for this review included but was not limited to:

Station Directive,

Administrative Policy Manual,

Technical Specifications,

Title 10 CFR

WORK REQUEST NUMBER	COMPONENT
19953	Unit 3 Generator Lockout Relay
25564	Valve 3 BS-2
28240	Valve 3 HP-366
28369	Unit 3 Hydrogen Gas Analyzer
52586B	Unit 3 Containment Emergency Hatch
28874A	Valve 2 RC-84
15237	Valve 3 HP-120
25876	Valve 2 LP-41
28235	Valve 3 HP-285
28232	Valve 3 HP-283
28234	Valve 3 HP-284
28280	Valve 3 HP-234
28683A	Valve 3 MS-87
91060B	3A Seal Supply Filter
28833A	Valve 2 RC-68
28832A	Valve 2 RC-67
52587B	Incore Tank

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

10. Pressurizer Safety Valve Adjustment

NUREG-0737 Item II.D.1 required that a relief and safety valve test program be conducted to verify operability of these valves under postulated accident conditions. Duke Power was provided the results of the testing, including the testing of Dresser 31739A Safety Valves with short inlet piping as are used at Oconee Nuclear Station. The results of the testing showed that the Dresser safety valve performance is significantly affected by backpressure and blowdown ring settings. Based on these findings, Duke Power began a three-phased approach to complete the analysis of Oconee safety valve performance. First, Duke initiated an analysis to determine the back-pressure which the valves would see under various conditions. Second, Duke analyzed the significance of safety valve blowdown on plant performance.

Also, Duke initiated an analysis of Oconee valves, to determine the optimum ring settings to be used.

On October 12, 1982, Dresser provided the blowdown ring settings of five of the eight Oconee valves (two per unit plus two spares). Duke became concerned with the difference between the valve settings and the recommended settings, shipped the two spare valves to Wylie Labs for adjustment, and subsequently replaced the valves on each of the units with properly adjusted valves.

For further specifics refer to LER RO-269/82-18.

11. Bulletins

(Closed) IEB 80-04 (50-269, 270, 287/80-BU-04). Duke Power Company responded to IE Bulletin 80-04 in a letter to the NRC dated May 7, 1980 and provided additional information for review in a letter dated July 23, 1982. NRC consultant, the Franklin Research Center, reviewed the submittals made by the licensee in response to IE Bulletin 80-04. Based on the Technical Evaluation Report dated September 28, 1982, the following conclusions were made regarding the postulated MSLB with continued feedwater addition for Oconee 1, 2, and 3:

1. There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition because the main feedwater system is isolated and auxiliary feedwater flow to the affected steam generator is restricted.

2. The emergency feedwater (EFW) pumps are adequately protected against a runout flow condition and therefore can be expected to carry out their intended function without incurring damage in the event of a MSLB.

3. All potential water sources were identified and, although a reactor return-to-power is predicted, the specified acceptable fuel design limits are not exceeded; therefore, the FSAR reactivity increase analysis remains valid.

This item is closed.

12. Review of Licensee Event Reports

The inspectors performed a review of nonroutine event reports to verify that the report details met license requirements, identified the cause of the event, described corrective actions appropriate for the identified cause, and adequately addressed the event and any generic implications. In addition, the inspectors examined selected operating and maintenance logs, and records and internal incident investigation reports. Personnel were interviewed to verify that the report accurately reflected the circumstances of the event, that the corrective action had been taken or responsibility assigned to assure completion, and that the event was reviewed by the

licensee, as stipulated in the Technical Specifications. The following event reports were reviewed:

Report Number	Title
RO-269/81-09	Startup Transformer CT-1 Locked Out
RO-269/81-11	Core Barrel Thermal Shield Bolts Broken
RO-269/81-14	High Pressure Service Water Pump Without Control Power
RO-269/81-15	RPS Modules Found To Have Errors
RO-269/81-16	MPT Indications on Eleven RCPIA2 Bolts
RO-269/81-18	Safety-Related Battery Surveillance Not In Compliance
RO-269/81-20	Hydrogen Purge Unit Declared Inoperable
RO-269/81-24	Penetration Room Ventilation Inoperable
RO-269/82-01	1B MDEFW Pump Inoperable
RO-269/82-02	Group 8 APSR's Operating in Restricted Region
RO-269/82-03	IA Steam Generator Tube Leak
RO-269/82-04	Flaw Indication in Normal Make-up Line Weld
RO-269/82-05	IA RBS Pump Inoperable Due to No Flow Operation
RO-269/82-06	IB Steam Generator Tube Leak
RO-269/82-07	Clogged Inlet Strainer Cause Inoperable Fire Hydrant
RO-269/82-08	Breach of Containment Integrity
RO-269/82-12	Both RBS Trains Inoperable
RO-269/82-13	Failure to Test EFW Initiation Circuitry
RO-270/81-16	Steam Generator Tube Leak
RO-270/81-18	Loss of Prime to Emergency CCW System

RO-270/81-19	Isolated Fire Protection System to CT-5 Transformer
RO-270/81-20	Channel A BWST Level Instrumentation Inoperable
RO-270/82-01	Reactor Building Equipment Hatch Not Seated
RO-270/82-02	Core Barrel Thermal Shield Bolts Broken
RO-270/82-03	Fuel Assembly Broken Holddown Springs
RO-270/82-04	2A2 HPI Thermal Sleeve Loose and Nozzle Cracked
RO-270/82-05	Experimental Zircaloy Spacer Grids Moved
RO-270/82-06	Spent Fuel Moved With SFP Ventilation Inoperable
RO-270/82-07	Secondary Vertical Shield Wall Tendon Broken
RO-270/82-09	Valve 2FDW-314 Would Not Fully Open
RO-287/81-14	RPS Settings Less Conservative than Tech. Spec.
RO-287/82-01	BWST Channel B Level Instrumentation Inoperable
RO-287/82-02	RPS Channel B Hot Leg Temperature Instrumentation Inoperable
RO-287/82-03	3A Steam Generator Tube Leak
RO-287/82-04	3A2 HPI Nozzle Thermal Sleeve Weld Broken
RO-287/82-05	Spent Fuel Moved With SFP Ventilation Inoperable
RO-287/82-06	Deformation of Internal Auxiliary Feedwater Header
RO-287/82-07	Broken Fuel Assembly Holddown Springs
RO-287/82-08	Thermal Shield Bolt Failures

These items are closed.