MAY 04 1984

MEMORANDUM FOR: E. Adensam, Chief
Licensing Branch, #4, DL

FROM: C. Berlinger, Chief
Core Performance Branch, DSI

SUBJECT. REQUEST FOR INFORMATION FOR VOGTLE UNITS 1 AND 2

Plant Name:	Vogtle Units 1 and 2
Docket Numbers:	50-424/425
Licensing Stage:	Operating License
Responsible Branch:	Licensing Branch #4
Project Manager:	M. Miller (X-24259)
DSI Review Branch:	Core Performance Branch
Requested Completion Date:	April 25, 1984
Review Status:	Incomplete - Additional Information Required

The Core Performance Branch has prepared the attached questions on Section 4.4, "Thermal-Hydraulic Design," of the Vogtle FSAR. The Physics questions were sent previously. The Fuels questions are being prepared with the aid of a contractor and should be done by June.

Original signed by: L. E. Phillips

for Carl H. Berlinger, Chief
Core Performance Branch, DSI

Enclosure:
As stated

cc: R. Mattson
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Comments and Questions

VOGTLE UNITS 1 AND 2

492.1 Georgia Power Company (GPC) has provided information on the loose parts monitoring system (LPMS) for Vogtle Units 1 and 2 which is called a Metal Impact Monitoring System (MIMS). However, the responses are not complete. Also, Section 4.4.6.4 of the FSAR states that conformance with Regulatory Guide 1.133 is discussed in Section 1.9 of the FSAR. Section 1.9 of the FSAR states that Westinghouse (W), with GPC concurrence, has taken a position which takes exception to any need for regulatory guidance relative to a LPMS. Also, GPC has taken exception to some items in Regulatory Guide 1.133. These items relate to seismic qualification, redundancy, separation, and in-containment calibration. Also, Item C.5.b, Section D and some of the technical requirements are not agreed to. However, the licensee has evaluated the requirements against an early draft version of Regulatory Guide 1.133 for which some of the requirements have been modified in the final version, Revision 1, May 1981. The licensee has not provided justification for these exceptions other than arguments with the Regulatory Guide 1.133 criteria. Since these criteria have been used for licensing for several years and since the cited version of Regulatory Guide 1.133 was issued with due consideration for industry comments, the justification provided is unacceptable. We will require the licensee to provide a LPMS consistent with the provisions of Regulatory Guide 1.133 as has been provided for other licensed reactors and to commit to provide, prior to power operation, a final design report which contains the following:

1. An evaluation of the LPMS for conformance to Regulatory Guide 1.133.

2. A description of the system hardware, operation and implementation of the loose parts detection program, including plans for start-up testing, acquisition of baseline data and alarm settings.

3. A description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

4. A description of the operator training program.

A sample table of contents of the LPMS description is provided in Enclosure 1.

492.2 Standard Format and Content of Safety Analysis Reports, Regulatory Guide 1.70, states that in Chapter 4 of the SAR

> "...the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operation modes..."

Are the analyses presented in Section 4.4 representative of the initial core only or have future cycles been analyzed? Provide a discussion of how power distributions for future cycles are considered in the FSAR analyses. Is there any assurance that the Vogtle Units can operate at the licensed power level without excessive DNB trips throughout future cycles? Will revisions to the design methodology be required in order to maintain sufficient thermal margin?

492.3 The SRP for 4.4, Thermal and Hydraulic Design, in Section II.9 states that information should be provided in response to NUREG-0737, Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling." Therefore the staff will require the applicant to provide the documentation itemized in Item II.F.2 of NUREG-0737.

492.4 The effects of fuel rod bowing must be included in the thermal-hydraulic design. The predicted extent of rod bow (gap closure) versus exposure and the effect of rod bowing on DNBR must be addressed. Provide the maximum projected assembly burnup and the gap closure for the rod bow penalty. Also, provide a table of rod bow penalty vs burnup (MWD/MTU).

492.5 Operating experience on two pressurized water reactors (not of the Westinghouse design) indicate that significant reduction in core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for Vogtle we will require provisions to assure that the flow rates are not lower than the minimum design flow allowed. Therefore, provide a description of the flow measurements capability for Vogtle as well as a description of the procedures to measure flow and the actions to be taken in the event of an indication of lower than design flow.

492.6 Please state your intent regarding the use of the Westinghouse optimized fuel assembly in your plant. If the use of this design is being considered, provide a discussion of the status and schedule for any revised submittals.

492.7 Please state your intent regarding the use of the Westinghouse "Improved Thermal Design Procedure" described in WCAP-8567, dated July, 1975. If you intend to use these methods, responses to the following questions will be required:

(a) Provide a block diagram depicting sensor, process equipment, computer, and readout devises for each parameter channel used in the uncertainty analysis. Within each element of the block diagram, identify the accuracy, drift, range, span, operating limits and setpoints. Identify the overall accuracy of each channel transmitter to final output and specify the minimum acceptable accuracy for use with the new procedure. Also identify the overall accuracy of the output value and maximum accuracy requirements for each input channel of this final output device.

(b) Discuss the method(s) for incorporating environmental effects (e.g., noise, EMI) on instrument channels into the uncertainty analysis.

(c) Provide data to verify that the plant instruments will perform with a high degree of confidence, within their design accuracies. This information may be obtained from operating history of identical instruments installed in other plants. This request pertains to the instruments affecting the uncertainties in the design procedures (as identified in question 1 above), the overtemperature ΔT trip, the high flow trip, the low pressure trip and the pump voltage trip.

(d) Provide the ranges of applicability of sensitivity factors.

(e) Demonstrate that the linearity assumption of equation 3-8 in WCAP-8567 is valid when the WRB-1 correlation is used.

