UNITED STATES
ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATIONS
REGION II - SUITE 818
230 PEACHTREE STREET, NORTHWEST
ATLANTA, GEORGIA 30303

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RO Inspection Report No. 50-270/73-9

Licensee: Duke Power Company
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Facility: Oconee Unit 2
Docket No.: 50-270
License No.: CPPR-34
Category: A3/B1

Location: Seneca, South Carolina

Type of License: B&W, PWR, 2568 MW(t)

Type of Inspection: Routine, Unannounced

Dates of Inspection: July 2-3, 1973, and July 17-20, 1973

Dates of Previous Inspection: June 20-22, 1973

Principal Inspector: F. Jape, Reactor Inspector
Facilities Test and Startup Branch

Accompanying Inspector: K. W. Whitt, Reactor Inspector
Facilities Test and Startup Branch

Other Accompanying Personnel: None

Principal Inspector: Frank Jape 7-31-73
F. Jape, Reactor Inspector
Facilities Test and Startup Branch
Date

Reviewed By: C E Murphy 7/31/73
C. E. Murphy, Chief
Facilities Test and Startup Branch
Date

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SUMMARY OF FINDINGS

I. Enforcement Actions

None

II. Licensee Action on Previously Identified Enforcement Matters

A. Violations

Welding Program Deficiencies (RO:II Letter to DPC, dated March 8, 1972, Item 5)

RO review of the report on welding deficiencies and documentation continues. This item remains open.

B. Safety Items

There are no previously identified safety items.

III. New Unresolved Items

73-9/1 Control Rod Drive Breaker Undervoltage Trip Assembly Deficiency

Corrective actions described in licensee's report, dated February 23, 1973, have not been completed. Installation of a 5 amp fuse in the reactor protection system channels remains to be completed. (Details I, paragraph 5)

IV. Status of Previously Reported Unresolved Items

73-8/1 Body Wall Thickness of Valves 2-51-244 and 2-51-245

Justification for body wall thickness of valves 2-51-244 and 2-51-245 being less than that permitted by RO letter, dated June 30, 1972, paragraph 3, remains to be resolved.

73-8/2 Valve Wall Thickness of Valve 2-RV-67

Calculation of valve wall thickness of valve 2-RV-67 in accordance with the applicable codes remains to be resolved.

73-6/1 Test Sequence for the Reactor Building Structural Integrity Test and the Integrated Leak Rate Test

The test sequence used by the licensee to conduct the reactor building structural integrity test and the integrated leak rate test was in agreement with Appendix J of 10 CFR 50. This item is closed. (Details I, paragraph 2)

73-1/1 Core Flooding System Testing Requirement

The results of TP 201/7, "Core Flooding System Flow Test," remain to be analyzed by the licensee and reviewed by the inspector. This item remains open. (Details I, paragraph 3)

V. Design Changes

None

VI. Unusual Occurrences

None

VII. Other Significant Findings

None

VIII. Management Interview

A management interview was held with J. E. Smith, Plant Superintendent, on July 3, 1973. During this meeting, the inspector discussed his findings on the integrated reactor building leak rate test.

A second management interview was held on July 20, 1973, at the conclusion of the inspection. Those in attendance included:

Duke Power Company (DPC)

J. E. Smith - Plant Superintendent
J. W. Hampton - Assistant Superintendent
R. L. Weber - Assistant Project Engineer
G. W. Cage - Assistant Operating Engineer

The following items were discussed:

A. Hot Functional Test Program

The inspector stated that he had reviewed the hot functional test program including several procedures, and that he had discussed questions and comments on the procedures with the technical support engineer. He then requested copies of the heatup and cooldown procedures for review. These were provided before the inspector left the site. (Details II, paragraph 2)

B. Pump Guide Vanes

The inspector stated that he had discussed this subject with DPC personnel. He explained what he had learned and asked if anyone disagreed. No disagreement was voiced. (Details II, paragraph 3)

C. Barksdale Pressure Switches

The inspector stated that he had discussed this matter with DPC personnel and had been informed that no Barksdale pressure switches of the particular type of primary interest were in use at Unit 2. A licensee representative replied that this information was correct. (Details II, paragraph 4)

D. Integrated Leak Rate Test

The inspector stated that he had reviewed the integrated leak rate test result and had no comment or questions. The unresolved item 73-6/1 regarding testing sequence has been resolved. (Details I, paragraph 2)

