830 Power Building

TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401

Central File

August 8, 1977

50-259
260
296

Mr. Norman C. Moseley, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II - Suite 1217
230 Peachtree Street, NW.
Atlanta, Georgia 30303

Dear Mr. Moseley:

This is in response to F. J. Long's July 19, 1977, letter, RII:JEO 50-259/77-8, 50-260/77-8, 50-296/77-8, which transmitted for our review an IE Inspection Report (same number). We have reviewed that report and do not consider any part of it to be proprietary.

Very truly yours,

H G Harris for

J. E. Gilleland
Assistant Manager of Power

B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N.W. SUITE 1217
ATLANTA, GEORGIA 30303

JUL 19 1977

In Reply Refer To:
RII:JEO
50-259/77-8
50-260/77-8
50-296/77-8

Tennessee Valley Authority
Attn: Mr. Godwin Williams, Jr.
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

This refers to the inspection conducted by Mr. J. E. Ouzts of this office on June 30 - July 1, 1977, of activities authorized by NRC License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Unit 1, 2 and 3 facilities, and to the discussion of our findings held with Mr. J. G. Dewease at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the attached inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of this inspection, no items of noncompliance were disclosed.

The licensee representative agreed to review the method he is currently using for deriving the feedwater signal from the feedwater flow instrument for use in the computer OD-3 program for core thermal power measurement and to report to NRC whether the current method was intended to be used in the original design.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the attached inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office requesting that such information be withheld from public disclosure. If no proprietary information is identified, a written statement to that effect should be submitted. If an application is submitted, it must fully identify the bases for which information is claimed to be proprietary. The application should be prepared so that

information sought to be withheld is incorporated in a separate paper and referenced in the application since the application will be placed in the Public Document Room. Your application, or written statement, should be submitted to us within 20 days. If we are not contacted as specified, the attached report and this letter may then be placed in the Public Document Room.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Very truly yours,

F. J. Long, Chief
Reactor Operations and Nuclear
Support Branch

Attachments:
RII Inspection Report Nos.
50-259/77-8, 50-260/77-8
and 50-296/77-8

cc: J. G. Dewease
Plant Superintendent
Box 2000
Decatur, Alabama 35602

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N.W. SUITE 1217
ATLANTA, GEORGIA 30303

Appendix A

Report Nos.: 50-259/77-8, 50-260/77-8 and 50-296/77-8

Docket Nos.: 50-259, 50-260 and 50-296

License Nos.: DPR-33, DPR-52 and DPR-68

Licensee: Tennessee Valley Authority
818 Power Building
Chattanooga, Tennessee 37401

Facility Name: Browns Ferry Units 1, 2 and 3

Inspection at: Browns Ferry site, Athens, Alabama

Inspection conducted: June 30 and July 1, 1977

Inspector: J. E. Ouzts

Reviewed by: R D Martin 7/18/77
R. D. Martin, Chief
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch
Date

Inspection Summary

Inspection on June 30 - July 1, 1977 (Report Nos. 50-259/77-8, 50-260/77-8 and 50-296/77-8)
Areas Inspected: Routine, unannounced inspection of plant surveillance program, procedures and schedule pertaining to operations, emergencies, maintenance and administration and review of APRM gain adjustment factor (GAF) data and the method used to obtain the feedwater flow signal for the process computer OD-3 program - Thermal Power Measurement. The inspection involved 16 inspector-hours on site by one NRC inspector.
Results: Of the areas inspected no apparent items of noncompliance or deviations were identified.

DETAILS I

Prepared by: J. E. Ouzts 7/18/77
J. E. Ouzts, Reactor Inspector Date
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

Dates of Inspection: June 30 - July 1, 1977

Reviewed by: R. D. Martin 7/18/77
R. D. Martin, Chief Date
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

1. Persons Contacted

*J. E. Dewease, Plant Superintendent
*L. Blankner, Nuclear Engineer
*T. Bragg, QA Supervisor
P. Crabb, Modifications Coordinator
R. McGee, SI Coordinator
Various Shift Engineers and Plant Operators

*Denotes those present at the exit interview.

2. Licensee Action on Previous Inspection Findings

None

3. Unresolved Items

(1) The APRM gain adjustment factors (GAFs) are higher than those given in Section 7 of the FSAR under the conditions they were taken. The inspector will discuss his findings with NRC management, and in addition will analyze more APRM dated prior to determing whether or not the maximum GAFs permitted are satisfactory with the present setpoints. This is identified as Unresolved Item 77-8/I-1.

(2) The inspector questioned the accuracy of the current method of obtaining the feedwater flow signal for the OD-3 program Core Thermal Power, as to the most accurate method available. The licensee will review the current method to determine if it is the method that was intended to be used, and will report to NRC on his findings, along with justification for continuing using the current method, if used. This is identified as Unresolved Item 77-8/I-2.

