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A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

December 19, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: RO Inspection Report
50-269/74-10
50-270/74-8
50-287/74-11

Dear Mr. Moseley:

We have reviewed RO Inspection Report Nos. 50-269/74-10, 50-270/74-8 and 50-287/74-11. Duke Power Company does not consider the information contained in this report to be proprietary.

With regard to the specific concerns identified in your letter, please find attached our responses. In addition to the information provided in the attached, we have taken actions to improve the effectiveness of our management control systems. Most significant of these actions is the recent reorganization within the Steam Production Department. This reorganization has served to clarify areas of responsibility, and the authority commensurate thereto, both within the Steam Production Department General Office and within the station organization. Also, the position of Manager, Nuclear Production has been established. The nuclear station Managers report to the Manager, Nuclear Production who, in turn, reports to the Vice President, Steam Production. This change has served to strengthen line management control and involvement with regard to nuclear station operations.

To assist management in effectively discharging its responsibilities, a program has been developed to maintain and periodically distribute a written compilation of all outstanding commitments relating to Oconee Nuclear Station. In this manner, it can be assured that proper and timely actions are taken with regard to items identified.

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It is felt that actions taken to date, and which are continuing, have served to significantly improve the effectiveness of the management control system with regard to the operation of Oconee Nuclear Station.

Very truly yours,

A.C. Thies

A. C. Thies

ACT:vr

RESPONSES TO RO INSPECTION REPORT
50-269/74-10, 50-270/74-8, 50-287/74-11

December 19, 1974

ITEM I.A.1.b:

Technical Specification 6.4.1 requires that procedures be developed and adhered to for activities affecting quality. Section 3.2.2.5 of the DPC Steam Production Department (SPD) Administrative Policy Manual (APM) for Nuclear Stations requires each station to establish a periodic testing schedule to assure that all safety-related testing is performed and properly evaluated in a timely manner.

Contrary to the above, a periodic testing schedule has not been developed and implemented as specified by paragraph 3.2.2.5 of the APM. (Details II, paragraph 4.a)

RESPONSE:

A computer program has been prepared for the purpose of scheduling periodic tests. This program is presently available for station use and the information for a number of periodic tests has been input to the program data base. Those tests are currently being scheduled thereby. Information for other periodic tests with a frequency of greater than weekly is currently being input to the program data base. This task should be completed by January 15, 1975, and, subsequent to that date, the program will be used for scheduling such periodic tests.

ITEM I.A.1.a:

Technical Specification 6.1.2.1.d.5 requires the Station Review Committee (SRC) to review proposed safety-related changes or modifications to the station design.

Contrary to the above, the SRC did not review station modification request 0-300-S, which changed the steam generator water level control from 95% to 50% for natural circulation cooling. (Details I, paragraph 3.a)

RESPONSE:

As noted in the inspection report, Details I, paragraph 3.a, the modification to change the steam generator level control setpoint for natural circulation cooling from 95% to 50% was recommended by the Babcock and Wilcox Company (B&W) in a letter dated July 24, 1974. The Station Review Committee (SRC) reviewed the safety implications of the recommended change on July 31, 1974. It is considered that this review of the safety implications of, and concurrence with, the proposed modification was adequate.

With regard to SRC review of future station modifications, new policies for the control of modifications are to be incorporated into the Steam Production Department's "Administrative Policy Manual for Nuclear Stations" on December 20, 1974, for implementation on January 1, 1975. These policies require that the SRC review each proposed modification to safety-related structures, systems and components subsequent to design of the modification and prior to implementation of the modification. These policies also require that the fact that such a review has been performed be verified prior to implementation of the subject modification.

ITEM I.A.1.c:

Technical Specification 6.4.1 requires that the station be operated and maintained in accordance with approved procedures, and Technical Specification 6.1.2.1.d requires the SRC to review safety considerations, unusual events, violations of Technical Specifications and new procedures that affect nuclear safety.

Contrary to the above:

(1) No documentary evidence could be located to verify that the following procedures had been reviewed by the SRC.

(a) PT/1/A/201/3, Core Flood System

(b) PT/0/A/204/9, Reactor Building Spray System Engineered Safeguards Tests

(c) PT/0/A/202/11, High Pressure Injection System Performance Test

(Details II, paragraph 2.d(2))

(2) An improper valve lineup resulting in an unplanned drop in the level of the Borated Water Storage Tank for Unit 1, on August 26, 1974, was not reviewed for safety considerations by the SRC. (Details III, paragraph 2.e)

RESPONSE:

(1) Subsequent to the exit interview of November 13, 1974, a review of those procedures noted as not having been reviewed by the Station Review Committee (SRC) was conducted. This review determined that documentary evidence is available with regard to SRC review of two of the noted procedures. PT/1/A/0201/03, Core Flood System, was reviewed by the SRC on October 17, 1972, as indicated by a copy of the procedure in the Master File. PT/0/A/0202/11, High Pressure Injection System Performance Test, was reviewed by the SRC on November 18, 1972, as shown by SRC minutes.

