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NS-TMA-2174

Mr. Darrell G. Eisenhut
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Operating Reactor
Nuclear Regulatory Commission
7920 Norfolk Avenue
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cc: R. Dinese

Dear Mr. Eisenhut:

Letter NS-TMA-2174, dated November 2, 1979 provided Westinghouses' response to three questions related to the fuel rod models used in Appendix K analyses.

On December 6, 1979, members of the NRC staff and Westinghouse personnel discussed the contents of that letter. The NRC staff advised Westinghouse that the response presented in that letter was not sufficient and additional information was required. This letter provides that additional information.

The three questions are rewritten below and addressed.

Question 1

In light of the data presented in the NRC draft report (NUREG-0630) do the Westinghouse fuel rod models meet 10 CFR 50 Appendix K?

Response 1

Yes.

The data (not the models) presented in (draft) NUREG-0630 were consistent with the data base used to develop the Westinghouse fuel rod models. A Westinghouse letter containing detailed comments on draft NUREG-0630 will be issued December 10, 1979.

Question 2

If the NRC fuel rod models were used in the Westinghouse ECCS evaluation model, would the affected plants meet 10 CFR 50.46 limits?

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Response 2

If the most recent loss of coolant accident (LOCA) analysis for each plant was modified to factor in the NRC fuel rod models, the peak clad temperature calculated in that modified analysis, in several cases, would be greater than the currently reported value. If the peak clad temperature (PCT) from the current analysis is sufficiently close to or at the 10 CFR 50.46 limit of 2200°F, and the behavior of the particular limiting break LOCA transient was such that the modified fuel rod models would impact the transient, then 10 CFR 50.46 limits would not be met at the current technical specification peaking factor limit.

Modifications to the fuel rod models could impact the LOCA analysis results for several reasons. Those reasons, and the estimated effect (on PCT) of changes to the fuel rod models follow.

Fuel rod burst:

Figure 47 of draft NUREG-0630 compares the Westinghouse burst curve from the February, 1978 version of the Westinghouse evaluation model to the NRC proposed burst model. Subsequent to preparation and submittal of NS-TMA-2147, Westinghouse recognized an apparent deficiency in the February, 1978 model in that the heatup rate dependence on burst was not properly considered. Westinghouse then performed a detailed study, including plant specific reanalyses in some cases, to evaluate the impact of the heatup rate dependence of fuel rod burst. Results of that evaluation were presented in letter NS-TMA-2163, dated November 16, 1979. Is it therefore appropriate to consider the impact of using the NRC burst model in place of the revised Westinghouse heatup rate dependent burst model discussed in NS-TMA-2163. A rough comparison of these models was made by superimposing Westinghouse burst curves for 10°F/sec and 25°F/sec heatup rates with the NRC curves at those heatup rates. The 10°F/sec heatup rate NRC curve was estimated by linearly interpolating between the NRC 0°C/sec and 25°F/sec curves. (The 10°F/sec heatup rate was selected as a lower bound for comparison. Letter NS-TMA-2163 justifies that the 10°F/sec lower bound is reasonable for this purpose). Restricting the comparison to the applicable 5 to 8 KPSI range, the maximum difference between comparable heatup rate curves occurs for the 10°F/sec cases at 8 KPSI. That difference is approximately 37°C (67°F), i.e., the NRC curve shows a lower burst temperature than does the Westinghouse curve by about 67°F.

Two areas of the LOCA analysis could be impacted by calculating burst at a lower clad temperature. They are:

1. The highest power rod (hot rod) is the rod that will experience the maximum clad temperature. The calculations of burst on the hot rod has a very localized effect in that it only changes the transient temperature calculation of that burst node. In some cases, the PCT occurs at that hot rod burst node. A set of three LOCTA code computer calculations, recently performed, can be used to establish the impact on the burst node maximum temperature and is described as follows:

