U. S. ATOMIC ENERGY COMMISSION
REGION II
DIVISION OF COMPLIANCE

Report of Inspection

CO Report No. 50-269/71-3

Licensee:	Duke Power Company Oconee 1 License No. CPPR-33 Category B
Dates of Inspection:	February 24-26, 1971
Date of Previous Inspection:	February 10, 1971

Inspected By: C. E. Murphy — 3-26-71
C. E. Murphy, Reactor Inspector (Operations) (In Charge) — Date

C. M. Upright — 3-26-71
C. M. Upright, Reactor Inspector (Operations) — Date

Reviewed By: W. C. Seidle — 3/31/71
W. C. Seidle, Senior Reactor Inspector — Date

Proprietary Information: None

SCOPE

A routine, announced inspection was made of the 2,452 Mw(t) pressurized water reactor under construction near Seneca, South Carolina, known as Oconee Station No. 1. Purposes of the inspection were:

1. To determine the construction status and significant changes to the schedule dates.

2. To review the outstanding construction items remaining to be completed at the facility.

3. To review the progress of the test program.

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SUMMARY

Safety Items - None

Nonconformance Items - Contrary to the requirements of the FSAR and established cleaning procedures, the licensee failed to properly protect the reactor vessel and internals from recontamination. (See Management Interview and Section F.)

Unusual Occurrences - None

Status of Previously Reported Items - The licensee's response to the CDN relating to the relocation of the station batteries, the electrical and instrumentation documentation and the procedure for cleaning the reactor coolant system was considered to be satisfactory.[1/]

Other Significant Items -

1. The damaged expansion bellows on the fuel transfer tube has been replaced with a new bellows. (See Exhibit A, Item 32.)

2. Wells advised the inspector that to his knowledge, there had been no overflights of the facility by military or private aircraft.[2/] Commercial flights over the facility are at several thousand feet in altitude and are not considered a hazard. Photographs of the facility are made by the licensee on a periodic basis from a helicopter, but the licensee does not permit this craft to fly above the plant nor is it permitted to fly so close that it could glide into the plant if it experienced an engine or other malfunction.

Outstanding Items - See Exhibit A for the current status of outstanding items.

Management Interview - The management interview was held on February 26 and was attended by Rogers, Beam, Hunnicutt, Smith and Hampton.

1. The inspector expressed concern about the wood splinters in the reactor vessel internals and the polyethylene trapped under the core shield support flange. He stated that it appeared that the vessel and internals had not been properly protected after they had been cleaned. Rogers stated that the ladder had been removed from the vessel immediately after the Compliance inspector had advised them

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1/ CDN issued January 29, 1971.

2/ Memo from CO:HQ (O'Reilly) dated February 1, 1971.

of the problem. He further stated that they had removed all the splinters that they had located and were considering methods of locating and removing any that might remain. He also stated that a study was being made by the licensee and Babcock and Wilcox Company (B&W) of the possible adverse effects of the polyethylene. The inspector stated that he would want to review any in-house report that was issued relating to these items. He also advised Rogers that it appeared that the licensee was in nonconformance with his FSAR and his cleaning procedures and that a CDN would probably be issued relative to these incidents. (See Section F.)

2. The inspector advised Hunnicutt that he had observed unidentified stainless steel welding rods on a workbench in the auxiliary building. He had also observed stainless steel transducer tubing without end caps or other means of protection in the same area. Hunnicutt stated that he would take corrective action to prevent recurrence of this type incident. (See Section C.)

3. The inspectors advised Rogers that they had observed a welding electrode holder clamped to an installed section of instrumentation tubing. Rogers stated that this was contrary to their accepted practices and would be discussed at the next foremen's meeting. (See Section C.)

4. In response to the inspector's question, Hunnicutt stated that the in-house report on the ITE relay failures had not been issued but that he would follow up on this item. (See Exhibit A, Item 50.)

5. The inspectors questioned the adequacy of the drip pans on the fuel handling cranes. Hunnicutt stated that he would review the design of the cranes to assure that pans were of adequate size and installed at all the required points. Smith stated that he would verify that procedures were developed that would require the pans to be inspected and maintained. (See Section K.)

6. In response to the inspector's question, Smith stated that the initial fuel loading procedure would be rewritten and that the inspector's comments would be included. (See Section H.)

7. The inspector stated that the revised administrative procedure, "Guide for Conducting the Oconee Initial Test Program," appeared to give the test coordinator more authority to make revisions to procedures than might be permitted by 10 CFR 50, Appendix B. Smith stated that he did not agree with the inspector's interpretation, but would review the requirements of 10 CFR 50 and would take corrective action if required. (The inspector received a telephone call from Smith on March 2, 1971, and was advised that the procedure

would be corrected to limit the coordinator's authority to correcting typographical errors and minor mistakes.) The inspector will review this item on the next inspection. (See Section G.)

8. In response to the inspector's question, Smith stated that he had checked the operation of the elevator in the auxiliary building and has found that the doors did not close automatically at the penetration room level. (See Exhibit A, Item 59.) He advised the inspectors that the licensee had not decided as yet what corrective action would be taken. The inspectors stated that this would be considered an outstanding item and they would review this item on a future inspection.

