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50-323 Diablo Canyon Nuclear Power Plant, Unit 2, Pacific Ga 05000323
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RECIP.NAME RECIPIENT AFFILIATION
MIRAGLIA,F.J. Licensing Branch 3

SUBJECT: Forwards Westinghouse burst & blockage calculations in response to NRC 791109 ltr re fuel rod models used in LOCA ECCS evaluation models.Current Westinghouse models are conservative & comply w/10CFR50,App K.

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March 10, 1981

Mr. Frank J. Miraglia, Jr., Chief
Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Docket No. 50-275
Docket No. 50-323
Diablo Canyon Units 1 and 2

Subject: ECCS Evaluation Model Concerning Cladding Swelling

Dear Mr. Miraglia:

The Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 to owners of operating 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 Swelling 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 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 the 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 concern for Westinghouse plants, and that all such plants conformed with NRC regulations.

In the November 9, 1979 letter the NRC requested that owners of operating light water reactors provide, within sixty (60) days, information

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to enable the NRC to determine, in light of the fuel rod model concerns, whether or not further action would be necessary. More recently the NRC has requested similar evaluations for NTOL plants. Enclosed is information on the LOCA analysis of the Diablo Canyon Power Plants in response to that request. Note, however, that a significant amount of discussion and information exchange between Westinghouse and the NRC has transpired since the November 2, 1979 letter (NS-TMA-2147) was prepared, and the basis for demonstrating compliance with 10 CFR part 50 has been modified. The following is an outline of the significant events that occurred since November 2, 1979 and is provided to update you on the situation.

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°F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979. Although independent of the NRC fuel rod model concern, the combined effect of this issue and 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 procedure to determine the clad heatup rate prior to burst and a re-evaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this re-evaluation, 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 concern for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979, NRC and Westinghouse personnel discussed the information 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 these 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 subsequently held in Bethesda on December 20, 1979. At this meeting NRC and Westinghouse personnel established (i) the currently accepted procedure for assessing the potential impact on LOCA analysis results using the fuel rod models presented in draft NUREG-0630 and (ii) acceptable benefits resulting from analytical model improvements that would justify continued plant operation for the interim until differences between the fuel rod models of concern are resolved.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in the Westinghouse letter NS-TMA-2175, dated December 10, 1979. That letter contains comments

on draft NUREG-0630. As stated in the letter, the current Westinghouse models are believed to be conservative and to be in compliance with 10 CFR 50, Appendix K.

Kindly acknowledge receipt of the material listed above on the enclosed copy of this letter and return it to me in the enclosed addressed envelope.

Very truly yours,

Philip A. Crane, Jr.

Enclosure

CC w/enclosure: Service List

ATTACHMENT
WESTINGHOUSE BURST AND BLOCKAGE
CALCULATIONS

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

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

BREAK TYPE - DOUBLE-ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT 0.8

WESTINGHOUSE ECCS EVALUATION MODEL VERSION February 1978

CORE PEAKING FACTOR 2.32

HOT ROD MAXIMUM TERMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD

2187 $^{o}F = PCT_B$

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NONRUPTURED REGION OF THE CLAD - 2020 $^{o}F = PCT_N$

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION - 1.16 Percent

MAXIMUM CLAD STRAIN AT THIS ELEVATION - 7.54 Percent

Maximum temperature for this nonburst node occurs when the core reflood rate is GREATER than 1.0 inch per second and reflood heat transfer is based on the FLECHT calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 6.0 Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 44.4 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 $2200^{o}F$ and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above $2200^{o}F$, individual effects 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 $2200^{o}F$ justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- $0.01\ \Delta FQ \rightarrow \sim 150^{o}F$ 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 $2200^{o}F$ limit is:

$$\Delta PCT_2 = 2200^{o}F - PCT_B = \underline{\ 13\ }\ ^{o}F,\ (PCT_B = 2187^{o}F)$$

The FQ reduction required to maintain the $2200^{o}F$ clad temperature limit is:

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

$$= \left(\frac{855}{} - \frac{13}{}\right)\ \left(\frac{.01}{150}\right)$$

$$= \underline{0.056}\ \text{(but not less than zero).}$$

2. <u>NONBURST NODE</u>

The maximum temperature calculated for a nonburst 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 of 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 nonburst 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. Therefore,

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

$$= \left(\frac{20}{.01}\right) (\underline{7.54} - \underline{1.16}) \times 10^{-2}$$

$$= \underline{\quad 127.6 \quad}$$

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,

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

$$= 1.25\ (50 - \underline{44.4}) + 2.36\ (75\text{-}50)$$

$$= \underline{\ 66\ }\ ^{\circ}F$$

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 nonburst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 127.6 + 0.0 = 127.6$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^{\circ}F - PCT_N = 2200 - 2020 = 180$$

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

$$\Delta FQ_N = \overset{127.6\ -\ \ 180}{(\Delta PCT_5 - \Delta PCT_6)}\ \left(\frac{.01\Delta FQ}{10^{\circ}F\ \Delta PCT}\right) < 0.0$$

$$\Delta FQ_N = \underline{\ 0\ }$$

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

or; $\Delta FQ_{PENALTY} = \underline{0.056}$

B. <u>FQ BENEFIT</u> - The effect on LOCA analysis results of using improved analytical and modeling techniques (which are approved for use on several Westinghouse fueled plants) 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 four-loop plants is an increase in the LOCA peaking factor limit of <u>0.20</u>.

C. <u>FQ ADJUSTMENT</u> - 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 (<u>but not greater than zero</u>).

$$FQ\ ADJUSTMENT = 0.20 - 0.056 > 0$$

$$FQ\ ADJUSTMENT = 0.0$$

[illegible]