50-271

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 9, 1996

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Brattleboro, VT 05301

SUBJECT: VERMONT YANKEE CORE SHROUD MODIFICATION (TAC NO. M95207)

Dear Mr. Reid:

In April 1995, the NRC staff reviewed the results of core shroud inspections at Vermont Yankee submitted by Vermont Yankee Nuclear Power Corporation (licensee) and concluded that there was adequate structural margin to allow operation for one additional fuel cycle. The staff required that the licensee reinspect and/or repair the core shroud prior to the startup from the 1996 refueling outage. The licensee determined to proceed with a modification of the core shroud. By letter dated April 15, 1996, the licensee submitted its proposed design modification for the core shroud at Vermont Yankee.

The licensee stated that the core shroud modification is a full structural repair in that it is capable of fulfilling all safety and operational design bases of the core shroud as defined in the Vermont Yankee Final Safety Analysis Report, assuming the most adverse combination of any or all core shroud circumferential welds not maintaining their structural integrity.

The modification is an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, and is being submitted for NRC approval under 10 CFR 50.55a(a)(3). An alternative repair concept became necessary since the typical method of repair under Section XI (removal of the flaw indication and rewelding) is not considered feasible for the existing core shroud indications.

The licensee's submittal consists of two parts. The first part is a non-proprietary summary of the modification, prepared in accordance with the format specified in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidance document, BWRVIP-04, dated October 1995. The second part is a proprietary design report containing engineering drawings and detailed engineering calculations.

The modification is designed by MPR and relies on the use of four tie rod assemblies with relatively less preload than the ten tie rod assemblies employed in the previous MPR design at FitzPatrick and Oyster Creek.

Based on its preliminary review of the design report and calculation results, the NRC staff has prepared a list of questions in the areas of structural design, materials and fabrication. The licensee's response to this request for additional information is needed before the staff can complete its evaluation.

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Therefore, we request that you respond within 30 days of the date of this letter.

If you have any questions, please contact me at (301) 415-1429.

Sincerely,

Daniel H. Dorman

Daniel H. Dorman, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
VERMONT YANKEE NUCLEAR POWER STATION
CORE SHROUD REPAIR
DOCKET NO. 50-271

1. The two documents (stated below) relating to the design specification for Vermont Yankee Nuclear Power Station (VYNPS) core shroud repair have been referenced in the submittal. Indicate how the two documents differ, and identify any areas of potential differing or overlapping requirements. The two referenced documents are:

   (i) VYS-046, "Specification for Design, Fabrication, and Installation Services for Reactor Pressure Vessel Core Shroud Repair at Vermont Yankee Nuclear Power Station," Yankee Atomic, Revision 1.

   (ii) "Design Specification for Vermont Yankee Nuclear Power Station (VYNPS) Core Shroud Repair, MPR Specification 249001-001, Revision 1."

2. Potential degradation of the tie rod assembly could cause loss of preload due to fatigue, corrosion, wear or other aging effects in combination with flow induced vibration and may render it incapable of meeting its design function. How can it be determined that the degradation mechanisms are identified on the inside and outside of the tie rod assemblies? Also provide plans to ensure that the structural integrity of the tie rod assembly will be maintained during the licensed life of the plant.

3. In the analysis of the recirculation line break (RLB) loads on the core shroud, it appears that the loads have been statically applied to the shroud structure. However, the RLB loading greatly fluctuates with respect to time, which could result in significant dynamic amplification of the loads onto the shroud. Provide an evaluation of the repaired core shroud with dynamically applied RLB loads or demonstrate analytically that the statically applied loads are more conservative from a design standpoint.

4. For a main steam line break (MSLB) event with a failed horizontal core shroud weld below the core support plate, it appears that the maximum core shroud upward motion is limited by the control rod guide tubes in addition to the core spray lines. As such, the repair tie rod assemblies will not carry all of the uplift loading from the resulting blowdown. What is the resulting loading on the control rod guide tubes from an MSLB event, and the effect of this loading on their structural integrity and control rod insertion capability?

Enclosure

5. Following the projected separation of the shroud at a horizontal weld location as a result of a postulated upset, emergency, or faulted event, the upper portion above the failed weld will fall back onto the lower portion. Has the kinetic energy of the falling mass been evaluated concerning its effect on the structural integrity of the shroud, fuel, control rods, reactor vessel and any other safety component? Also provide the results of this evaluation or basis for concluding the evaluation is unnecessary.

6. Once the tie rods have been installed and pretensioned, the top nut is tightened and the thin section on the end is crimped to the rod to prevent nut rotation and loss of preload. This crimped area of the nut could result in susceptibility to IGSCC. If through-wall cracking of the nut occurs, what is the consequence of the loss of the anti-rotation capability of the nut? Can cracking in this area be easily detected during routine inspections? If the nut were to rotate, how can the tie rod preload be verified?

7. The membrane plus bending stresses in the reactor vessel wall due to the radial seismic support loads have been evaluated as secondary membrane plus bending stresses. However, stresses due to seismic or LOCA loads in the vessel should be considered as primary stresses since they are not self-limiting in nature. Therefore, provide an evaluation of these stresses as primary stresses. In addition, the radial seismic supports will also cause shear stresses in the vessel wall. Provide an evaluation of the shear stresses as they interact with the principal stresses.