9. The inspector asked Smith if he planned to perform the reactor building controlled leak rate test at both the accident pressure and at 50% pressure. Smith stated that he did not think that the test would be performed at both pressures but that he would advise the inspector as soon as a definite decision was made. The inspector was advised by telephone on March 2, 1971, that the tests would be made at both pressures.

10. In response to the inspector's question, Smith stated that administrative steps were being taken to insure that all comments were incorporated into the plant procedures prior to their approval and use. (See Section G.)

DETAILS

A. Persons Contacted

Duke Power Company (Duke)

R. L. Dick - Manager of Construction
J. C. Rogers - Construction Manager, Oconee Station and McGuire Station
D. G. Beam - Project Engineer, Oconee
J. R. Wells - Project Engineer, McGuire
G. L. Hunnicutt - Principal Field Engineer, Oconee
J. E. Smith - Plant Superintendent
J. W. Hampton - Assistant Plant Superintendent
M. D. McIntosh - Operating Engineer
R. M. Koehler - Technical Support Engineer
C. A. Price - Electrical Design Engineer

B. Administration and Organization

The following changes have been made to the licensee's organization since the last inspection:

1. Rogers has been promoted to the position of Construction Manager with overall responsibility for both Oconee and McGuire.

2. Beam has been promoted to the position of Project Engineer, Oconee.

3. Wells has been promoted to the position of Project Engineer, McGuire, but at the time of the inspection, he had not transferred from Oconee.

4. Hunnicutt has been promoted to the position of Principal Field Engineer, Oconee.

5. L. E. Summerlin, formerly Technical Support Engineer, Operations, has been given a staff position and Koehler has been promoted to the position of Technical Support Engineer.

C. Quality Assurance

Welding

During a tour of the plant, the inspectors observed a welding electrode holder clamped to an installed safety-related instrument sensing tubing. They also noticed several unidentified stainless steel welding rods on a workbench in the auxiliary building and a number of lengths of stainless steel tubing without end caps or other means of protection in the area of the workbench. The tubing was the type normally used for safety-related instrumentation transducer sensing lines. These items were brought to Hunnicutt's attention and he was reminded that welding rod which could not be identified to source had been observed during the previous inspection. Hunnicutt stated that he recognized the seriousness of these occurrences and that he would discuss the items with his inspectors and would take whatever corrective action necessary to prevent a recurrence. These items were discussed in the management interview and the inspectors plan a followup inspection during the next site visit.[1/]

[1/] CO Report No. 50-269/71-2.

D. Construction Progress

1. The reactor vessel internals, except for the plenum, have been installed.

2. Installation of the control rod drive mechanisms is in progress.

3. The erection of the fuel handling cranes is underway. The manufacturer's technicians are currently making modifications and adjustments to correct deficiencies noted by the licensee.

4. Mirror insulation is presently being installed on the reactor vessel.

5. Approximately 10% of the nuclear steam supply system flushing and hydrostatic testing has been completed.

E. Construction Schedule

The following dates were given the inspectors as the best information available to the licensee and is based on a computer printout on February 25, 1971:

1.	Reactor Coolant System Hydrostatic Test	April 21, 1971
2.	Reactor Building Leak Rate Test	May 1, 1971
3.	Hot Functional Test - Start	May 8, 1971
	- Complete	July 11, 1971
4.	Keowee Functional Test	July 11, 1971
5.	Fuel Loading	July 1971
6.	Start Power Ascension	August 31, 1971
7.	Achieve 100% Power	October 25, 1971

F. Reactor Vessel and Internals

Installation of Vessel Internals - Attachment K

The licensee has installed the lower grid, thermal shield, core barrel and core support shield in the reactor vessel. The inspectors reviewed the records relative to this installation. The Vessel Internals Installation Procedure, FIP-22, included information concerning the handling, assembly and installation of the reactor internals. A B&W supplied procedure gave detailed steps for the installation of instruments to measure flow-induced vibration of the reactor internals. Tool control procedures and cleanliness requirements were included in Appendix A of this procedure. No deficiencies were noted in these records.

The licensee maintained a daily summary log of significant events during the installation of the internals. The inspectors reviewed the logs for the period from January 9, 1971, through February 20, 1971. The log contained information concerning guide block settings, fitup of internals into vessel, results of laboratory tests for component cleanliness, and plenum drilling logs. The welding procedures used during the assembly, welding inspection records, security watch logs and tool checklists were also included. No deficiencies were noted in these records.

