UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 21 1981

Docket Nos.: STN 50-482
and STN 50-483

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Dear Gentlemen:

Subject: SNUPPS FSAR - Request for Additional Information

As a result of our review of your application for operating licenses we find that we need additional information regarding the SNUPPS FSAR. The specific information required is as a result of the Thermal-Hydraulic Section of the Core Performance Branch's review and is listed in the Enclosure.

To maintain our licensing review schedule for the SNUPPS FSAR, we will need responses to the enclosed request by May 27, 1981. If you cannot meet this date, please inform us within seven days after receipt of this letter of the date you plan to submit your responses so that we may review our schedule for any necessary changes.

Please contact Mr. Dromerick, SNUPPS Licensing Project Manager, if you desire any discussion or clarification of the enclosed report.

Sincerely,

R Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

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Request for Additional Information

SNUPPS - FSAR

492.0 THERMAL-HYDRAULICS SECTION, CORE PERFORMANCE BRANCH

492.2 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. Use of the staff report "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," February 16, 1977, represents an acceptably conservative treatment of rod bowing.

492.3 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 Callaway and Wolf Creek we will require provisions to assure that the minimum design flow rates are not exceeded. Therefore, provide a description of the flow measurements capability for Callaway and Wolf Creek 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.

492.4 The NRC approval of the THINC-IV code, for use in the thermal-hydraulic design, indicates that the pressure gradient at the core exit must be modeled. Provide a revised THINC-IV calculation at the steady state reactor design conditions including the modeling of the core exit radial pressure gradient. Provide the following specific information from that calculation:

1. minimum DNB ratio (value and location)
2. hot channel flow vs. axial position
3. hot channel enthalpy vs. axial position
4. hot channel quality vs. axial position
5. hot channel void fraction vs. axial position
6. the assumed core exit pressure gradient.

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492.5 Insufficient information has been provided to justify the design power level of 2389 Mwt (70% of full power) during three-loop operation. Temperature differences in the active cold legs of a few degrees could exist during three-loop operation. Therefore a radial power tilt and an increase in enthalpy rise factor could result. As a result, we request that a complete detailed description of the following items be provided:

1. The method of determining the temperature distribution among the cold legs and the associated radial power tilt;

2. The method of accounting for differences (if any) in the three-loop thermal-hydraulic design;

3. The instrumentation available and monitoring procedures during three-loop operation;

4. The DNBR Technical Specification and how it will be implemented for three-loop operation;

5. The reactor protective system setpoints related to DNBR protection and how they are generated;

6. The effects of anticipated operational occurrences on the cold leg temperature distributions and how this effect is included in the design.

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 devices 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 procedure (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.