E. Unresolved Item 73-1/1, "Core Flooding System Testing Requirement"

The results of TP 201/7, "Core Flooding System Flow Test," were reviewed by the inspector. This unresolved item remains open pending resolution of a question requiring further analysis of the test results. (Details I, paragraph 3)

F. Review of Emergency Procedures

The inspector stated that he had completed his review of the emergency procedures and that previous questions have been resolved. (Details I, paragraph 4)

G. Control Rod Drive Breaker Undervoltage Trip Assembly Deficiency

The corrective actions described in the licensee's report regarding this design deficiency were reviewed. The inspector stated that the corrective actions have not been completed and that this item will be carried as Unresolved Item 73-9/1 pending completion of the work. (Details I, paragraph 5)

DETAILS I

Prepared By: Frank Jape 7-31-73
F. Jape Date
Reactor Inspector
Facilities Test and Startup Branch

Dates of Inspection: July 2-3, 1973
July 17 and 19-20, 1973

Reviewed By: C. E. Murphy 7/31/73
C. E. Murphy, Chief Date
Facilities Test and Startup Branch

1. Individuals Contacted

Duke Power Company (DPC)

J. E. Smith - Plant Superintendent
R. C. Collins - Unit 2 Performance Engineer
R. M. Koehler - Technical Support Engineer
J. W. Hampton - Assistant Plant Superintendent
M. D. McIntosh - Operating Engineer
L. E. Summerlin - Staff Engineer
L. E. Schmidt - Assistant Operating Engineer
O. S. Bradham - Instrument and Control Engineer

Bechtel Power Corporation (Bechtel)

D. Dundas - Engineer
G. Cranston - Engineer

2. TP 150/3, "Reactor Building Integrated Leak Rate Test"

The review of TP 150/3, "Reactor Building Integrated Leak Rate Test," and the witnessing of the test has been completed.

a. Resolution of Test Procedure Comments

Comments were discussed with the licensee's representatives during a previous inspection.[1] The inspector reviewed the resolution of these comments by the licensee and found all had been incorporated into the test procedure in a satisfactory manner.

1/ RO Inspection Report No. 50-270/73-7, Details I, paragraph 2.

b. Conduct of the Test

The inspector witnessed the surveillance leak rate test which was performed on July 2 and 3, 1973. During a previous inspection,[2/] a question was raised regarding test sequence.

The test sequence selected and used by the licensee was in agreement with Appendix J of 10 CFR 50. The surveillance test was run followed by the accident pressure test. The inspector had no further comments on this question.

The measured leak rate at the surveillance test pressure of 29.75 psig was 0.00826% per day, and at the accident test pressure of 59 psig the leak rate was 0.00207% per day. These values are within the acceptance criteria for the test.

The reactor building pressure was reduced to atmospheric at 1630 hours on 7-6-73. The inspector observed that the 8-inch test pressurization line had been blanked off and that air pressure to the isolation valves was in service.

3. TP 201/7, "Core Flooding System Flow Test"

The results of TP 201/7, "Core Flooding System Flow Test," were reviewed by the inspector. The inspector commented that the analysis of the test results do not appear to lead to the conclusion that the core flooding system will perform as described in the FSAR. Specifically, paragraph 6.1.3.2 of the FSAR presents a response statement for the core flooding system and Figure 14-35 presents information regarding the performance characteristics for the system. It would appear that a relationship between these references and the test results could be established. The licensee's representative stated that this question would be reviewed. This previously identified unresolved item will remain open pending resolution of the analysis of test results.

4. Review of Emergency Procedures

During a previous inspection,[1/] a lack of three emergency procedures (EP) was identified. The licensee's representatives stated that either an EP would be provided or a related alarm procedure (AP) would be expanded to cover the condition.

2/ RO Inspection Report No. 50-270/73-7, Details I, paragraph 7.
1/ RO Inspection Report No. 50-270/73-6, Details I, paragraph 11.

The licensee elected to expand related alarm procedures.

The inspector reviewed the following AP's and found them to satisfy the three conditions:

AP 1703/33, "Instrument Air System Trouble"
AP 1702/59, "ICS Emergency Power Failure"
AP 1702/23, "ICS Auto Power Failure"
AP 1702/24, "ICS Manual Power Failure"
AP 1702/27, "RC Pressurizer Level High-Low"
AP 1702/28, "RC Pressurizer Level Emergency High-Low"

This completes the review of EP's.