During discussions of the installation with Hunnicutt, the inspectors were advised that on January 27, 1971, a strip of 4-mil-thick polyethylene had been found trapped between the mating flanges of the core shield and core barrel.[1/] The daily summary log of significant events documented the actions taken by the licensee after the polyethylene had been found. Details of this incident are as follows:

> The core support shield lower flange mates to the upper flange of the core barrel The two assemblies are bolted together with 120 bolts spaced on approximately 3-11/16-inch centers. During assembly of these components in the vessel, the licensee had draped the 4 mil polyethylene sheeting over the walls to afford protection and to maintain the established clean conditions of the vessel and the internals.
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> After all bolts had been installed and their retaining clips had been tack welded to them, it was found that one section of polyethylene about six feet long had been trapped between the mating surfaces. (See Exhibit B.)
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> The licensee attempted to pull the trapped material out and succeeded in removing a strip approximately three feet long. The width of this strip was approximately 1/8 inch. Attempts to remove the remaining piece were unsuccessful. The licensee estimates that the remaining piece is 1/8 to 3/16 inch wide and 3 feet long. Since the distance from the bolt holes to the edge of the flange is 5/8 inch, the licensee is of the opinion that this would represent the maximum possible width of the strip. Two plies of the material are trapped for a distance of about six inches at one end. The licensee has made tests of the polyethylene sheet and found that when compressed to the degree calculated to be approximately represented by the torqued bolts, that the thickness is reduced from 4 mils to 1 mil. The

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1/ Inquiry Memorandum to Compliance Headquarters (Henderson) from Region II (Seidle) dated March 1, 1971.

licensee has postulated that the polyethylene will melt during operation and does not consider that the relaxation of the bolts will be a problem. If the material does melt and is released into the system, the licensee feels that it will be trapped in the purification system filters or will plate out on the steam generator tube sheets. The studies were not complete as yet and Hunnicutt was not able to state what would be the effects of the material in other portions of the coolant system. Both the licensee and B&W are having chemical analyses made of the polyethylene to determine if any halogens are present.

When the inspector attempted to examine the installation, he found that the portable steel ladders had been removed and a rope ladder with wood rungs had been installed. In climbing on this ladder, the inspector observed wood splinters falling from the rungs. Other splinters were observed on the core barrel flange. Examination of the rungs indicated that numerous splinters possibly had been dislodged from the rungs and fallen into the internals. When the inspector pointed out that the use of this type ladder compromised the vessel cleanliness, the ladder was removed immediately. Because of the installation of the vessel internals, however, Hunnicutt could not advise the inspector of a method that could be used to assure that the splinters could be located and removed from the reactor vessel. He stated that Region II would be kept appraised of the results of any studies or analyses that were made regarding the polyethylene or the splinters. The inspector stated that it appeared that the licensee may be in nonconformance with the FSAR in that the vessel had not been properly protected after cleaning and a CDN would possibly be issued relating to these items. The inspector will follow up on these items during future inspections.

G. Preoperational Testing (Attachment M, PI 5800/2)

1. General Review of Testing Program (PI 5805.01)

The preoperational testing program will be performed as described by a general guidance procedure, "Guide for Conducting the Oconee Initial Test Program," to assure that commitments in Section 13 of the FSAR will be met. Recent revisions to the procedure were reviewed and discussed with Smith and Hampton during the inspection. The guide states that procedures will be reviewed prior to approval by various technical support groups such as the Duke Engineering Department and B&W. The master file contains the latest revision of each procedure with comments that have been received from the various groups, but there was no method of

insuring that comments not yet received would be considered in the final approved procedure. Smith stated that he would consider withholding his signature until the Test Work Group has completed its final review approximately one week prior to conducting the test. The final review will include checking that comments have been received and included.

The definition of changes that will be permitted without approval of the Station Review Committee and station superintendent had been revised and was considered unsatisfactory by the inspectors. The revision states that modifications that do not change the intent of the original procedure would be made by the station test coordinators (shift supervisors). The inspectors pointed out that such a definition could mean correcting a valve number or completely rewriting the procedure, depending upon the interpretation of the individual making the change. After a lengthy discussion, Smith agreed to reword the definition to more clearly state the type of changes to be allowed after a procedure has been approved for use. Smith notified Region II by telephone on March 3, 1971, that the test coordinators would only have the authority to correct typographical errors or make such changes as correcting valve lineups.

Documentation of the test program is contained in a master file maintained by the station superintendent's staff. A similar file will be maintained in the Steam Production Department general office at Charlotte, North Carolina. The master file contains a correspondence folder, test specifications, test procedures, system description, and operating procedure pertaining to a particular test. A working file is maintained for use by the supervisors conducting the test.

Any discrepancies observed during the performance of a test are recorded on a deficiency sheet attached to each test procedure. Correction of deficiencies is the responsibility of the station test coordinator assigned to the test and the corrective action is recorded on the deficiency sheet when complete. Retests will be performed as necessary to verify the adequacy of the corrective action. Hampton stated that a system will be devised to give the status of outstanding discrepancies as well as the status of the overall testing program.

2. Licensee's Test Results Evaluation Method (PI 5805.02)

Evaluation of test results falls into two categories, (a) onsite evaluation of results against essentially go-no-go acceptance criteria and (b) offsite evaluation of results against more general criteria

not amenable to a complete go-no-go acceptance criteria. Examples of tests in the latter category are functional tests, operational tests, physics tests, power escalation tests and reactor building tests. Review by offsite groups such as Duke Engineering Department, B&W, or Bechtel, as required, will include analysis, conclusions, and endorsement that the test is satisfactory or recommendations if the test is considered unsatisfactory. Copies of such reports will be sent to the station superintendent for his consideration prior to approving test results. Test results are not considered final until approved by the station superintendent. Safety-related tests and their results will be audited by the General Office Review Committee.

