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Mr. Giambusso	4			SENT AEC PDR X SENT LOCAL PDR X

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DESCRIPTION:
Ltr notarized 2-15-73, requesting that the Tech Specs for Unit 1 be revised, trans the following:

ENCLOSURES:
REPORT: Results of Oconee I Hot Functional Testing Internals Vibration Monitoring Program

(22 cys of encl rec'd)

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ACKNOWLEDGED

PLANT NAMES: Oconee Unit No. 1

FOR ACTION/INFORMATION 2-20-73 AB

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Regulatory File Cy.

DUKE POWER COMPANY

POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

February 15, 1973

RECEIVED
FEB 20 1973
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr. Angelo Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Re: Oconee Nuclear Station, Unit 1
Docket No. 50-269

Dear Mr. Giambusso:

Pursuant to 10CFR50.59, Duke Power Company is hereby requesting that the Oconee Nuclear Station Unit 1 Technical Specifications be revised. Oconee Unit 1 is under operation pursuant to License No. DPR-38.

Attached are 22 copies of a report entitled, "Results of Oconee 1 Hot Functional Testing Internals Vibration Monitoring Program." We believe that this report provides sufficient information to confirm the redesign of the reactor internals as presented in Babcock & Wilcox Topical Report BAW-10051 and shows that the prototype vibration measurement program identified in Babcock & Wilcox Topical Report BAW-10038 has been successfully completed. Therefore, Duke Power Company requests that Technical Specification 3.11, "Maximum Power Restriction," which is attached to License DPR-38 as part of Appendix A, be rescinded.

Sincerely,

A C Thies

A. C. Thies

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1173

A. C. THIES, being duly sworn, states that he is Senior Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Atomic Energy Commission this application for change in Technical Specifications; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

ACThies

A. C. Thies
Senior Vice President

ATTEST:

John C. Goodman, Jr.

John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me on this 15th day of February, 1973.

Edna B. Farmer

Notary Public

My Commission Expires:

October 24, 1977

RESULTS OF OCONEE I HOT FUNCTIONAL TESTING
INTERNALS VIBRATION MONITORING PROGRAM

1.0 INTRODUCTION

BAW-10038, Rev. 1, "Prototype Vibration Measurement Program for Reactor Internals (177-Fuel Assembly Plant)," outlined the program to be implemented at Oconee Nuclear Station during hot functional testing (HFT) of Unit 1. During the period November 1972 to January 1973, this program was successfully carried out.

The purpose of this report is to summarize the results of the vibration monitoring program to confirm the design of the reactor internals as presented in BAW-10051, Rev. 1, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration."

2.0 COLLECTION OF DATA

The response of the reactor internals and incore instrument nozzles was monitored by 40 strain gages and 12 accelerometers. In addition, there were pressure cells attached to the core support shield and thermal shield to determine the static and dynamic pressure variations. Pressure sensing lines were also connected to an incore instrument nozzle and mating guide tube to provide data for calculating the cross-flow velocities in the vessel lower head. Figure 1 shows the instrument locations.

Strain gages were placed on two incore instrument nozzles, on two incore instrument guide tubes (one gussetted and one non-gussetted), two thermal shield lower support bolts, six thermal shield support pads, at two locations on the plenum cylinder between the outlet holes, on one surveillance specimen holder tube, and on one shroud tube. Accelerometers were placed on the thermal shield, inside a guide tube to record the motions of the flow distributor, and on the core screen. External accelerometers were placed on the vessel support skirt, and on a vessel head stud. The accelerometer on the screen was utilized to monitor for loose parts, and those external on the reactor vessel were used for reference. Seven pressure transducers were located around the outside diameter of the thermal shield, and four around the outside diameter of the core support shield. Pressure sensing lines were placed, four each in an incore instrument nozzle and four each in an incore guide tube.