No written evidence can be found showing that PT/0/A/0204/09, Reactor Building Spray System Engineered Safeguards Test, was reviewed prior to its approval on March 10, 1973, as required by Technical Specification 6.1.2.1.d. However, a complete revision to this procedure was reviewed by the SRC on July 25, 1974, prior to the initial utilization of this procedure. It is felt that since the date of this incident, i.e., March, 1973, that the control of the preparation, review and approval of procedures has continually improved. This is evidenced by the fact that, although PT/0/A/0204/09 was not originally reviewed by the SRC, the July, 1974 revision to the procedure did receive SRC review.

(2) The incident cited occurred on August 26, 1974. On August 27, 1974, an investigation was conducted to determine if Technical Specifications had been violated and it was concluded that a Technical Specifications violation had not occurred. Based on this conclusion, this incident was

not classified as an Abnormal Occurrence or an Unusual Event and, therefore, was not reviewed by the SRC. Since the date of this incident, however, management has placed increased emphasis on the importance of reviewing incidents promptly for their safety significance and reportability to the AEC as an Abnormal Occurrence or Unusual Event - see response to Violation I.A.1.d. Incidents similar to the one noted are currently being reviewed by the SRC as required. It is believed that, as a result of these corrective actions, present methods are adequate to assure future compliance with the Technical Specifications.

ITEM I.A.1.d:

Technical Specification 6.1.2.1.d.3 requires the SRC to review all unusual events. Technical Specification 6.6.2.1.b requires that unusual events be reported to the Directorate of Regulatory Operations within 30 working days.

Contrary to the above:

(1) The SRC did not review a condition that permitted both doors of the Unit 2 reactor building personnel hatch to be opened at the same time, which could have resulted in a loss of containment integrity. (Details III, paragraph 1a)

(2) A report of the unusual event was not submitted to the Directorate of Regulatory Operations. (Details III, paragraph 1.a)

RESPONSE:

The deficiency noted in Details III, paragraph 1, of the inspection report resulted from the failure to note the interlock failure in the shift supervisor's log. The incident was noted in the Unit 2 reactor operations log, but no entry was made in the shift supervisor's log due to the fact that (1) existing unit conditions (cold shutdown) did not require containment integrity, (2) timely maintenance action, and (3) the administrative controls imposed (a man was stationed inside the hatch to control traffic and prevent both doors from being opened simultaneously) during the time the interlock was inoperable. If the incident had been noted in the shift supervisor's log, it would have been brought to the attention of the Technical Services Engineer, who is more knowledgeable of the Oconee FSAR, and a better determination of the reportability of the incident would have been made.

As noted in Details III, paragraphs 1 and 2, the Technical Services Engineer reviews the shift supervisor's log to identify any incidents or conditions which warrant further investigation. The results of this investigation, including recommended corrective action, are summarized in an Incident Investigation Report. These reports are reviewed by the Station Review Committee (SRC), station management, and the Licensing Unit in the General Office. If it is determined that the incident is reportable to the AEC under the definitions of Abnormal Occurrence or Unusual Event contained in the Technical Specifications, the Incident Investigation Report, with SRC and station Manager comments, is forwarded to the Licensing Unit for preparation of a report to the AEC.

ITEM I.A.1.e:

Technical Specification 6.4.1 requires that operation of the station be conducted in accordance with procedures appropriate to the circumstances.

Contrary to the above:

(1) The emergency procedure stating action to be taken in the event of a loss of coolant flow was not changed, following a change of the steam generator level setpoints. (Details I, paragraph 3.a)

(2) The controlling procedure for unit startup, OP/2/A/1102/1, was not revised to reflect Change No. 6 (issued May 29, 1974) to the technical specifications. (Details III, paragraph 2.a)

(3) Procedure PT/0/B/200/5, Running Reactor Coolant Pump Motors, was inadequate, in that the procedure permitted installation of jumpers on safety-related equipment and did not specify removal of the jumpers following completion of testing. (Details III, paragraph 2.c)

RESPONSE:

With regard to the three items listed, the following corrective action has been taken:

(1) Change 1 to EP/0/A/1800/06 was made on December 4, 1974 to change the OTSG level setpoint from 95% to 50%.