492.8 The following items relate to the Technical Specifications which should include:

1. A declaration that prohibits N-1 loop operation unless it is adequately justified in plant-specific analysis.

2. Appropriate surveillance to ensure acceptable flow rates and to recognize crud buildup.

3. A discussion in the basis of the Technical Specification of any generic or plant-specific margins that have been used to offset the reduction in departure to nucleate boiling ratio (DNBR) due to rod bowing.

492.9 In Section 15.1.5 of the Vogtle FSAR the Steam Line Break (SLB) accident two cases are presented: Case 1, which is at an initial no-load condition with offsite power available; and Case 2, which corresponds to Case 1 with additional loss of offsite power at the time the SI signal is generated. From the figures presented the minimum RCS pressure appears to be approximately 500 psia for Case 1 (Figure 15.1.5-2) and approximately 950 psia for Case 2 (Figure 15.1.5-5). The following information is requested:

1. What DNBR correlation is used in the analysis for SLB?

2. Provide information on the applicable range of the parameters for the DNB correlation and compare with the range experienced (especially pressure) in the SLB accident.

3. Provide information on the DNBR margin available over the design limit for the SLB.

492.10 Section 4.4.2 of the Vogtle FSAR refers to model tests for obtaining core pressure drop using correlations from one-seventh scale model hydraulic test data of the San Onofre and Connecticut-Yankee reactor models. Provide information on the similarity of these model reactors to the design of the Vogtle reactor and how any design differences were addressed in utilizing the results of the tests.

ATTACHMENT 1

SAMPLE TABLE OF CONTENTS
LOOSE PART DETECTION PROGRAM DESCRIPTION

I. System Description

   A. Scale piping diagram showing LPM sensor locations
   B. Sensor specifications (type, manufacturer, sensitivity, temperature rating, etc.)
   C. Sensor mounting details (drawing and procedure)
   D. Preamplifier or line driver (type, manufacturer, location and specifications)
   E. Functional description of LPMS

      1. Theory of operation, detection logic, alarm display
      2. Data recorder specifications (No. of channels, length of recording, frequency range, and conditions under which recording is initiated)

II. Operational Procedures

   A. System Calibration Procedures and Results

      1. Initial and subsequent calibrations
      2. Functional check, as defined in Regulatory Guide 1.133
      3. Channel check, as defined in Regulatory Guide 1.133

   B. Plant Operator Instructions for Use of LPMS

      1. Procedures for routine operation
      2. Procedures to be used following indication of a loose part

         a. Method to confirm existence of loose part
         b. Method to diagnose a loose part (size and location)

III. Evaluation for Conformance to Regulatory Guide 1.133 and Justification for any Deviations

May 30, 1984

M. S. Dunenfeld
Core Performance Branch
U.S. Nuclear Regulatory Commission

Attached are the questions (Milestone 1 of Task 8 of FIN No. B2544) concerning the FSAR for Vogtle-1 and -2 (Docket Nos. 50-424 and 50-425, respectively).

W. J. Bailey and C. E. Beyer
Pacific Northwest Laboratory

cc: R. Lobel

Vogtle

490.0 Reactor Fuels Section, Core Performance Branch

490.1 The reference in Section 4.2.1.3 for fuel rod models does not appear to be correct. Please confirm that it should be Reference 6 rather than Reference 5.

490.2 Several fuel performance models, i.e., for rod bowing, fuel and cladding materials properties, creep collapse, the PAD code, and rod internal pressure, referenced in Section 4.2 of the Vogtle FSAR are not approved models. Have analyses using the approved versions (if different from the unapproved versions) of these models been performed for the Vogtle plant? If the results have changed using the approved models, what are the new results?

490.3 Methods and criteria for evaluating fuel cladding stress, strain and fatigue have been presented; however, the specific results of an analysis for the Vogtle plant have not been presented. Have such analyses been performed for Vogtle? If so, what are the results of these analyses?

490.4 The analysis of combined seismic and LOCA loads has referenced an unapproved model that has subsequently been approved for analysis. Are the analysis methods for the unapproved version of this model applicable to the methods defined in the approved version? If not, has this analysis been performed using the approved methods and, if so, what are the results?

490.5 The use of the CVCS letdown monitor for detecting fuel rod failures has been explained in the Vogtle FSAR. Is there a definite commitment and plan for the active use of this system to monitor fuel failures, as per SRP Section 4.2?

490.6 Does the analysis of the fuel handling accident (Section 15.7.4 of the FSAR) take into account that the peak pellet burnup of approximately 50,000 MWd/tonne of uranium shown on p. 4.2-2 exceeds the value (i.e., approximately 45,000 MWd/t) stated in Footnote a, 3 of Table 15.7.4-2 (Sheet 13 of 13)?

490.7 Tables 4.1-1 and 4.4-1 show the nominal coolant pressure as 2250 psia. Does the use of 2280 psia for reactor coolant pressure in the ECCS analysis (see Table 15.6.5-1) provide more conservative results?

490.8 On p. 4.1-2 of the FSAR, it says hafnium or silver-indium-cadmium absorber rods are to be used. The NRC staff (Ref. 1) believes that a minimal surveillance program consisting of a visual inspection of representative rods should be carried out at the first two plants (expected to be Callaway-1 and Comanche Peak-1) to have the new hafnium control rods. If for any reason the startup of one of those plants should be delayed beyond the startup of Vogtle and if Vogtle is to use hafnium absorber rods, then the applicant should at that time provide an acceptable surveillance program that would be implemented at Vogtle.

Reference

1. L. S. Rubenstein (NRC), Memorandum for R. L. Tedesco (NRC), "SSER Input for Callaway Concerning Hafnium Rod Surveillance", June 30, 1982.