William J. Cahill, Jr.
Vice President

CENTRAL FILES

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

May 8, 1980

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

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pa 19406

Dear Mr. Grier:

Our response to IE Bulletin No. 80-04 is provided in the Attachment to this letter.

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

Very truly yours,

William J. Cahill, Jr.
Vice President

attach.

cc: Office of Inspection and Enforcement
Division of Reactor Operations Inspection
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

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ATTACHMENT

Item (1)

Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

Response (1)

In response to item 1, an evaluation was made to compare the expected Indian Point Unit 2 containment pressure response to a steam line break assuming auxiliary feedwater runout flow (or runout protection failure) with that recently calculated for a similar Westinghouse four loop plant ("reference plant"). This recent analysis for the reference plant included a detailed containment pressure response calculation for a steam line break with auxiliary feedwater runout protection failure.

Comparison of the expected blowdown transient in the containment for the runout protection failure for Indian Point Unit 2 with the blowdown transient for the reference plant showed that the expected transient for Indian Point Unit 2 is very similar to that of the reference plant. Therefore, the reference plant blowdown assuming failure of the auxiliary feedwater runout protection system is representative of that expected for Indian Point Unit 2.

A review of the containment parameters shows the containment net free volume of the reference plant and Indian Point Unit 2 to be the same. The containment heat sink and heat removal capabilities are similar, thus, the expected containment response to the steamline break should be similar for both units.

A review of the reference plant analysis results shows that the steamline break case which considers a failure of the auxiliary feedwater runout protection system yielded a maximum containment pressure which was ~ 5 psi less than the maximum containment pressure for the limiting case and would not result in containment overpressurization. This applies over the entire spectrum of breaks. Therefore, it is expected that auxiliary feedwater runout flow would not result in the potential for containment overpressurization following a main steam line break at Indian Point Unit 2.

A main steam line break (or main feed line break) inside containment will result in actuation of the engineered safeguards system and will cause automatic isolation of all main steam lines and the main feedwater and condensate system. Indian Point Unit 2 emergency procedures require verification of the actions and manual initiation, if required. In addition, the emergency procedures require identification of the affected steam generator, using steam line pressure instrumentation, and isolation of auxiliary feedwater to the affected steam generator.

Each motor driven auxiliary feed pump is provided with a discharge pressure sustaining control system to prevent the pump from "running out" on its curve. Runout flow conditions on the auxiliary feedwater pumps are also precluded by procedural requirements to maintain the auxiliary feedwater flow regulating valves in a throttled position. Should failure of the runout protection system result in the inoperability of the motor driven AFW pump feeding the damaged steam generator, the other motor driven AFW pump feeding intact steam generators and the steam driven AFW pump will remain operable and be available for maintaining the plant in a safe shutdown condition following the transient.

Item (2)

Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory of the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system.

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power.

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

Response (2)

In response to item 2, a review of the reactivity analysis following a main steamline break for Indian Point Unit 2 was undertaken. Core transient analyses are based upon the following assumptions:

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. Full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater flow is completely terminated following feedwater isolation.

Based on the above manner by which the analysis is performed for Westinghouse plants, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal. Greater feedwater flows during the large steamline breaks serve to reduce secondary pressures and accelerate the automatic safeguards actions, i.e. steamline isolation, feedwater isolation and safety injection. The assumptions described above are therefore appropriate and conservative for the short-term aspect of the steamline break transient.

The limiting portion of the transient occurs during the first minute due to higher steam flow inherently present early in the transient. The auxiliary feedwater flow does not become a dominant factor in determining the duration and magnitude of the transient until the later stages of the transient when the core response has already been terminated due to the introduction of boron to the core via the safety injection system.

In conclusion, based on the evaluation of the effect of runout auxiliary feedwater flow on the core transient for steamline break, it has been determined that the assumptions presently made are appropriate and adequate. the concerns outlined in the introduction of IE Bulletin 80-04 relative to, (1) limiting core conditions occurring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown, and (2) incomplete isolation of main feedwater flow, are not representative of the Westinghouse NSSS designs and associated Balance of Plant requirements including Indian Point Unit 2.

Item (3)

If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed.

Response (3)

Auxiliary feedwater runout flow during a main steamline break would not result in the potential for containment overpressurization for Indian Point Unit 2, as discussed in response to item 1. The impact of auxiliary feedwater runout flow on the core transient is minimal for Indian Point Unit 2 as discussed in response to item 2. Thus, no corrective actions are necessary to prevent containment overpressurization or reactivity increases greater than previously analyzed during the core transient following a steamline break with auxiliary feedwater runout flow.