   Run #1 - Used the Westinghouse 25°F/sec heatup rate burst curve

   Run #2 - Used the Westinghouse 10°F/sec heatup rate burst curve

   The only difference between runs #1 and #2 is the burst model. This is a burst node limited case and PCT for run #2 was 186°F greater than the PCT in run #1. (This extreme sensitivity on the burst node is not uncommon and is explainable). The shift in burst temperature in going from the 25°F/sec to the 10°F/sec heatup rate burst curve is approximately 54°F. It is therefore estimated that the maximum impact on the burst node of Westinghouse plants from using the NRC burst model is

   $$\left(\frac{67}{54}\right)(189) = 235^{\circ}F$$

   on the burst node itself. Run #3 of the set of LOCTA calculations shows that a peaking factor reductin of 0.01 reduces the PCT by 149°F. Therefore, for a plant which is burst node limited and for which the current technical specification peaking factor results in a PCT of 2200°F, the maximum peaking factor reduction required to stay within 10CFR50 limits is

   $$\frac{235^{\circ}F}{149^{\circ}F/.01\ \Delta FQ} = 0.015$$

   Note that the PCT changes between runs 1, 2 and between runs 2, 3 cover the same range which is in the neighborhood of 2200°F. Therefore, the sensitivity can be expressed in terms of ΔFQ_{LIMIT} as a linear function of burst curve shift.

2. The second area of the LOCA calculations that could be impacted by burst occuring at a lower clad temperature is the burst of the hot assembly. Burst in the hot assembly is calculated for a representative average hot fuel assembly rod (AVG rod) and is used to determine flow redistribution from the hot assembly for the fluid enthalpy calculation during the reflood phase of the accident resulting from blockage of the hot assembly.

The potential impact of changing burst characteristics of the AVG rod cannot be fully quantified unless the magnitude of hot assembly blockage is considered. This impact will be estimated below in the discussion of Blockage.

Strain:

Figure 50 of draft NUREG-0630 compares the NRC composite burst strain model and the Westinghouse strain model.

Inspection of several LOCA analysis results show that burst occurs below 1600°F (871°C). This is from calculations using the Westinghouse burst temperature model. Using the NRC burst model could result in yet lower burst temperatures. It is therefore sufficient to evaluate the differences between the two strain models below 900°C.

As with the hot rod burst calculation, the strain model has the potential to effect only the hot rod burst node temperature. The impact of a modification to the amount of burst strain is difficult to assess because of competing effects. One effect is that the larger strain results in a larger clad surface area being available for reaction with steam which results in a larger rate of heat generated by reaction which increases the clad temperature. The larger strain also results in a higher resistance to heat flow from the fuel pellet to the clad which reduces the clad temperature. Since the magnitudes of these competing effects are not known at this time, the net impact on PCT cannot be quantified. Furthermore, since the curves of the Westinghouse and NRC strain models cross in the range of interest, the net impact estimate is further complicated. In attempt to address this issue, we estimate that the impact on the peaking factor limit ranges from a slight benefit to a possible peaking factor reduction of 0.03.

Blockage:

The impact of a change in the magnitude of blockage has been established from the results of several LOCTA code calculations:

a. Increase of blockage from 44% to 55% increased PCT 17°F.

b. Increase of blockage from 36% to 55% increased PCT 18°F.

c. Increasing blockage from 47% to 75% increased PCT by 66°F.

It appears, from these results, that at "lower" (up to 55%) blockage levels, PCT increases 1.0 to 1.5°F per percent increase in blockage. Blockage increases at the "higher" end of the blockage spectrum (50 to 75%) result in a PCT increase of 2.36°F per percent blockage. In attempting to bound the impact of using the NRC fuel rod models, let

Δ PCT from the current blockage value to 50% blockage equal 1.25°F per percent

Δ PCT from 50% blockage to 75% blockage equal 2.36°F per percent.

The maximum expected impact of using the NRC fuel rod models can be conservatively determined by using the above sensitivities and increasing the current value of blockage to the maximum value shown in the NRC model, or to 75%. The Westinghouse plants impacted by this change show current levels of blockage ranging from a minimum value of 32% to maximum values near 47%.

The PCT increase in the worst case is estimated to be

$$(1.25)\ (50-32) + (2.36)\ (75-50) = 81.\ ^{\circ}F$$

Based on the above estimates, the NRC burst (as applied to the HOT rod) models would have the greatest impact on plants that are currently burst node limited. For plants in this category, the maximum impact is estimated to be a required reduction in peaking factor of approximately 0.015. In order to experience that impact the following conditions must be met:

- The clad heatup rate must be approximately 10°F/sec at burst time.
- The clad temperature must be in the approximate range of 750 to 825°C at burst time.
- The current analysis must indicate a PCT of 2200°F at the technical specification peaking factor limit.

