William J. Cahill, Jr.
Vice President

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January 24, 1980

re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

In response to your letters of November 9 and November 27, 1979, Con Edison's January 8, 1980 submittal addressed the recent communications between the NRC and NSSS vendors regarding those portions of ECCS evaluation models dealing with fuel cladding swelling, the incidence of rupture and fuel assembly blockage. As stated therein, we reviewed the fuel models used in our current ECCS calculations and those proposed in draft NUREG-0630 and determined that additional calculations would be performed to demonstrate compliance within the limits of 10 CFR 50.46. Accordingly, Westinghouse has performed a plant specific evaluation of the potential impact of using fuel models presented in draft NUREG-0630 on the LOCA/ECCS analysis for Indian Point Unit No. 2. The reevaluation indicates that all acceptance criteria of 10 CFR 50.46 are still met at the present peaking factor (F_Q) limit of 2.31. The details of the plant specific reevaluation are presented in Attachment 1 to this letter.

Should you or your staff have any further questions, please contact us.

Very truly yours,

William J. Cahill, Jr.
William J. Cahill, Jr.
Vice President

attach.

Subscribed and sworn to before me this 24 day of January, 1980.

Angela Roberti
Notary Public
ANGELA ROBERTI
Notary Public, State of New York
No. 41-8593813
Qualified in Queens County
Commission Expires March 30, 1980

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ATTACHMENT 1

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 Indian Point Unit 2.

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* and [1.0]**

CORE PEAKING FACTOR 2.31 [2.31] - used the February, 1978 model

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2024.°F = PCT_B [2137]

ELEVATION - 6.0 Feet [6.0]

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2135.°F = PCT_N [2066]

ELEVATION - 7.5 Feet [7.25]

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 0.4 Percent [4.16]
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 3.2 Percent [10.0]

Maximum temperature for this node occurs when the core reflood rate is greater than 1.0 inch per second and reflood heat transfer is [same] based on the flecht calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 6.25 Feet [6.0]

HOT ASSEMBLY BLOCKAGE CALCULATED - 42.0 Percent [35.9]

NOTES:

* The Modified February 1978 Westinghouse ECCS Evaluation Model version was used for the C_D=0.6 case. Specifically, the fuel rod burst model was modified to factor in heatup rate dependence as documented in WCAP-8970-P-A "Westinghouse Emergency Core Cooling System Small Break, October 1975 Model." Fuel rod burst curves used in this analysis represented clad heatup rates of 10 Degrees F/Second for the Hot Rod and 10 Degrees F/Second for the Average Hot Assembly Rod. In addition, the water residing in the accumulator lines, not accounted for in the previous analysis (December, 1978 - IP2 LOCA Analysis - break spectrum for 6% uniform steam generator tube plugging), was taken credit for.

** The C_D = 1.0 break from the latest break spectrum (6% steam generator tube plugging) was also evaluated. Those numbers are in brackets [ ]. For this analysis, the heatup rate prior to burst (for the burst node of the hot rod) was evaluated and the burst curve used in the analysis was determined to be appropriate (i.e., approximately 25F/sec).

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 ΔPCT 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:

- $0.01\ \Delta FQ \rightarrow \sim 150°F$ BURST NODE ΔPCT
- Use of the NRC burst model could require an FQ reduction of 0.015
- 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:

$$\Delta PCT_1 = (.015 + .03)\ (150°F/.01) = 675°F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.°F - PCT_B = 176.°F \quad [63.]$$

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

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

$$= (675 - 176)\ \left(\frac{.01}{150}\right)$$

$$= 0.033 \text{ (but not less than zero).} \quad [0.0408]$$

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 during blowdown" indicated above.

Therefore:

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

$$= \left(\frac{20}{.01}\right)(.032 - .004)$$

$$= 56.°F \quad [117.]$$

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,

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

$$= 1.25\ (50 - 42) + 2.36\ (75-50) \quad [NA]$$

$$= 69°F$$

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 non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 56 + 0 = 56.°F \quad [117.]$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200°F - PCT_N = 65°F \quad [134.]$$

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

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

$$\Delta FQ_N = -.009 \text{ but not less than zero.} \quad [-.017]$$
$$= 0 \quad [= 0]$$

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N, or; $\Delta FQ_{PENALTY} = 0.03$ [0.041]

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).

$$FQ\ ADJUSTMENT = 0.20 - 0.03 \quad [0.20 - 0.041]$$
$$\rightarrow 0 \quad [\rightarrow 0]$$