All of the sensor output signals and the conditioning electronics were in good working order when the reactor coolant pumps were initially operated. For the duration of the preoperational testing period, the sensors and the data acquisition system continued to work with very good reliability and accuracy. Some data channels, however, did become inoperable during the test. At the end of HFT, there were 14 channels (mostly strain gages) that were inoperative. The failure of these signal channels did not, however, prevent the obtaining of sufficient data to assess the structural adequacy of the internals.

During HFT a total of 280 hours were accumulated (logged) with four reactor coolant pumps operating (full flow conditions) at conditions of approximately 530°F and 2155 psig. Consequently, the thermal shield which had the lowest measured structural frequency of 12 H_z, was subjected to more than 10^7 cycles.

The reactor internals were also subjected to an additional 280 hours of conditions other than full flow operation. This was sufficient to accumulate an additional 10^7 cycles of vibration on the thermal shield.

Outputs of the sensors recorded on magnetic tape were continuously monitored by an on-line computer.

3.0 RESULTS

Table 1 compares the stresses and deflections measured during the hot functional testing vibration monitoring program with the predicted and allowable values as presented in BAW-10038. As indicated by the large ratio of allowable values to measured values, a substantial acceptance margin exists for all components. It is noted that the predicted responses of the various components were calculated using conservative criteria. Consequently, the generally lower values of measured response are not unexpected.

4.0 POST HOT FUNCTIONAL TESTING INSPECTION

Following the conclusion of HFT and the concurrent vibration monitoring program, the reactor internals were removed from the vessel and the post-test inspection program was implemented. The purpose of this program was to visually inspect all major internals, surfaces and/or parts for any indications of distress, loose parts, cracking, fretting, or distortion as a result of HFT.

The results of this inspection program indicated that the reactor internals sustained no structural damage as a result of HFT. It was determined that no deterioration that might affect the structural integrity of the internals had occurred. Only expected, minor indications of fretting and wear were observed on metal-to-metal contact surfaces; e.g., internals vent valve seats.

It was discovered, however, prior to HFT, that the surveillance specimen holder tubes required more torque than anticipated to rotate them for internals insertion into the reactor vessel. The same situation existed after HFT. This situation was investigated as part of the inspection routine and it was determined that bearing and tube misalignment were the cause of the increased torques. Corrective actions were taken:

(a) The surveillance tubes were straightened to improve alignment. Because of schedule, the identical surveillance tubes for Oconee 2 were installed on Oconee 1. The Oconee 1 tubes will be used on Oconee 2.

(b) The installation procedure was modified to improve the alignment.

This situation did not affect the structural adequacy of the surveillance tubes.

5.0 CONCLUSIONS

Evaluation of the measured vibrational responses and the direct visual examination of the reactor internals has revealed that the structures experienced extremely low stresses during operation and sustained no structural damage. All data indicate that the internals will perform in the intended manner with no adverse deterioration occurring over the extended life of the unit.

Table 1

VIBRATION TEST RESULTS

Component	Measured Values		Predicted Values	Acceptance Criteria	Ratio of Allowable To Measured (Peak-to-Peak÷2)
	RMS	Peak-to-Peak÷2			
Nozzle	<100 psi	<100 psi	3,000 psi	5,800 psi	>58.0
Guide Tube:					
Gussetted	<100 psi	<100 psi	2,050 psi	7,000 psi	>70.0
Non-Gussetted	<100 psi	<100 psi	2,700 psi	8,600 psi	>86.0
Flow Distributor	.002 in.	.003 in.	.011 in.	.025 in.	8.3
Surveillance Specimen Holder Tube	250 psi	300 psi	10,000 psi	13,500 psi	45.0
Upper Thermal Shield Support	200 psi	300 psi	4,150 psi	13,500 psi	45.0
Lower Thermal Shield Bolt	760 psi	3,900 psi	3,500 psi	7,900 psi	2.0