4. Exit Interview

The inspector met with the licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on July 1, 1977. The inspector summarized the purpose and scope of the inspection and findings. No items of noncompliance or deviations were identified.

5. Units 1, 2 and 3 Procedures

The inspector reviewed the following procedures and records to verify that reviews, approvals and changes were in accordance with the technical specifications and that changes reflected technical specification revisions and were in conformance with 10 CFR 50.59(a) requirements and that records of changes to procedures pursuant to 50.59(a) were being maintained:

(a) Nine operating procedures as required by technical specifications and identified in Paragraph C of Regulatory Guide 1.33.

(b) Six emergency procedures identified in Paragraph F of Regulatory Guide 1.33.

(c) Eight maintenance procedures associated with systems whose operating procedures were reviewed per 5(a) above.

(d) Three administrative procedures identified in Paragraph A of Regulatory Guide 1.33.

(e) Record of technical specifications changes recommended by plant personnel (24 documents reviewed for the period of February 1976 to June 1977).

(f) Correspondence from Power Production Department relating to technical specification changes (reviewed documents for the period between June 1976 and June 1977).

(g) Four Safety-related work packages to verify requirements to revise affected procedures and to evaluate the modification for safety evaluation requirements.

(h) Safety Evaluation Form TVA-10551-DED-2-76 (This form is included in all work packages.)

As a result of these reviews no apparent items of noncompliance or deviations were identified.

6. Units 1, 2 and 3 APRM Gain Adjustment Factors (GAF)

(a) Gain adjustment factors for all six APRM channels were determined for data taken for 46 selected dates between January and June 1977. 276 sets of Surveillance Instruction 4.1.B.2 data were reviewed with the following observations:

(1) In 183 of the 276 sets of data the GAF was greater than 1.0.

(2) In 78 of the 276 sets of data the GAF was greater than 1.02.

(3) In 14 of the 276 sets of data the GAF was greater than 1.05.

(b) Section 7.5.7.4 and Figures 7.5.15 and 7.5.16 of the FSAR addresses the capability of the APRM channels to track core power. Figure 7.5.15 shows that the six APRM channels will track true core power with ± 2% starting at 100% power and 100% flow to below the 65 flow point. Figure 7.5.16 shows that the six APRM channels will track true core power within ± 2% with control rod motion from the most restrictive case and full withdrawal of a control rod from limiting conditions at rated power. Section 7.5.74 further states that normal control rod manipulation results in good agreement (less than 5% deviation on the worst APRM channel through a wide range of power levels and that the adequacy of the flow reference and APRM scram setpoint is demonstrated to be adequate in preventing fuel damage as a result of abnormal operational transients by analyses in Section 14 of the FSAR.

(c) Following the discussion of SI-4.1.B.2 data results the licensee stated that the GAFs calculated from these data was not representative of GAFs maintained after power had been escalated to the operating level and steady state conditions achieved. He considers the GAFs obtained from the core performance computer printouts to be more representative of actual GAFs at steady state conditions. He also stated that he believed that they could justify an average GAF of as much as 1.05, but were attempting to maintain an average maximum GAF of 1.02 to 1.03. This position does not appear to be consistent with the statements made in the FSAR, particularly since the FSAR instrument variations from true power as shown, are with flow and rod position transients. The inspector will review his findings with NRC management and in addition analyze more operating APRM data prior to determing if the GAFs the licensee in maintaining are satisfactory with present APRM setpoint. This is identified as Unresolved Item 77-8/I-1.

7. Method of Obtaining Feedwater Flow Signal for OD-3 Computer Program

During a recent inspection by the inspector a review of the feedwater control system drawing and the OD-3 computer program showed that the feedwater flow signal for the OD-3 program Core Thermal Power Measurement, was taken after the square root converter in the feedwater flow circuitry. The circuit is different in this system from other BWR plants in R-II, in that this signal at other plants is taken ahead of the square root converter, and the square root conversion is part of the OD3 computer program, since the process computer can perform the square root conversion with greater accuracy than the feedwater flow instrument. During discussions with General Electric personnel it was learned that the signal should have been taken ahead of the square root converter in order to achieve the best accuracy in the core thermal power measurement. The licensee will review his current method to determine if it is the method that was intended to be used, and will report to NRC on his findings, along with justification for continuing using the current method, if used. This is identified as Unresolved Item 77-8/I-2.

8. Tour of Control Room Areas

A tour was made of Control Rooms 1, 2 and 3 and adjacent areas to inspect plant conditions, identify any limiting conditions for operation and discuss plant operations with the shift engineers and plant operators. As a result of this tour no apparent items of noncompliance or deviations were observed.