The maximum estimated impact of using the NRC strain model is a peaking factor reduction of 0.03 for burst node limited plants only.

If a plant is currently not burst node limited, there is a possibility that changes to the fuel rod models would cause the burst node to become limiting. However, the impact would be less than that of a plant currently burst node limited.

Use of the NRC burst (as applied to the AVG rod) and blockage models has the greatest potential impact on plants that are currently "steam cooling" limited. The maximum impact is estimated to be an increase in PCT of 81°F

In summary, use of the NRC fuel rod models in place of the fuel rod models in the February, 1978 version of the Westinghouse ECCS evaluation model is estimated to have the following maximum impact:

- The change in the calculation of the burst node temperature transient could require a 0.015 FQ reduction to account for using the NRC burst model and a 0.03 FQ reduction to account for using the NRC strain model for a total maximum of 0.045 reduction.

- The change in the calculation of the limiting non-burst node could result in a PCT increase of 81°F.

Note that these effects are <u>NOT</u> <u>ADDITIVE</u>. <u>Only</u> one of the penalties can effect a given plant.

<u>Question 3</u>

If any plants do not meet 10CFR50.46 limits, what is Westinghouse's interim and long term recommendation for resolution?

<u>Response 3</u>

<u>Interim Recommendation:</u>

The maximum impact on the burst node was estimated to cause a peaking factor reduction of 0.045. An analysis for a Westinghouse plant has previously been performed, and is currently being reviewed by the NRC, which accounts for a reduction in the uncertainty of the average fuel temperature used in the LOCA anlalyses. The change was made to a burst node limited case and indicated that the reduction in fuel temperature uncertainty increases the limting peaking factor by approximately 0.05. The Δ FQ increase for a non-burst node limited plant would be at least 0.02.

Another analysis, for a Westinghouse plant, which uses modifications to the drift flux calculations in the SATAN computes code has also been submitted to the NRC for review. That analysis showed an allowable peaking factor increase of 0.27 <u>and</u> a decrease in PCT of 127° due to the modification.

Another analysis was performed by a Westinghouse plant using the BART computer code to calculate heat transfer from the fuel rods during the core reflood phase of the accident. This analysis showed a PCT reduction for the limiting break of 94°F. This analysis was recently completed and will be sumbmitted to the NRC. It should be noted that this model change specifically effects the reflood phase of the analysis.

The three LOCA analysis model improvements mentioned above apply to all Westinghouse plants. Although the benefits associated with each change will vary for different plant types, we are confident that application of these model changes would affect the maximum impact of the NRC fuel rod models if applied, and would certainly compensate for any plant specific impact of the NRC fuel rod models.

Therefore, Westinghouse interim recommendations is to not impact current plant operational limits as justified by the analytical model improvements outlined above.

<u>Long Term Recommendations:</u>

Westinghouse recommends that the Westinghouse comments on draft NUREG 0630 (to be submitted December 10, 1979) be given due consideration so that fuel rod behavior is appropriately modeled in LOCA analysis codes such that Appendix K requirements are met and also such that the conservatism associated with those models is justified.

One portion of the results of the Westinghouse review of the NRC model is indicated in the attached Figure 11. (Figure 11 is taken from the draft letter of Westinghouse comments on draft NUREG 0630 which will be submitted December 10, 1979). Figure 11 shows the Westinghouse intrepretation of the combination of the ORNL blockage data and Westinghouse Single Rod Burst Test data in terms of blockage for 5°F/sec and 25°F/sec heatup rates as a function of clad hoop stress at rupture time.

If a linear interpolation between the 5°F/sec and 25°F/sec curves is performed to estimate the position of a 10°F/sec curve, it can be observed that, up to approximately 8 KPSI hoop stress which is roughly the upper bound hoop stress for Westinghouse plants, the current Westinghouse blockage curve is close to that 10°F/sec curve. Since 10°F/sec approximately the lower bound of heatup rate in current LOCA analyses, if the Westinghouse interpretation of the blockage data is correct, there is no significant penalty associated with use of the ORNL blockage data as Westinghouse concluded previously in the letter NS-TMA-2147. We believe that this is in fact the case.

Very truly yours,

T. M. Anderson, Manager

Nuclear Safety Department