(2) Change No. 20 to OP/2/A/1102/1 revised OP/2/A/1102/01, the controlling procedure for unit startup, to reflect Change No. 6 to the Technical Specifications. Change No. 20 was approved on October 18, 1974.

(3) Change No. 2 to PT/0/B/0200/05, dated October 18, 1974, corrects the deficiency noted in this procedure.

To prevent recurrence of incidents similar to those noted above, a more complete review of station modifications and Technical Specification changes will be performed, and procedure changes will be made in a timely manner. Specifically with regard to station modifications, new policies for the control of modifications are to be incorporated into the Steam Production Department's "Administrative Policy Manual for Nuclear Stations" on December 20, 1974, for implementation on January 1, 1975. These policies will require that verification of the completion of a modification include verification that any required procedure changes have been made.

ITEM I.A.1.f:

Technical Specification 6.1.1.5 requires that a training program be established for all personnel, including security procedures, which meet the provision of ANSI N18.1.

Contrary to the above, training and retraining of personnel in security procedures has not been effected for all station employees. (Details I, paragraph 4.d)

RESPONSE:

Employees have previously received training in station security although a formal training program had not been established. In order to improve employee training in station security requirements, a formal training program has been developed, and will be presented to all station employees during January, 1975. Personnel employed after February 1, 1975 will receive training in the area of security procedures as part of the orientation program for new employees. Retraining of all Oconee employees in security procedures will be conducted annually.

ITEM I.A.1.g:

Technical Specification 6.1.2.2.b states that the activities of the Nuclear Safety Review Committee (NSRC) shall be guided by a written charter, and lists the activities that must be contained within the charter.

Contrary to the above, the following activities, required to be contained in the NSRC charter, are not discussed in the by-laws which serve as a charter.

(1) Subjects within the purview of the committee.

(2) Identification of management position to which the group reports.

(3) Provisions for assuring that the committee is kept informed of matters within its purview.

RESPONSE:

The By-Laws (Charter) of the Nuclear Safety Review Committee are presently being revised to fully reflect the requirements of the current Technical Specifications. The By-Laws will be revised within one month following each future revision of the Technical Specifications affecting the Nuclear Safety Review Committee. It is expected that the current revision of the By-Laws will be completed by February 1, 1975.

ITEM I.A.1.h:

Technical Specification 6.1.2.2.i requires the NSRC to review abnormal occurrences and unusual events.

Contrary to the above, the NSRC did not review the abnormal occurrence (incorrectly reported as an unusual event), relating to the failure of the Unit 2 Low Pressure Injection Valve. (Details II, paragrapn 5.g(1))

RESPONSE:

The Nuclear Safety Review Committee (NSRC) reviewed the unusual event relating to the failure of a Unit 2 low pressure injection valve as Oconee Incident Report B-174 during their review of the July 15, 1974 minutes of the Station Review Committee (SRC). This review is documented as Item 7 of the September 27, 1974 Minutes of the NSRC. Prior to the September 27, 1974 meeting of the NSRC, the Chairman distributed Unusual Event Report No. UE-270/74-3, which describes this valve failure to all members for review. Members then had the opportunity to ask questions or make comments, as appropriate, during the September 27, 1974 meeting.

ITEM I.A.1.i:

Criterion XVI of Appendix B to 10 CFR 50 requires that prompt corrective action to preclude recurrence of nonconforming items be taken.

Contrary to the above, prompt corrective action was not taken of deficiencies identified in an internal QA audit performed on June 10, and in a reaudit of August 16, 1974, concerning document control and tagging of equipment. The QA records do not reflect that corrective action to prevent recurrence had been taken as of October 23, 1974. (Details I, paragraph 6.a)

RESPONSE:

Quality Assurance Department personnel conducted Level II audit 0-74-1 during the week of June 10, 1974, and issued a report thereof on June 21, 1974. On July 26, 1974, Mr. J. E. Smith, Manager, Oconee Nuclear Station, responded to the concerns noted in the audit report. Quality Assurance Department personnel conducted a reaudit on August 16, 1974, the report of which was issued on August 20, 1974. The report of the reaudit did not specifically request a response and, therefore, none was prepared at that time. A response to the reaudit has since been transmitted by Mr. Smith to the Steam Production Department General Office. This response is currently being processed and will be forwarded to the Quality Assurance Department by January 1, 1975.