After a test is completed and the results approved, all material pertaining to that test will be filed for permanent documentation in the master file and the working file for the completed test will be discontinued. Each completed and approved test folder in the master file will contain the approved procedure, data collected, calculations, conclusions, final disposition of discrepancies, and approval of the completed test.

3. Review, Witnessing, and Evaluation of Tests (PI 5805.03)

a. Procedure Review

Comments on the following procedures were discussed with Smith, Hampton and McIntosh:

TP 1A 150 3	- Reactor Building Integrated Leak Rate Test
TP 1A 150 6 2	- Reactor Building Isolation Penumatic Leak Test
TP 1B 150 9 1	- Reference Vessel System Leak Test
TP 1A 204 5	- Reactor Building Spray Pump Engineered Safeguards Test
TP 1B 210 5	- Chemical Addition and Sampling Functional Test
TP 1A 230 5	- Soluble Poison Control Operational Test
TP 1B 250 3	- Low Pressure Service Water Functional Test
TP 1A 600 15	- Control Rod Drive System

The inspector's comments on specific procedures are on file in Region II and will be included in the tests unless the licensee disagrees with the comment, in which case the differences will be discussed with the inspector. Due to the repetitive nature of the weaknesses noticed in many of the procedures, the following general items were discussed and will be considered in preparing and revising all of the Oconee test procedures:

(1) Provide for signoffs of items of significance and approvals.

(2) Notify required groups prior to start and after completion of tests.

(3) Give specific acceptance criteria where applicable.

(4) Reference procedures by which prerequisite conditions are established.

(5) Provide concise, detailed and applicable limitations and precautions. Do not include prerequisites and procedural steps.

(6) Include critical path chart prerequisites.

During the discussion of the containment integrated leak rate test, the inspectors pointed out that the test must be conducted to assure that Technical Specification requirements will be met even though the specifications are still in preparation and have not been accepted by DRL. The procedure did not include a controlled leak rate test to verify instrument sensitivity following the full pressure test, but only after the reduced pressure test. The inspectors considered that the controlled leak rate test following each 24-hour integrated leak rate test was required by Section 15.4.3.1 of the proposed Technical Specifications. Smith did not believe that a controlled leak rate test was necessary at both pressures, but agreed to review the Technical Specification requirements before making his decision. Smith notified Region II by telephone on March 3, 1971, that the procedure would be revised to include a controlled leak rate test following the full pressure test as well as the reduced pressure test.

b. Flushing

McIntosh stated that approximately 10% of all flushing is complete. Marked up flow diagrams are maintained to indicate the status of the flushing program and to assure that clean systems are not contaminated by subsequent flushing operations.

Test procedures used during the reactor building spray system flush and the low pressure injection system flush were reviewed and found acceptable. Both procedures were properly approved for use and discrepancies observed during the tests were recorded on the deficiency sheet attached to

each procedure. Recorded test data met the specified acceptance criteria. All deficiencies had been cleared on the reactor building spray system flush and the test was approved by the station superintendent. All deficiencies had not been cleared on the low pressure injection system flush and the test had not been approved.

Each procedure folder contained a test log which had been maintained by the test coordinator during the performance of the test. Test logs were reviewed by the inspector for the reactor building spray and low pressure injection systems and all significant events in these logs were recorded on the deficiency sheets.

Space is provided on test cover sheets for the signatures of the test coordinator and vendor representatives witnessing a test, but all procedures were not signed by these persons. Hampton stated that the signatures were not required until a test is complete and the cover sheets in question involved tests that were not yet complete. Since the shift supervisor's signature is specified on the cover sheet and some tests will span several different shifts, Hampton was asked to consider having test witnesses sign the cover sheet as the test is performed. Hampton stated that this was probably closer to the intent of the test program guide and the present system would be reviewed to assure that all responsible test witnesses are identified.

c. Hydrostatic Testing

Hydrostatic testing usually follows system flushing and is approximately 10% complete. Marked up flow diagrams are maintained to indicate the overall status of the hydrostatic test program.

Procedures were originally written to test an entire system but completing the test was found to be impractical since piping installations were completed by area and not by system. A general procedure was devised to test portions of systems. This procedure was reviewed and discussed with Smith, Hampton and McIntosh.

The general procedure had been reviewed and approved by the Station Review Committee and station superintendent, but the actual procedure to be used in the field was not to be reviewed by the same group. Before the test is to be conducted, the test coordinator identifies the pressure boundary, makes

up valve checklists, and specifies test pressures and relief valve settings. The licensee was informed that this method of procedure preparation did not provide appropriate review and was contrary to Duke's procedure preparation guide which requires that procedures and major procedure modifications be resubmitted to the Station Review Committee and station superintendent for approval.

Smith and Hampton stated that the hydrostatic test procedure for a portion of a system would be reviewed and approved by the Station Review Committee and superintendent prior to the test. In addition to the proposed procedure, the committee will be supplied a flow diagram showing portions of the system already tested and the portion to be tested to insure that at completion the entire system has been satisfactorily tested.