8. In the evaluation of the stresses in the shroud support plate and reactor vessel due to the tie rod repair, the loading combinations from the original design report were evaluated to determine the allowable tie rod loads. However, for the design of the tie rods themselves, it appears that the more conservative loading combination of a LOCA and seismic event were evaluated. Therefore, it would appear that the support plates and the reactor vessel should also be evaluated for these conservative loading combinations since they are required to restrain the tie rod loads. Provide stress calculations that demonstrate the structural adequacy of the support plates and reactor vessel for these more conservative load combinations. Also, since the bending and shear stresses were evaluated separately, provide an analysis of the interaction between these stress components.

9. Similar to above question, it appears that in determining the allowable loads on the reactor vessel from the radial seismic supports, the more conservative loading combination of LOCA and seismic loads together was not considered in the original design report. Provide the revised allowable loads considering this more conservative load combination.

10. For the recirculation line break LOCA, spatial and time varying lateral pressure loadings are generated in the shroud/reactor vessel annulus. It is stated that the initial acoustic phase of the transient is very abrupt relative to the shroud inertia and frequencies and does not have

a significant effect on the shroud. The remainder of the transient extends over a relatively long period of time and is considered as a static pressure loading. Provide the analytical justification for neglecting the initial acoustic phase of the transient.

11. The tie rod assembly is attached to the shroud support plate at the lower end. Failure of welds H8 and H9 would affect the functionality of the tie rod assemblies. What are the minimum lengths of the welds H8 and H9 in the vicinity of the attachments needed to assure the functionality of the tie rod assemblies?

12. Provide the calculations performed to obtain the results summarized in Enclosure C (Tables 9 and 10) of the design report related to the following tie rod component stresses:

    a. Table 9, Service Level B (Load Case: Thermal Transient)
       - Upper Radial Support, axial load through bracket shelf
       - Bracket, on the ledge at connection to upper shroud flange and the axial contact of restraint at ledge
       - Nut, at threaded connection to top adapter, and interface with radial support

    b. Table 10, Service Level B (Load Case: OBE)
       - Bottom Adapter shaft
       - Spring Rod

13. Provide your plans for inspection of the individual tie rod components subsequent to installation.

14. The tie rod construction is such that many load-bearing parts contain crevices and, as such, may be subject to crevice corrosion at the mating surfaces. Provide test data which indicates the susceptibility of these components in their particular configurations to more fully address the potential for these components to undergo crevice corrosion. How will degradation due to crevice corrosion be monitored?

15. In the fabrication of the tie rod assemblies, please identify any controls in the machining, grinding or threading of the parts for the purpose of minimizing surface cold work.

16. In the fabrication of the tie rod assemblies, please identify any heat treatment (e.g., solution annealing or stress relieving) being performed on parts after machining, grinding or threading for the purpose of removing or minimizing surface residual stresses.

17. The Design Specification 249001-001 states that "the design input values are based on current operation conditions. In addition, the repair

design shall consider an increase in core shroud pressure differential of 20%." (Pg. 5-10). However, pg A-2 of the same specification, Note 2, states that "differential pressure values in parentheses include a factor of 1.15 to account for the potential of a future increase in core flow and/or a power uprate." The values on pg A-2 appear to be the values used in the calculations. Is 15% increase in differential pressure the intended value instead of 20%? What does the 1.15 increase in differential pressure represent in terms of rated core power and flow (e.g. 110% rated core power and 110% rated core flow)?

18. Page 9 of Calculation Number 2499502-404, Section 4.1.6, lists the maximum normal operating differential pressures in the upper, mid, and lower core regions based upon power uprate conditions. These values are used to calculate the mean flow velocity which in turn is used to calculate the flow rate through postulated circumferential weld cracks. The value given for the lower core region differential pressure is 28.75 psi. On page A-2 of Design Specification 249001-001, Note 3, 28.75 psi is listed as the maximum core plate differential at any time in the main steam line break transient. Please explain why the maximum main steam line break differential pressure is used in the calculations instead of the maximum normal/upset differential pressure as listed in the table on page A-2.

19. Page 13 of the BWRVIP "Guide for Format and Content of Core Shroud Repair Design Submittals (BWRVIP-04)," Section 3.2.5.1 describes the requested information about bypass flow with regards to the repair. Page 13 states, "this section should provide a discussion of the new bypass flow paths introduced by the repair and an evaluation of the increased bypass flow due to the installation of the repair. Also, an evaluation of the increase in bypass flow due to the limiting postulated shroud weld cracking should be provided. Any other sources of leakage (e.g. access hole covers) should be included in the evaluation. The leakage flows should be provided for rated conditions and maximum and minimum core flow conditions..." Confirm that all sources of leakage applicable to Vermont Yankee were considered in the evaluation and explain how leakage due to access hole covers was addressed in the evaluation.

20. Page 6-1 of the Vermont Yankee Nuclear Power Station Core Shroud Repair Design Report, Volume 1: Design Summary Report states that "a small temporary vertical separation could occur during an SSE or SSE plus main steam line break event." What is the magnitude of the temporary separation? At which circumferential weld does this postulated separation occur?

July 9, 1996

Therefore, we request that you respond within 30 days of the date of this letter.

If you have any questions, please contact me at (301) 415-1429.

Sincerely,

ORIGINAL SIGNED BY:

Daniel H. Dorman, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

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