On November 5, 1974, Mr. William O. Parker, Jr., Vice President, Steam Production, established a policy for future handling of Level II audit reports. Also, on December 20, 1974, a revision to the Steam Production Department's "Administrative Policy Manual for Nuclear Stations" is to be issued which will include requirements for the proper and timely correction of identified deficiencies. As a result of these actions, incidents of the type noted should not recur.

ITEM I.A.1.j:

Criterion XVIII of Appendix B to 10 CFR 50 requires that audits be performed by appropriately trained personnel in the areas being audited.

Contrary to the above, none of the Level I QA auditors performing audits of the operating facilities have received appropriate training or are experienced in reactor operations. (Details I, paragraph 6.c)

RESPONSE:

During 1975, selected Quality Assurance Department personnel involved in performing Level I audits will receive formal training in reactor operations. This training will consist of classroom instruction and discussion, supplemented with appropriate video tapes, on nuclear station systems and procedures. It is anticipated that each participant in the training will receive approximately 100 hours of training.

ITEM I.A.1.k:

Technical Specification 6.1.2.1.a requires the superintendent to appoint an on-site review committee (SRC) consisting of at least five members of the station supervisory staff. Technical Specification 6.1.2.1.c specifies that a quorum shall consist of the chairman plus two members.

Contrary to the above, SRC meetings were conducted without the required quorum of supervisory members present. (Details II, paragraph 2.a)

RESPONSE:

The Technical Specifications 6.1.2.1a requires that the Manager appoint a Station Review Committee of at least five members of the Station Supervisory Staff. It requires representation from Operations and Technical Services and that personnel with expertise appropriate to items being considered participate on the Committee.

The initial committee was made up of the Station Assistant Superintendent as Chairman, the Operating Engineer, the Technical Support Engineer, the Maintenance Supervisor, and the Health Physics Supervisor. With the initiation of station operation and development of the expertise of other station personnel, the number of individuals in the station supervisory organization qualified to serve on this committee has increased. The Manager has performed a continuing review of personnel in the station supervisory organization as regards their qualifications and listed those who are qualified to serve on a basic five man committee. The Intrastation Letter is a list of supervisory personnel qualified to participate on this committee and is not intended to infer that this is a thirty-three man committee.

Normally, meetings of this committee are scheduled with five members present with representation from the Operations and Technical Services Groups participating. Where items of a particular discipline are being considered, a member of the supervisory organization in the area being considered is included in the committee meeting. Special called meetings require at least a quorum present, with participation of a member of the supervisory organization who has expertise in the area being considered.

As regards Section 12A.5 of the FSAR, this is only a partial listing of the station supervisory organization, and it is not required or intended that all station supervisory personnel be identified. Section 12A.5 is intended to establish that there are in the station organization adequate supervisory personnel with technical capabilities to safely operate and maintain the station in keeping with the safety review and in keeping with ANSI 18.1.

The Intrastation Letter is being revised to clarify any misunderstanding that may have resulted.

ITEM I.A.2.a:

Technical Specifications 6.5.2.d and 6.5.2.h require station maintenance histories for safety related structures, systems and components, including periodic testing records, be prepared and maintained for a minimum of six years. Technical Specification 6.4.1 states that the station shall be operated in accordance with approved procedures.

Contrary to the above:

(1) No documentation could be provided to show that the replacement of pressurized safety valves was done in accordance with an approved procedure and the maintenance history and periodic test records of the installed safety valves could not be located. (Details II, paragraph 4.b(1))

(2) Records of the periodic tests of the Core Flood System could not be located. (Details II, paragraph 4.b(2))

RESPONSE:

(1) As indicated in the inspection report, Details II, paragraph 4.b(1), documentation is available which verifies that the Oconee 1 pressurizer safety valves were properly adjusted by Dresser Industries, the valve manufacturer. Discussion with the supervisor involved in the replacement of the valves determined that a copy of PT/0/A/0200/29 was used and completed at that time. During the subsequent review and audit process, however, the completed procedure was misplaced. Consequently, the periodic test records and maintenance histories were not updated.

Reorganization of the station Maintenance Group in July of this year introduced a Planning Section, which has the responsibility for maintaining maintenance histories. Document transmittal controls are now in effect which require a document receipt signature when documents are received for permanent storage. This method of handling should preclude similar occurrences in the future.

(2) OP/ /A/1102/01, Controlling Procedure for Unit Startup, has been revised to require that PT/ /A/0201/03, Core Flood System, be completed during each unit startup. This change will assure that the necessary testing is adequately documented.

Ltr to Duke Power Company from N. C. Moseley dated NOV 27 1974
RO Inspection Report Nos. 50-269/74-10, 50-270/74-8
and 50-287/74-11

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