H. Initial Core Loading Procedure Review (Attachment N-12, PI 5900)

The Initial Core Loading Procedure, OP 1502 4, was reviewed in accordance with PI 5900 and discussed with Smith, Hampton, and McIntosh. The licensee will rewrite this procedure and give consideration to the following items:

1. Method of determining that specified limitations are being met on a continuing basis.

2. Definition of "unexpected" as used in Step I.M.

3. Longer counting times than those specified in Step II.B.4.

4. Elimination of references to irradiated fuel.

5. Access control to reactor building.

6. Continuous recirculation of borated water.

7. Periodic checklists for maintaining satisfactory status of all equipment and events important to safety.

8. Final QC checks on core components.

9. Use of plant nuclear instrumentation.

10. Audible annunciation of source range monitors.

11. Recording of flux monitor signals.

12. Checkout of fuel handling equipment.

13. Specification of minimum pool level.

14. Status of manual containment isolation valves.

15. Operability of emergency boron addition system and conditions for using.

16. Water quality.

17. Limit on number of fuel assemblies that may be in route between the fuel storage area and the reactor vessel.

18. Critical path chart prerequisites.

19. Health Physics Group participation and personnel monitoring requirements.

20. Use of status boards and other appropriate records such as verification of proper enrichment, location, orientation and seating of components.

21. Method used to normalize count rate after source or detector relocation.

22. Job assignments giving consideration to the minimum permissible crew size and maximum allowable working hours.

23. Criteria and method of initiating containment evacuation as indicated in FSAR, Section 7.4.3.

24. Criteria for stopping fuel loading and authorization to continue.

25. Involvement of groups discussed in FSAR, Section 13.1.1.

26. Additional information on core makeup and detector locations during the loading process.

Generation of adequate fuel handling procedures is on the outstanding items list.

I. Control Rod Drive Mechanisms - Attachment L

1. Records Review (4905.05)

The inspector reviewed the records relating to the installation of the control rod drives mechanisms to the vessel head. A

detailed procedure had been prepared for each major step in the assembly of the mechanisms. The data sheets included spaces for signoffs for each assembly step and for recording the serial numbers associated with each mechanism. Revisions on the procedures were documented and approved prior to use. No deficiencies were noted in these procedures by the inspector.

The engineer responsible for the installation of the mechanisms maintains a daily log of events. This log indicated that the mechanisms which had been previously identified as having incorrectly installed torque tubes[1/] had been repaired. Other problems which had been encountered during the installation of the mechanisms were described and the resolution of the problems was documented. Since the licensee did not maintain a separate list of the problems and their resolution, it was necessary to review the log in detail to determine that a particular deficiency had been corrected. The inspector asked Hunnicutt if the licensee had considered keeping a list of the deficiencies in order to minimize the possibility of failing to correct them. Hunnicutt stated that he recognized the advantage of maintaining a separate list and that he would follow up on this comment.

2. Observation of Work (4905.06)

The licensee has established a clean area around the reactor vessel head and the assembly of the control rod drive mechanisms is being done in this area. When received the site, the components had been sealed in plastic bags packed in wooden crates. They had been inspected for shipping damage during the inspection of the guide bearings and torque tubes.[2/] During installation, the plastic bags were not removed until the components were placed in the clean area. At the time of this inspection, the pressure thimbles had been installed, the installation of the stator and position indication assemblies was essentially complete and the installation of the seismic restraints was in progress. The inspector did not note any deficiencies in the assembly process and plans no further action on this item at this time.

J. Electrical and Instrumentation

1. Control Rod Drive Controls - Attachment H

1/ CO Report Nos. 50-269/70-8 and 50-269/70-11.

2/ CO Report Nos. 50-269/70-3 and 50-269/70-11.

a. Review Cable of QC System (5205.04)

The inspector reviewed the requirements for the installation of the cables and wireways for the control rod drive system. The requirements for these items are identical to those for other electrical and instrumentation systems previously inspected.1/ Only the cable for the scram magnets are considered by the licensee to be safety-related and requiring special routing and separation. The licensee's use of color coded cable simplifies the verification of separation of redundant circuits. Power cables are installed in a single layer in the tray with maintained separation and loading of these cables must conform to IPCEA requirements. Wireways for instrumentation cables are not permitted to be loaded above the side rails. No deficiencies were noted and the inspector plans no further action on this item at this time.

b. Followup Observation of Work (5205.06)

Essentially all of the cables for the control rod drive system have been installed in the auxiliary building and in the reactor building. The cables and trays for the control rod drives were installed in accordance with the licensee's procedures and the proper materials were used. A QC inspector is assigned to stay with each cable pulling crew to insure that the installation is made in accordance with approved drawings and procedures. During a tour of the installation, the inspector observed that a QC inspector was with each crew.

2. Nuclear Instrumentation - Attachment H

a. Records Review and Observations of Work (5105.05 and .06)

The inspector reviewed the nuclear instrumentation records with Price. These records had been inspected previously and found to be deficient.2/ The licensee has now completed a review of the documentation and corrected the deficiencies which had been found. The nuclear instrumentation chassis are mounted on the main control console. A review of this installation indicated that the equipment had been installed in accordance with approved drawings. The inspector plans no further action on these items at this time.

1/CO Report No. 50-269/70-10.

2/CO Report No. 50-269/70-12.

b. Cables (5205.04 and .06)

The QC and installation requirements for the nuclear instrumentation cable and wireway installation are the same as for the control rod drives as discussed in Section J.1. The inspector did not observe any deficiencies in the QC requirements or the installation and plans no further action on these items at this time.