5. Control Rod Drive Breaker Undervoltage Trip Assembly Deficiency

The corrective actions described in the licensee's report regarding the design deficiency, dated February 23, 1973, were reviewed by the inspector.

The undervoltage trip device has been replaced on the six control rod drive breakers. The dropout voltage has been measured to be as follows:

Breaker	Dropout Voltage
SN-1	54 VAC
SN-2	53 VAC
SN-3	65 VAC
SN-4	57 VAC
SN-5	55 VAC
SN-6	61 VAC

The field change to install a 5 amp fuse in each of the four reactor protection system channels has not been completed.

This work is scheduled to be completed prior to performing tests on the CRD system. This item will be carried as Unresolved Item No. 73-9/1 until completion of the work and subsequent inspection.

6. Preoperational Testing Program Status

	No.	%
Tests completed and results approved	126	55
Tests completed, results not approved	70	31
Tests in progress	32	14
	228	100

DETAILS II

Prepared by: CEMurphy for 7/31/73
K. W. Whitt, Reactor Date
Inspector, Facilities Test
and Startup Branch

Dates of Inspection: July 17-20, 1973

Reviewed by: CEMurphy 7/31/73
C. E. Murphy, Chief Date
Facilities Test and
Startup Branch

1. Individuals Contacted

Duke Power Company (DPC)

J. E. Smith - Plant Superintendent
J. W. Hampton - Assistant Plant Superintendent
R. M. Koehler - Technical Support Engineer
O. S. Bradham - Instrument and Control Engineer
D. G. Beam - Project Manager, Construction
*T. F. Wyke - Principal Mechanical Design Engineer

*Contacted by conference phone.

2. Hot Functional Test Program

The hot function test program was reviewed for content of the program and for quality of approved test procedures. Eleven hot functional test procedures were reviewed and comments on two were discussed with the technical support engineer. Questions on others were resolved. The procedures for heatup and cooldown were not reviewed, but they will be before or during the next inspection. The comments submitted were as follows:

a. 1P/1/A/330/20, "CRD System Patching Scheme and Functional Cabling and Patching Test"

(1) Comment - According to the Administrative Policy Manual for Operational Quality Assurance, this procedure number indicates that it applies only to Unit 1 and has not been properly approved for Unit 2.

Response - This condition will be corrected.

(2) Comment - The procedure has three changes attached to it. Each of these changes has different dates, and yet all three are numbered No. 1.

Response - This will be evaluated and corrected as necessary.

(3) Comment - Change 1 dated April 3, 1973, references an attached procedure. There does not appear to be an attached procedure.

Response - Marked pages of the procedure should be attached to the change to show latest revision. This will be corrected.

b. TP/2/B/600/8, "Component Cooling System Operational Test"

Comment - How is the requirement of prerequisite 7.3 met? How is the total number of instruments required to be calibrated determined?

Response - This prerequisite will be revised to better identify the instrumentation calibration requirements.

c. Other Procedures Reviewed

(1) 1P/0/B/340/11-1, "Minimum Run-Latch-Unlatch-Current Test."

(2) TP/2/A/600/15, "CRD System Operational Test."

(3) 1P/0/B/330/1, "CRD System Integrated Test."

(4) TP/2/A/203/5, "Low Pressure Injection System Functional Test."

(5) TP/2/A/600/14/2, "Pipe and Component Hanger Hot Functional and Injection Test."

(6) 1P/0/A/330/3A-1, "CRD Rod Drop Time Test."

(7) TP/2/B/0200/05, "Reactor Coolant Pump Initial Operation Test."

(8) TP/2/A/203/6A, "500 Psig Low Pressure Injection E.S."

(9) TP/2/B/202/7, "High Pressure Injection Operational Test."

3. Pump Guide Vanes

None of the reactor coolant pumps nor the main feedwater pumps were manufactured by Byron-Jackson, and the guide vanes were not field installed. The three hot well pumps and the three condensate booster pumps are Byron-Jackson pumps, but the guide vanes were not field installed. This information was provided by the project manager, construction. No further inspection effort is planned for this item.

4. Barksdale Pressure Switches

The use of Barksdale pressure switches was discussed with licensee personnel. Of particular interest was Model No. B2T-A12SS. The inspector was informed that no Barksdale pressure switches of this type were in use or planned for Unit 2. Licensee personnel indicated that they were aware of the problems encountered by others regarding these switches and that efforts were being made to prevent such problems at Oconee. No further inspection effort is planned for this item.