SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT 1

DOCKET NO. 50-269

DECEMBER 19, 1972

APPENDICES

Appendix A - Chronology of Regulatory Review of Oconee Unit 1 Internals Failure and Redesign

1.0 INTRODUCTION

The Duke Power Company (applicant), has requested a license to construct and operate three pressurized water reactors, identified as Units 1, 2, and 3 at its Oconee Nuclear Station in Oconee County, South Carolina.

On June 2, 1969, the applicant filed, as Amendment 7, the Final Safety Analysis Report required by Section 50.34(b) of Chapter 10 of the Code of Federal Regulations as a prerequisite to obtaining an operating license for each unit.

The regulatory staff published its Safety Evaluation Report (original Safety Evaluation Report) for Unit 1 December 29, 1970.

Subsequently a supplemental review of the plant's emergency core cooling systems was performed in accordance with the criteria described in the Interim Policy Statement published in the FEDERAL REGISTER on June 29, 1971 (36 F.R. 12247). Based upon this review the staff issued Supplement 1 to the original Safety Evaluation Report on March 24, 1972.

In March 1972, the Oconee Unit 1 suffered damage to the steam generators and reactor vessel internals requiring significant design modifications. We have reviewed these design modifications and our evaluation is contained in this document. This document is

in the same format as the original Safety Evaluation Report for ease of reference. Our evaluation of the reactor internals prior to modification is contained on pages 20-32 of that report.

Failure of and damage to vessel internals are believed to have been caused by flow induced vibration, and damage to the steam generators was caused by loose parts resulting from the vessel internals failures. The Babcock & Wilcox Company (B&W), the system supplier, has assessed the failures and damage and analyzed the cause through extensive examination, laboratory and full scale tests and system mockups. The results of this assessment and analysis have been reviewed by the regulatory staff. In addition, B&W has redesigned the vessel internals and modified them where required to prevent a recurrence of these failures.

The damage to Oconee Unit 1 has been repaired and the modifications required by the new design have been completed. B&W has provided for an extensive vibration and loose parts monitoring program to be carried out during continuation of hot functional tests in Unit 1 to assure that the system response is well understood and is within design predictions and limits. We have reviewed these programs, have found them to be acceptable, and will follow closely the results of the program throughout the hot functional tests now underway.

5.0 REACTOR COOLANT SYSTEM

5.1 General

At the conclusion of the first phase of the hot functional testing program of Oconee Unit 1 on March 11, 1972, an inspection of the reactor coolant system revealed that several reactor internals components had failed and had caused significant damage to hardware within the reactor vessel and to both steam generators. This incident was formally reported to the AEC on April 4, 1972. Since learning of the incident the regulatory staff has been actively involved in reviewing the matter both through visual inspections and the review of data and of corrective engineering design developed by the applicant in order to assess and evaluate the safety implications. Appendix A provides a chronology of the regulatory review including field trips, meetings with the applicant and submittal of important documents. The following sections of this supplement summarize our safety evaluation.

5.2 REACTOR COOLANT SYSTEM COMPONENTS

Damage to the steam generators in the primary system was caused by the impact of portions of the failed internals components from the reactor vessel. This damage was confined to the upper steam generator plenum region and consisted of deformation of the tube ends, the tube sheet clad and the dome wall clad. Since the steam generators could be repaired by approved repair procedures judged

to be adequate to restore the generators to their original quality no further safety evaluation was warranted other than to verify the acceptability of the restoration by inspection and test which has been done. This safety evaluation deals with the cause and prevention of flow induced vessel internals failure. The failures experienced were:

a. Incore instrument nozzles (21 broken off and remaining 31 damaged).

b. Incore instrument guide tubes (4 broken off, 4 cracked and remaining 44 no apparent damage).

c. Thermal shield (evidence of movement, wear and mating surface damage).

d. Instrumentation guide tubes (2 broken off and remaining 4 no apparent damage).

As amended through Amendment 37 the Duke Power Company application for an operating license for Oconee Unit 1 references four B&W topical reports dealing with the reactor internals, their failure and redesign. They are:

BAW-10037, Revision 2, November 1972, "Reactor Vessel Model Flow Tests"

BAW-10038, Revision 2, November 1972, "Prototype Vibration Measurement Program For Reactor Internals"

BAW-10050, Revision 1, November 1972, "Evaluation of Oconee Reactor Failure"

BAW-10051, Revision 1, November 1972, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration"

Our safety evaluation is based principally upon these reports; visits to the reactor site and vendor's facilities; and meetings held with the applicant and B&W.