3. Pressurizer Level Instrumentation - Attachment H

Review of Cable QC System (5205.05)

The requirements for the cable and wireway installation are the same as for the control rod drive controls, Section J.1. The inspector did not observe any deficiencies in these requirements and plans no further action on this item at this time.

4. Uninterrupted QC Power System - Attachment I

a. Review of Cable QC System (5205.04)

The requirements for the cable and wireway installations are the same as for the control rod drive controls, Section J.1. The inspector did not observe any deficiencies in these requirements and plans no further action on this item at this time.

b. Followup Observation of Work (5205.06)

The installation of the cables and wireways associated with the uninterrupted a.c. power system was reviewed by the inspector and appeared to be in accordance with the QC requirements. The inspector plans no further action on this item at this time.

5. Battery System - Attachment I

a. Review of Cable QC System (5205.04)

The requirements for the cable and wireway installation for the station control batteries are the same as for the control rod drive control, Section J.1. The inspector did not observe any deficiencies in these requirements and plans no further action on these items at this time.

b. Followup Record Review (5205.05.a.1 and a.2)

The records relating to the NDT requirements for cables had been previously reviewed and no deficiencies had been noted.[1] The licensee has provisions for the quarantine of nonconforming cable but, to date, no cable has been received at the site.

c. Followup Observations of Work (5205.06)

The inspector reviewed the battery cable and wireway installation. No deficiencies were noted and the inspector plans no further action on this item at this time.

K. Miscellaneous

1. Valve Numbering System

The inspector advised Smith and Hampton that during a review of the test procedures, he had noticed that the valve numbers in the procedures did not conform to those in the FSAR. Smith stated that the numbers in the test procedures and the operating manual were assigned by the Operating Department and the numbers on the valve identification tags would conform to these. Numbers in the FSAR had been assigned by either B&W (primarily nuclear steam supply system) or the Duke Engineering Department (primarily secondary systems). The valves on the system design drawings had originally been identified by B&W or by Duke Engineering or by both. Subsequently, the drawings had been revised to also include the Operating Department numbers. The numbering system used by the Operating Department is similar to that used by B&W in that both use a system designation followed by a component number. The high pressure injection system valves in both cases are designated "HP." The inspector pointed out that in having multiple designations on the drawings, the possibilities of operator errors were increased. In addition, in communicating with other organizations, the multiple numbers could lead to confusion and errors. Smith stated that he would determine what could be done to minimize or eliminate the type occurrences pointed out by the inspector.

2. Fuel Handling Equipment

In reviewing the installation of the fuel handling cranes, the inspectors observed that the drip pans under some of the bearing

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[1] CO Report No. 50-269/70-10.

boxes appeared to be too small to hold the lubricant that might leak into them. One of the drip pans appeared to be tilted downward at one corner, further reducing its capacity. Because of the positions of the cranes over the fuel canals, it was not possible to make a complete inspection nor an accurate evaluation of the adequacy of the pans. This item was discussed in the management interview. Hunnicutt stated that he would review the designs to determine if the pans were adequately sized and at the necessary locations. Smith stated that he would verify that procedures were developed that would require periodic inspection and maintenance of the pans.

3. Fuel Transfer Tubes

Hunnicutt advised the inspector that the expansion bellows on the fuel transfer tube which had been previously reported as damaged had been replaced with a new bellows. The inspector plans no further action on this item at this time.

Attachments:
Exhibits A and B

LICENSEE Duke Power Company

FACILITY Oconee Station No. 1

DOCKET & LICENSE NOS. 50-269, CPPR-33

REACTOR OUTSTANDING ITEMS

IDENTIFIED	ITEM	CLOSED
1. 68-2, 3/5/68, NC	Concrete test cylinder breaks below specs	68-3, D.5., 6/19/68
2. 68-3, 6/19/68, NC	Unauthorized revision to Cadweld specifications	68-4, Summary, 9/25/69
3. 68-3, 6/19/68, NC	Failure to provide concrete inspector	68-4, Summary, 9/25/69
4. 68-4, 9/25/68 NC	Failure to properly test Cadweld splices	69-1, Summary, 1/6/69
5. 69-8, 9/9/69, NC	Failure to properly qualify weld procedures	69-9, G, 11/3/69
6. 69-8, 9/9/69, NC	Failure to properly qualify weldors	69-9, G, 11/3/69
7. IEB, 4/11/69	Procedure for repair of arc strikes not available	70-5, Summary, 4/27/70
8. CDN, 1/8/70	NDT of core flooding valves	Memo, WCS to HQ, 2/2/70
9. 70-1, 1/6/70, NC	Welding and NDT deficiencies, CDN issued	Memo, WCS to HQ, 3/26/70
10. Bingham 69-1, 12/9/69, NC	Main coolant pump discrepancies	Memo, WCS to HQ, 4/21/70
11. 70-4, 4/27/70, NC	Low strength concrete	Memo, WCS to HQ, 8/7/70
12. IEB, 5/1/70	Pressure vessel safe ends	Memo, WCS to HQ, 8/5/70
13. 70-6, 5/25/70, NC	Tendon stressing discrepancies	Memo, WCS to HQ, 8/7/70
14. 70-8, 8/3/70, NC	Tendons and stress gages	Memo, WCS to HQ, 10/8/70
15. 70-8, 9/1/70, UN	Fissures in primary coolant pipe cladding	FSAR, Amend.24, 12/17/70
16. IEB, 9/11/70, UN	a. Determination of safety system response to axial power imbalances b. Availability of in-core detectors	

