VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

February 13, 1995

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No.	94-746
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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
CORE UPRATE - MECHANICAL ENGINEERING BRANCH
REQUEST FOR ADDITIONAL INFORMATION

The Surry Core Uprate Technical Specification change request was submitted for NRC review by an August 30, 1994 letter (Serial No. 94-509). By a December 16, 1994 letter, the NRC (NRR Mechanical Engineering Branch (EMEB)) requested additional information to support their review of our core uprate submittal. The questions and our responses are provided in Attachment 1.

If you have further questions or require additional information, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachment 1- Responses to Request for Additional Information - Proposed Amendment for Power Uprate - Surry Power Station Units 1 and 2 - Mechanical Engineering Branch

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Mr. M. W. Branch
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ATTACHMENT 1

RESPONSES TO
REQUEST FOR ADDITIONAL INFORMATION -
PROPOSED AMENDMENT FOR POWER UPRATE -
SURRY POWER STATION UNITS 1 AND 2

Mechanical Engineering Branch

1. **Section 4.1.1 states that the reactor vessel internals were reviewed and found to be acceptable for the uprated core power operating conditions as part of the previous evaluation performed to assess the Surry Improved Fuel design. Provide a discussion of the method used for the review that was conducted on the reactor vessel internals for the power uprate condition including effects of the flow induced vibration. Identify the reactor internal components that were reviewed and the Code used for acceptability of the reactor vessel internals. Also, provide a summary of maximum critical component stresses and calculated fatigue usage factors.**

RESPONSE:

The reactor vessel internals for the Surry units are of the same basic design as those used in numerous other three loop reactors which were supplied by Westinghouse or Westinghouse licensees in the United States, Japan, Belgium, France and Sweden. Some of these plants were initially placed into service at power levels which exceed the desired uprated power level for the Surry units, and these plants have operated successfully for a number of years at these higher power levels. For example, a domestic reactor was initially designed, licensed, and operated at 2660 MWt, whereas the proposed uprated power for Surry is only 2558 MWt (2546 MWt core power). As another example, a foreign reactor has been uprated to a power level of 2660 MWt. Similarly, the North Anna units have been uprated to a power level of 2905 MWt.

The Surry reactor internals were designed and fabricated before the existence of Subsection NG (Core Structures) of the ASME Code, therefore no ASME code design or stress report exists for those reactor internals. More importantly, because of the successful operating experience of the other three loop reactors, no detailed flow induced vibration and structural analyses were necessary as part of the uprating program. However, a Surry uprating feasibility study was performed and that study included a systematic investigation of the potential impact of operating at the uprated conditions on the reactor internals components. Some parts of the investigation were performed by making plant specific scoping calculations. Other parts of the investigation were based on the use and extension of existing analyses that had been performed for similar plants which were designed and built strictly in accordance with ASME Subsection NG requirements.

The only parameters which are affected by the uprating and potentially impact the reactor internals are:

- a less than 3°F reduction in core inlet temperature and
- a less than 5% increase in total heat generated in the core barrel and baffle region, thermal shield, and upper and lower core plates.

There was no projected change in thermal transients and LOCA blowdown forces due to the uprating, so there was no need to review these two issues.

Evaluations were performed which demonstrated that the effect of these changes on the thermal-hydraulic and structural performance of the reactor internals was negligible. The small reduction (3°F) in inlet coolant temperature caused an insignificant increase in coolant density, resulting in an also insignificant effect on the flow induced vibration of the reactor internals. The average temperatures in the components surrounding the reactor core changed by less than 2°F and temperature differences between the components were also negligibly affected. Thus, it was concluded that thermal stresses and fatigue usage factors would not change significantly as a result of the uprating. In conclusion, operating at the uprated conditions is not expected to have any significant effect on the reactor internals, as evidenced by the safe operation of a number of similar reactors at higher power levels than are planned for the Surry units.

2. In Section 4.1.5, it is stated that review of the previous analysis for the uprated core power with 7% tube plugging verified that the components continue to remain in compliance with the ASME Code, Section III requirements. Provide a discussion addressing the methodology, assumptions and the Code edition used in the evaluation of critical steam generator components. Specify the limiting components evaluated for the uprated power conditions and list the maximum stresses, fatigue usage factors and location of highest stressed areas for both the current design and the uprated power conditions.

