POWER AUTHORITY OF THE STATE OF NEW YORK
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January 29, 1980
IPN-80-9

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
ECCS Reanalysis

References: (1) Authority Letter from Mr. P. J. Early to Mr. A. Schwencer (IPO-86), dated April 13, 1978.
(2) NRC Letter from Mr. D. Eisenhut to All Operating Light Water Reactors, dated November 9, 1979.
(3) NRC Letter from Mr. D. Eisenhut to All Operating Light Water Reactors, dated November 27, 1979.

Dear Sir:

The Authority's ECCS analysis of record for the Indian Point 3 facility was transmitted to the NRC via Reference (1). This acceptable Appendix K ECCS was based on the NRC approved Westinghouse October 1975 evaluation model with the appropriate Zr-H_2O reaction. This analysis was performed based on 0% steam generator tube plugging. The Authority has plugged up to 3.65% of the tubes in the IP-3 steam generators during the current refueling outage. The Authority requested the Westinghouse Electric Corporation to provide a new Appendix K ECCS analysis based on the NRC approved February 1978 evaluation model considering 4% steam generator tube plugging. This new analysis has been completed and demonstrates that the IP-3 facility complies with the requirements of 10 CFR §50.46 and no changes to the Technical Specifications are required. Forty copies of this analysis are enclosed for your review.

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In accordance with the provisions of 10 CFR §170.22, the Authority classifies this application for review as Class III, since it involves the single safety issues of the ECCS reanalysis. A check in the amount of $4,000 is enclosed, which the Authority pays under protest pending a final determination of the legality of the fee schedule.

References (2) and (3) requested the Authority to provide information on the impact of Draft NUREG-0630 on current ECCS analyses considering new fuel clad rupture and flow blockage effects. The Authority requested Westinghouse to perform a sensitivity study, which is included as an Attachment to this letter. This study included the models presented in Draft NUREG-0630, as well as acceptable benefits from analytical model improvements which could be utilized until differences between the NRC's fuel rod model and Westinghouse's model are resolved. The study demonstrates that the net impact on current IP-3 plant Technical Specification limits is zero. This study constitutes the Authority's response to References (2) and (3). It is anticipated that when the current situation concerning differences in heatup rates and clad burst models is resolved, the Authority will request Westinghouse to perform another ECCS analysis.

The ECCS reanalysis and the sensitivity study have been reviewed by the Authority's Plant Operating Review Committee and Safety Review Committee. The Safety Committees have determined that these results (a) do not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; (b) do not increase the probability for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; (c) do not reduce the margin of safety as defined in the basis for any Technical Specification; and (d) do not constitute an unreviewed safety question pursuant to 10 CFR §50.59.

Very truly yours,

P J Early

Paul J. Early
Assistant Chief Engineer-Projects

Att.

Subscribed and sworn to before me this 28 day of January 1980.

Ruth G. Zapf
Notary Public

RUTH G. ZAPF
Notary Public, State of New York
No. 30-4663428
Qualified in Nassau County
Commission Expires March 30, 1980

cc: See attached

cc: Hon. George V. Begany
White Plains Public Library

Mr. T. Rebelowski, Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

ATTACHMENT

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 3.

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

BREAK TYPE - Double Ended Cold Leg Guillotine

BREAK DISCHARGE COEFFICIENT - 1.0

WESTINGHOUSE ECCS EVALUATION MODEL VERSION - February, 1978

CORE PEAKING FACTOR - 2.17

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2094. $^{O}F = PCT_B$

ELEVATION - 6.25 Feet

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2003. $^{O}F = PCT_N$

ELEVATION - 7.25 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION - 2.38 Percent

MAXIMUM CLAD STRAIN AT THIS ELEVATION - 10.0 Percent

Maximum temperature for this 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 - No Burst

HOT ASSEMBLY BLOCKAGE CALCULATED - 0.0 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 (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.^{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 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:

$$\Delta PCT_1 = (.015 + .03)\ (150^{o}F/.01) = 675^{o}F$$

Margin to the $2200^{o}F$ limit is:

$$\Delta PCT_2 = 2200.^{o}F - PCT_B = 106.^{o}F$$

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

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

$$= (675 - 106)\ \left(\frac{.01}{150}\right)$$

$$= .038 \text{ (but not less than zero).}$$

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 $+20.^{o}F$ per percent decrease in strain at a 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:

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

$$= \left(\frac{20}{.01}\right)(.100 - .0238)$$

$$= 152^{\circ}F$$

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,

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

$$= 1.25\ (50-0) + 2.36\ (75-50)$$

$$+121^{\circ}F$$

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 increased for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 152 + 0 = 152.^{\circ}F$$

Margin to 2200°F limit is

$$\Delta PCT_6 = 2200^{\circ}F - PCT_N = 197.^{\circ}F$$

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

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

$$\Delta FQ_N = -0.045, \text{ but not less than zero.}$$

$$=0$$

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

or: $\Delta FQ_{PENALTY} = 0.038$

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).

$$FQ \text{ ADJUSTMENT} = 0.20 - 0.038$$

$$= 0$$

REFERENCE: NS-TMA-2174,
Mr. T. M. Anderson (Westinghouse) to
Mr. D. Eisenhut (NRC), dated December 7, 1979