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REACTOR OUTSTANDING ITEMS

IDENTIFIED	ITEM	CLOSED
	c. Measurements of flow and temperature during initial operation	
	d. Verification of bypass flow	
	e. Verification of axial peak effects on DNBR	
	f. Data during startup for single loop, two pump operations	
	g. Inspection of reactor internals after completion of preoperational tests	
	h. Field test of steam generator	
	i. Low strength concrete and omitted tendons	Memo, WCS to HQ, 10/8/70
	j. Penetration room valves	70-12, Summary 12/1/70
	k. Strain gauge failures	Memo, WCS to HQ, 10/8/70
	l. HP and LP injection system startup times	
	m. Core flooding tank MO valve	
	n. Reactor building spray pump performance	
	o. Condenser cooling water crossover header valve	
	p. Spent fuel accident filters	
	q. Administrative control of MCP startup	
	r. Flow tests per 200/12 and 200/13	
	s. Flow distribution chart	

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REACTOR OUTSTANDING ITEMS

	IDENTIFIED	ITEM	CLOSED
17.	70-2, 2/19/70, UN	Vendor NDT records for safeguards systems cables	70-11, F, 10/26/70
18.	70-4, 3/23/70, UN	Verification of separation of transducer tubing	
19.	70-8, 8/3/70, UN	Control rod drive guide bushings and torque tubes	71-3I, 2/24/71
20.	70-8, 8/3/70, UN	Completion of HP facilities	
21.	70-8, 8/3/70, UN	Completion of HP procedures	
22.	70-8, 8/3/70, UN	Completion of HP personnel training	70-12, Summary 12/1/70
23.	70-8, 8/3/70, UN	Crane load test	71-1, 1/4/71
24.	70-8, 8/3/70, UN	Verify that test procedures are properly revised and approved when changes are required	
25.	70-8, 8/3/70, UN	Verify that analysis of containment is made	FSAR, Amend. 24
26.	70-8, 8/3/70, UN	Adequate fuel handling procedures	
27.	70-8, 8/3/70, UN	Main steam pipe hangers	
28.	70-9, 9/1/70, UN	Steam generator skirt adapter indications	
29.	70-9, 9/1/70, UN	HP injection pump QC records	70-11, C, 10/26/70
30.	70-9, 9/1/70, UN	Basis for particle size in flushing procedures	70-11, G, 10/26/70
31.	70-9, 9/1/70, UN	Protection of instrumentation during hydro test	
32.	70-10, 9/28/70, UN	Fuel transfer tube expansion joint replacement	71-3L, 2/24/71
33.	70-10, 9/28/70, UN	Routing of cables exterior to cable trays	Memo, WCS to HQ 1/18/71

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REACTOR OUTSTANDING ITEMS

	IDENTIFIED	ITEM	CLOSED
34.	DRL Rpt. No. 1, 7/24/70, UN	Installation of additional environmental monitoring equipment	
35.	DRL Rpt. No. 1, 7/24/70, UN	Vent valve replacement test	
36.	DRL Rpt. No. 1, 7/24/70, UN	Strong motion accelerometer installation	
37.	DRL Rpt. No. 1, 7/24/70, UN	Penetration room flow indication and adjustment	
38.	DRL Rpt. No. 1, 7/24/70, UN	Instrumentation bypass keys	Tech Specs Change 12/70
39.	DRL Rpt. No. 3, 9/15/70, UN	Internals vibration test	
40.	DRL Rpt. No. 3, 9/15/70, UN	Core flooding tank valves	
41.	70-10, 9/28/70, UN	Hydrostatic test pressures	71-1, 1/4/71
42.	70-11, 10/26/70, UN	Cleaning reactor coolant system piping and equipment	71-2, 1/25/71
43.	70-11, 10/26/70, UN	Sensitized stainless steel in reactor coolant pump discharge piping	71-1, 1/4/71
44.	IEB, 12/22/70	Reactor coolant pump tests	
45.	IEB, 10/30/70	Safety injection system testing	
46.	70-12, 12/1/70 UN	Vibration testing - equipment and piping	
47.	70-12, 12/1/70 NC	Location of station batteries	
48.	70-12, 12/1/70 NC	Nuclear instrumentation vendor tests	
49.	70-12, 12/1/70 NC	Electrical QC data packages	