RESPONSE:

A comparison of parameters applicable for the uprating case was made with the design parameters of the replacement steam generator model 51F. This comparison, shown on page 21 of the Surry Core Uprate Licensing Report, indicates that most of the key parameters are identical. A small change in some of the parameters is not considered to have any significant effect on the stress levels of the steam generator components. There are no changes to the design transients. Critical parameters, such as the primary and secondary side pressures, as well as the vessel outlet and secondary side temperatures, remain essentially unchanged. Based on the stress evaluation of the replacement steam generator, it was concluded that the critical steam generator components would meet the required stress and fatigue limits of the ASME Code (1974 Edition through Winter 1976 Addendum).

3. In Section 4.1.6, provide an evaluation of the increased temperature difference between the hot leg and the cold leg on the pressurizer spray piping and nozzles at the uprated power conditions in comparison with the current design basis analysis. Discuss in detail the evaluation of pressurizer surge line, the pressurizer safety valve and discharge piping, and the pressurizer relief tank for compliance with the design criteria. Specify the Code and edition used for acceptability of pressurizer components.

RESPONSE:

For the pressurizer spray nozzle, the uprating was determined to have no impact. The original analysis remains valid. The stress calculation for the pressurizer spray piping was reviewed, and it has been determined that the small increase in temperature difference between the hot leg and the cold leg will have an insignificant effect on stress. The applicable design code for the pressurizer spray piping is ASA B31.1, Standard Code for Power Piping, 1955 Edition including Code Case N7, September 1962.

The design adequacy of the pressurizer surge line is determined by the ability of the surge line to pass the flow required to prevent reactor coolant system overpressurization during a design basis safety valve discharge event. For the safety valve discharge piping, the design adequacy is determined by the ability of the discharge piping to maintain the back pressure on the safety valves below the design value of 500 psia. In both cases, an evaluation of the hydraulic capability of the lines in question was performed, and the existing piping was found to be acceptable for the uprated conditions. The ASME code edition that was used to determine acceptability of the pressurizer components was the 1965 Edition through the Summer 1966 Addendum.

As required by NRC Bulletin 88-11, entitled "Pressurizer Surge Line Thermal Stratification," stress and fatigue analysis of the pressurizer surge line was performed for normal thermal, seismic (OBE and DBE), normal/upset thermal transients including effects of thermal stratification and striping. The analysis was performed conforming to the requirements of the 1986 edition of the ASME Section III, Subsection NB with Addenda through 1987. The staff review and approval of the results were obtained by an NRC letter dated March 13, 1992. The temperature of the reactor coolant loop hot leg is not altered by the core uprate, and the changes in loadings due to core uprate remain essentially unchanged. Therefore, the currently approved results of the surge line analysis remain valid.

The pressurizer safety valve discharge piping system, including the pressurizer relief tank (PRT), was recently analyzed in order to meet the requirements of NUREG-0737. The analysis was performed for normal operating loads, seismic (OBE and DBE), and pressure transient loads resulting from the actuation of pressurizer safety and relief valves. The loading conditions used in the analysis essentially envelope the anticipated loadings during the core uprate condition. The PRT operates with two rupture discs set to 100 psig. The sizing basis for the original design of the tank is based on the PRT being able to cool 110% of the pressurizer full power steam volume. A review of the proof-of-design calculation determined that the current setpoints and sizing of the PRT remain applicable at the current and uprated conditions.

The pressurizer safety valves were recently replaced with valves of equal capacity. Evaluation of the replacement valves concluded that they have sufficient capacity for uprated conditions. The replacement valves are designed to the requirements of the 1977 Edition of the ASME Section III code through the Summer 1977 Addendum.

4. In Section 4.1.7, provide a detailed discussion regarding the evaluation of the NSSS piping and pipe supports, equipment nozzles, and in-line components for the power uprate condition, and include a summary of the analysis results in comparison to the existing design values. The discussion should include the analysis methods and assumptions, and demonstration of compliance with requirements in the Code edition used for the evaluation with regard to stress levels and fatigue considerations.