GENERAL ARRANGEMENT OF REACTOR VESSEL AND INTERNALS SHOWING INSTRUMENTATION LOCATIONS

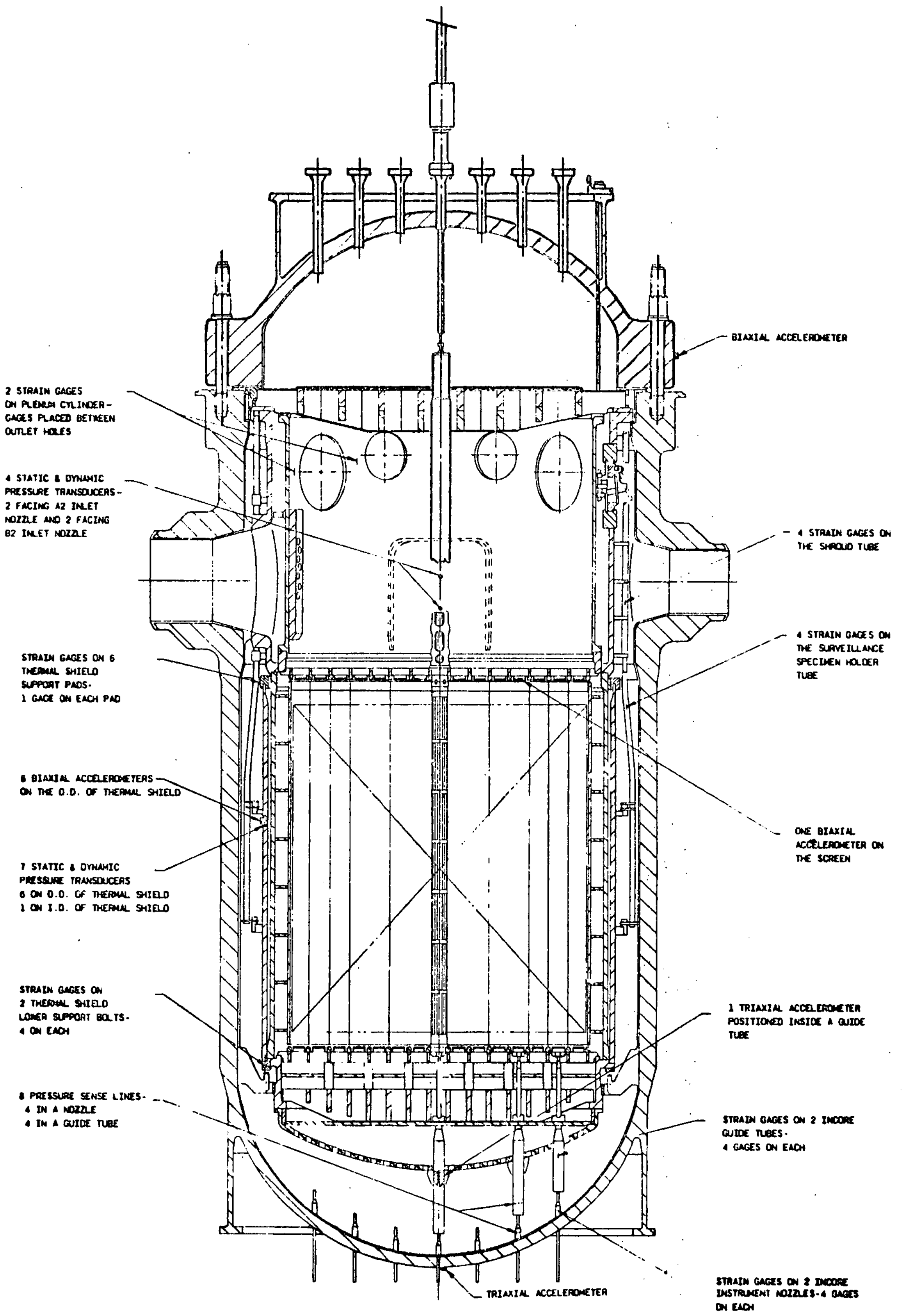

Figure 1