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REACTOR OUTSTANDING ITEMS

	IDENTIFIED	ITEM	CLOSED
50.	70-12, 12/1/70 UN	ITE relays	
51.	70-12, 12/1/70 UN	Heater and heat tracing tests	
52.	70-12, 12/1/70 UN	Control rod drive cooling system tests	
53.	70-12, 12/1/70	Containment and auxiliary building vent system filters	
54.	FSAR, Amend 25 UN 12/30/70	Installation of strain gages	
55.	71-2, 1/25/71 UN	Keowee battery room ventilation	
56.	71-2, 1/25/71 UN	Switchyard battery blocking diode tests	
57.	71-2, 1/25/71 UN	Remove temporary steam line at 4 kv switchgear	
58.	71-2, 1/25/71 UN	Controlled leak rate tests	
59.	71-2, 1/25/71 UN	Penetration room elevator opening	
60.	71-2, 1/25/71 UN	Verification of separation of redundant circuits	
61.	71-2, 1/25/71 UN	Cleanup of cable trenches	
62.	71-2, 1/25/71 UN	Adequacy of leak rate tests	
63.	71-2, 1/25/71	Replacement of feedwater pipe	
64.	71-3, 2/24/71 NC	Cleanliness of reactor vessel and internals	

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FACILITY Oconee Station NO. 1

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REACTOR OUTSTANDING ITEMS

IDENTIFIED	ITEM	CLOSED
65. 71-3, 2/24/71 UN	Drip pans on fuel handling cranes	
66. 71-3, 2/24/71 UN	Containment leak rate tests	

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BOLT CIRCLE BETWEEN MATING FLANGES OF THE CORE SHIELD & CORE BARREL SHOWING LOCATION OF TRAPPED POLYETHYLENE

BOLT NUMBERING SYSTEM

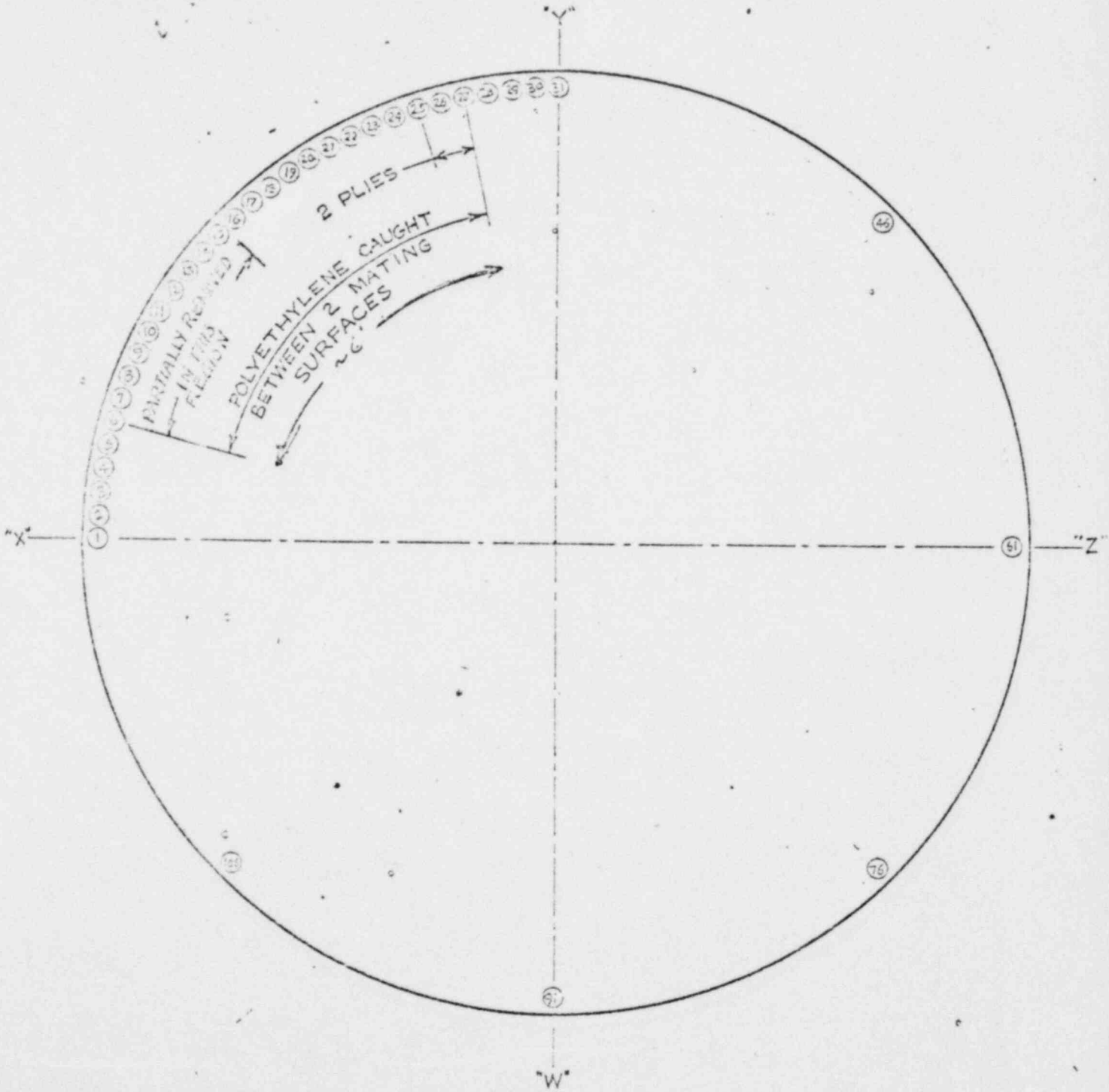

Exhibit B
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