SNUPPS
Standardized Nuclear Unit
Power Plant System

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SLNRC 81-076 FILE: 0290
SUBJ: CPB (Reactor Fuel) Review

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: STN 50-482, STN 50-483, and STN 50-486

Dear Mr. Denton:

In discussions with Dr. Gordon Edison, NRC project manager for the SNUPPS applications, it was learned that additional information was required in order for the Reactor Fuel Section of the Core Performance Branch to complete their review. The purpose of this letter is to provide that information.

1. Concerning rod worth tests that would detect leakage from boron-containing control rods, the absorber material used for the SNUPPS control rods will be either all hafnium or all Ag-In-Cd. Boron-containing control rods are not planned for use in the SNUPPS plants.

2. Concerning cladding collapse time, Westinghouse used an NRC-approved cladding collapse model (WCAP-8377, 8381, Reference 6 on SNUPPS FSAR page 4.2-42) for the SNUPPS first core. This same model is routinely used for reloads. The model provides assurance that cladding collapse will not occur for the lifetime of the fuel assembly.

3. Enclosure A provides the required supplemental ECCS calculations.

4. Concerning NUREG-0609, Appendix E, the following information is provided. The methodology and analytical techniques given in Section 3.0 of WCAP 9402, Verification, Testing and Analyses of the 17x17 Optimized Fuel Assembly" apply to, and are presently used in, the analysis of Westinghouse 17x17 Inconel Grid Fuel Assemblies (i.e., SNUPPS type fuel assemblies). This report augments the information presented in WCAP-8236. However, the analysis results are only applicable to a number of Westinghouse four loop plants with full cores of optimized fuel.

   Based upon a preliminary assessment of the seismic design of the SNUPPS plants, Westinghouse is confident that upon completion of the analysis it will be demonstrated that the SNUPPS core can maintain a geometry capable of being cooled during a postulated seismic or LOCA event.

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With respect to the requirements of NUREG-0609, Appendix E (which is merely a restatement of proposed Appendix A to SRP 4.2, Revision 1), Westinghouse has demonstrated that a simultaneous SSE and LOCA event is highly unlikely. The fatigue cycles, crack initiation and crack growth due to normal operating and seismic events will not realistically lead to a pipe rupture (See WCAP-9283). The factor applied to the LOCA grid impact load due to flashing is considered unrealistic since the thermal/hydraulic conditions for flashing are not present at the time of peak grid impact load. Nevertheless, the combined LOCA and SSE loads will be provided, and the combined values are expected to be below the established limits.

5. Concerning the on-line fuel monitoring system, refer to FSAR Section 11.5.2.2.2.6.

6. Concerning post-irradiation surveillance of fuel, the following information is provided. The Standard Review Plan on page 4.2-12 states:

   "The extent of an acceptable (post-irradiation fuel surveillance) program will depend on the history of the fuel design being considered, i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features.

   "For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling."

   As detailed in WCAP-8183, Revision 10 ("Operational Experience with Westinghouse Cores - up to 12/31/80), significant 17X17, 12-foot, Inconel grid fuel assembly operating experience has been obtained with two plants completing two cycles of operation and three plants completing one cycle of operation. Region average burnups up to 29,000 MWD/MTU were obtained and, in general, the 17X17 fuel assemblies were in excellent condition.

   Therefore it is currently anticipated that post-irradiation poolside surveillance of the SNUPPS fuel assemblies will not exceed the above suggested nominal visual examinations.

7. The NRC questioned the applicability to SNUPPS of questions and responses on WCAP-9500. The Westinghouse responses to NRC Questions on WCAP-9500, "Reference Core Report 17X17 Optimized Fuel Assembly," Section 4.2, "Mechanical Design" (transmitted by Westinghouse letters to the NRC dated January 12, 1981 and April 21, 1981) are generally applicable to Westinghouse 17X17 Standard Inconel grid fuel assemblies (i.e. the type to be used in the Callaway first core). The design bases, design limits, and design evaluation methods used in the mechanical design of standard fuel are unchanged for the mechanical design of optmized fuel. All responses given

in the above correspondence related to core components (i.e., control rods, burnable poison rods, source rods, and thimble plugs) are directly appliable to the SNUPPS core components (with the exception of responses related to B_4C control rods). Certain questions and responses are concerned with design features unique to optimized fuel (e.g. Q231.14 regarding OFA guide tubes which have a reduced ID, etc.). Obviously, these responses are not applicable to SNUPPS fuel.

In summary, since the majority of the WCAP 9500 Section 4.2 responses were intended to demonstrate Westinghouse compliance with SRP 4.2, Revision 1, the responses generally apply to all Westinghouse fuel types.