In the process of reviewing the failure, we have evaluated the above reports. We concur with the conclusions set forth in BAW-10050 Revision 1 that the recommended design modifications on the internals have been based upon a conservative application of the response and failure data from Oconee Unit 1. We concur with the conclusions set forth in BAW-10037 Revision 2 that the reactor vessel scale model flow test approach used will verify the core flow distribution, However, due to a lack of valid flow forcing functions, B&W has not yet demonstrated a dynamic analysis to predict the structural behavior of reactor internals when subjected to transient loadings. The redesigned internals are accepted for Oconee Unit pending satisfactory completion of the new hot functional preoperational tests. The results obtained from the preoperational tests will be evaluated prior to permitting significant power operation to confirm this acceptability.

The lack of valid vibration predictions precludes our acceptance at this time of BAW-10038 and BAW-10051 with respect to the designation of Oconee 1 as a prototype for follow-on plants.

In BAW-10050 Revision 1, B&W describes its investigation on the cause of the preoperational test failure. On the basis of the metallographic examination of the failure surfaces B&W concluded that fatigue due to flow induced vibratory motion was the major failure mode. Component redesign was based upon (a) separation of structural frequencies further from vortex shedding frequencies, and (b) reduction of the stresses to a level further below the material endurance limit. We concur with B&W that such design modifications will improve the structural integrity of the reactor internals.

In BAW-10037, Revision 2, B&W describes the reactor vessel flow testing conducted on a one-sixth scale model to investigate flow distribution, pressure loss and the pattern of flow mixing from the various inlets. The flow characteristics inside the core and vent valve testing were emphasized. Both the original and the modified designs were tested. The results of the tests showed that the modified design provides more uniform flow distribution with acceptable pressure loss. The flow rate was slightly higher at certain portions of the core and required further minor modifications in design. We concur with B&W on the approach used to verify the core flow distribution.

In BAW-10051, Revision 1, B&W describes the attempts made to justify the reactor internals design modifications by computing

responses of modified components to flow induced vibration. However, the actual flow forcing functions may not be verified until completion of the new preoperational vibration test program for Oconee Unit 1. Until the required verification is available, we cannot concur with the applicant's conclusion on this matter. The applicant has stated that further steps will be taken, including component testing of instrument guide tubes and incore nozzle assembles, to provide a better understanding of the vibration behavior. In addition a more definitive understanding of the thermal shield vibration response characteristics will be sought through further evaluation of the Oconee Unit 1 response and failure data.

In BAW-10038, Revision 2, B&W describes its prototype preoperational vibration testing program for reactor internals. The applicant cannot provide valid vibration predictions as required by Safety Guide 20, "Vibration Measurements on Reactor Internals" for prototype qualification because of the lack of conclusive dynamic analysis. Therefore, we cannot complete our prototype qualification evaluation of Oconee Unit 1 at this time.

5.3 Conclusion

Based on the above evaluation and the information presented in our original Safety Evaluation Report we conclude that there is reasonable

assurance that the redesign of the Oconee Unit 1 reactor internals is acceptable pending confirmation by the vibration testing to be conducted during the preoperational tests. Operations of Unit 1 will be restricted to no greater than 5.0% full rated power until the results of preoperational testing have been evaluated by the regulatory staff.

APPENDIX A

CHRONOLOGY OF REGULATORY REVIEW OF OCONEE UNIT 1 INTERNALS FAILURE AND REDESIGN

1.	March 29-30, 1972	Site visit to Oconee to view Unit 1 internals and steam generator damage.
2.	April 6, 1972	Meeting at Barberton with B&W and Duke to discuss steam generator repairs and vibration testing of internals for Oconee Unit 1.
3.	May 24, 1972	Meeting at Bethesda with B&W and Duke to discuss repair of steam generators and vessel internals for Oconee Unit 1.
4.	August 7, 1972	Meeting at Bethesda with B&W and Duke to discuss vibration monitoring of vessel internals.
5.	September 15, 1972	Application Amendment No. 35 provided B&W topical reports, BAW-10037, BAW-10038, BAW-10050, BAW-10051.
6.	October 25, 1972	Meeting at Bethesda with B&W and Duke to discuss reactor vessel internals redesign and vibration monitoring.