RESPONSE:

The reactor coolant system piping was evaluated in 1985 to the USAS B31.1 Power Piping Code, 1967 Edition. The results of that analysis were provided to the NRC by a December 3, 1985 letter (Serial No. 85-136A). No formal computer analysis was performed in the evaluation of the Surry reactor coolant loop piping and supports and primary equipment nozzles for the uprating project. However, since the temperature difference between the existing full power conditions and the uprated full power conditions is small, the system displacements, loads, and stresses were judged to be unaffected by uprating. This judgment was based on engineering tolerances associated with analyses of this type, and the conclusion was that the existing design basis results for the loop piping and supports continue to apply without modification for the uprated condition.

5. **In Section 4.2, the evaluation of balance of plant (BOP) systems did not address effects of the power uprate on the design basis analyses of the postulated pipe rupture locations, pipe whip and jet impingement loads that may affect the adequacy of the safety-related systems, equipment and components. Please provide relevant information concerning these evaluations.**

RESPONSE:

Core uprate has the potential to affect the postulated high energy line break (HELB) analysis in the balance of plant (BOP) systems. A review of the affected high energy systems shows that, in the core uprated condition, the increase in thermal stress in the systems is less than 3%. The increase in stresses in a system is essentially uniform and is not expected to alter the postulated break locations. Since the systems do not experience pressure increases as a result of core uprate, the pipe whip and jet impingement energy is bounded by the existing analysis. Therefore, the postulated HELB results described in the Surry UFSAR remain valid.

6. In Sections 4.2.7 and 4.2.8, provide an evaluation of the effect of increased pressure at discharge nozzles shown in Tables 4.2.7-1 and 4.2.8-1, on the adequacy of the high pressure and low pressure heater drain discharge piping at the uprated conditions.

RESPONSE:

The Surry core uprated temperatures and pressures at the suction and discharge nozzles of the high pressure heater drain pumps (1-SD-P-1A, 1-SD-P-1B, 2-SD-P-1A, 2-SD-P-1B) are enveloped by the values used in the existing pipe stress evaluations. Therefore, the core uprate operating parameters will have no effect on the pipe stress evaluations.

The Surry core uprate temperatures at the suction and discharge nozzles, as well as the pressures at the suction nozzle of the low pressure heater drain pumps (1-SD-P-2A, 1-SD-P-2B, 2-SD-P-2A, 2-SD-P-2B), are enveloped by the original/existing values used in the pipe stress evaluations of piping attached to the pumps. Although the pressure at the discharge nozzle increased by 57 psi (from 515 to 572 psig), the core uprate pipe stress and pipe support review provides an evaluation of the piping attached to the discharge side of the pumps for a pressure of 586 psig.

7. **In Section 4.5, it is stated that the 5th point extraction steam heater drain piping nozzle loads exceed the existing nozzle load allowables. The nozzle will require modification for operation at the uprated temperature. Provide a discussion on the planned modification, schedule, and operational experience at similar operating conditions to ensure that these nozzles will be operating within the design allowables. Table 4.5-1 shows that the operating temperature at the uprated power level is more severe for the 3rd point than for the 5th point extraction steam heater drain piping. Provide a summary of the calculated 3rd point heater drain piping nozzle loads and maximum piping and pipe support stresses in comparison to the appropriate code allowables.**

RESPONSE:

The overstress in the fifth point heater drain nozzles is a result of the location of the heater and the configuration of the drain piping. The fifth point heaters are located inside the condensers and protrude through the condenser walls. The drain piping from the fifth point heater drain nozzle has only a small offset before it penetrates the condenser wall. This offset leaves little room for thermal expansion.

The fifth point feedwater heater drain nozzles will be reinforced by welding reinforcing pads 3/8" thick by 4" wide around the nozzles (each unit has two fifth point feedwater heaters, each feedwater heater has one nozzle). This reinforcing of the nozzles is scheduled for the currently ongoing Surry Unit 2 refueling outage and the Surry Unit 1 refueling outage, currently scheduled to begin in September 1995. The work is being performed by a design change package (DCP). The reinforcement will not adversely affect current plant operation, and the DCP has been approved by the Station Nuclear Safety and Operating Committee.

The pipe stress evaluation of the piping of the third and fourth point heater drains to the condenser nozzles was originally evaluated with a minimum offset (nomograph) method using a temperature equal to the core uprate value. The existing evaluation shows a maximum calculated stress less than the Code allowable stress. For the most part, the pipe support system uses rod hangers and spring hangers and would not be significantly affected by the expected changes in temperature of the piping.