Very truly yours,

N A Petrick

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Enclosure A to SLNRC 81-076

The Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 to operators of light water reactors regarding fuel rod models used in Loss of Coolant Accident (LOCA) ECCS evaluation models. That letter describes a meeting called by the NRC on November 1, 1979 to present draft report NUREG-0630, "Cladding Swellng and Rupture Models for LOCA Analysis." At the meeting, representatives of NSSS vendors and fuel suppliers were asked to show how plants licensed using their LOCA/ECCS evaluation model continued to conform to 10 CFR Part 50.46 in view of the new fuel rod models presented in draft NUREG-0630. Westinghouse representatives presented information on the fuel rod models used in analyses for plants licensed with the Westinghouse ECCS evaluation model and discussed the potential impact of fuel rod model changes on results of those analyses. That information was formally documented in letter NS-TMA-2147, dated November 2, 1979, and formed the basis for the Westinghouse conclusion that the information presented in draft NUREG-0630 did not constitute a safety problem for Westinghouse plants and that all plants conformed with NRC regulations.

As a result of compiling information for letter NS-TMA-2147, Westinghouse recognized a potential discrepancy in the calculation of fuel rod burst for cases having clad heatup rates (prior to rupture) significantly lower than 25 degrees F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979, and although independent of the NRC fuel rod model concern, the combined effect of this issue and the effect of the NRC fuel rod models had to be studied. Details of the work done on this issue were presented to the NRC on November 13, 1979 and documented in letter NS-TMA-2163 dated November 16, 1979. That work included development of a reevaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this reevaluation, the Westinghouse position on NUREG-0630 was reviewed and it was still concluded that the information presented in draft NUREG-0630 did not constitute a safety problem for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979 NRC and Westinghouse personnel discussed the information thus far presented. At the conclusion of that discussion, the NRC staff requested Westinghouse to provide further detail on the potential impact of modifications to each of the fuel rod models used in the LOCA analysis and to outline analytical model improvements in other parts of the analysis and the potential benefit associated with those improvements. This additional information was compiled from various LOCA analysis results and documented in letter NS-TMA-2174 dated December 7, 1979.

Another meeting was held in Bethesda on December 20, 1979 where NRC and Westinghouse personnel established: 1) The currently accepted procedure for assessing the potential impact on LOCA analysis results of using the fuel rod models presented in draft NUREG-0630 and 2) Acceptable benefits resulting from analytical model improvements that would justify continued plant operations for the interim until differences between the fuel rod models of concern are resolved.

Encl. A

The information following on pages 3 - 6 satisfies the NRC request for information on SNUPPS.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in letter NS-TMA-2175, dated December 10, 1979, which contains Westinghouse comments on draft NUREG-0630. As stated in that letter, Westinghouse believes the current Westinghouse models to be conservative and to be in compliance with Appendix K.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for SNUPPS.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT 0.6

WESTINGHOUSE ECCS EVALUATION MODEL VERSION Feb. '78

CORE PEAKING FACTOR 2.32

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 1906.2 $^oF = PCT_B$

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2088.3 $^oF = PCT_N$

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 2.44748 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 2.44748 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is less than 1.0 inch per second and reflood heat transfer is based on the steam cooling calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 6.0 Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 46.6 Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200°F, individual effects (such as Δ PCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges, but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200°F justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- $0.01\ \Delta FQ = \sim 150^{o}F$ BURST NODE ΔPCT

- Use of the NRC burst model and the revised Westinghouse burst model could require an FQ reduction of 0.027

- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (0.027+ .03)\ (150^{o}F/.01) = 855^{o}F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.^{o}F - PCT_B = \underline{293.8}\ ^{o}F$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\,\Delta FQ}{150^{o}F}\right)$$

$$= (\underline{855}-\underline{293.8})\ \left(\frac{.01}{150}\right)$$

$$= \underline{0.037}$$ (but not less than zero).

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generically in this evaluation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of

RCS blowdown" indicated above. (Note: For this case this difference is zero. To be conservative, it will be assumed that the new burst model will result in no clad swelling prior to burst).

Therefore.

$$\Delta PCT_3 = \left(\frac{20^oF}{.01 \text{ strain}}\right) \text{(MAX STRAIN - BLOWDOWN STRAIN)}$$

$$= \left(\frac{20}{.1}\right) (\underline{2.45} - \underline{\ \ 0\ \ }) \left(\frac{1}{.100}\right)$$

$$= \underline{\quad 49 \quad}$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore:

$$\Delta PCT_4 = 1.25^oF \text{ (50 - PERCENT CURRENT BLOCKAGE)} + 2.36^oF \text{ (75-50)}$$

$$= 1.25 (50 - \underline{46.6}) + 2.36 (75-50)$$

$$= \underline{63.3}\ ^oF$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 112.3^oF$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^oF - PCT_N = 111.7^oF$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6) \left(\frac{.01 \Delta FQ}{10^oF \Delta PCT}\right) = 0.00$$

ΔFQ_N = ____0____ but not less than zero.

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N.

or, $\Delta FQ_{PENALTY}$ = __0.037__

B. The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate ΔFQ credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero).

FQ ADJUSTMENT = $\underline{\ 0.20\ }$ - $\underline{\ 0.037\ }$ = + 0.163

Since the FQ Adjustment is greater than zero, there is no penalty in FQ required.

FQ = 2.32 - 0 = 2.32