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Vice President

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October 25, 1976

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

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Consolidated Edison was requested by your letter of August 11, 1976 to evaluate system designs to determine susceptibility to overpressurization events, analyze the possible events, and propose interim plus permanent modifications to systems and procedures to reduce the likelihood and consequences of such events. On September 3, 1976, we transmitted a letter stating that a task group of utilities with Westinghouse plants had been formed to evaluate this problem and that we would review our operating procedures to minimize the likelihood of overpressurization events. We also stated that at the end of the 60-day period addressed in your letter of August 11, a report of progress in this evaluation and review would be provided. This letter provides that report.

A meeting was held by the utility group on September 23, 1976, to review and discuss actions performed by the utilities and Westinghouse. The following presents major items reviewed and the conclusions of that meeting:

A. The overpressurization events which have occurred on Westinghouse designed plants were discussed and the cause of each of the events was noted. In addition, effectiveness of assumed mitigating systems, such as relief valves, was considered. The review of these occurrences indicated that single equipment failure or operator error caused each event and some form of pressure relief would reduce the consequences of such events.

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B. The grouping of various plants was considered as a means to reduce the amount of analysis necessary to evaluate the effectiveness of using the pressurizer power operated relief valves. The review of parameters which would affect analysis results indicated that plant groupings was not necessary because Westinghouse plants are sufficiently similar to envelope the plants by use of a bounding analysis.

C. The pressurizer power operated relief valves were found to have significant water relief capability and relatively quick opening valves, i.e., approximately 3-second (max) opening time.

D. The preliminary evaluation of mass injection induced transients from all possible dynamic sources indicates that the pressurizer power operated relief valves are of the proper mass flow characteristics to limit the pressure surges of such events.

E. The preliminary evaluation of component temperature difference induced transients following a reactor coolant pump start indicates that the pressurizer power operated relief valves may be capable of mitigating the pressure surge of such an event. The equipment temperature difference in conjunction with a pump start induced pressure transient will require a detailed transient analysis to assure acceptability of the system modification selected.

F. Due to the preliminary indications that the pressurizer power operated relief valves may be capable of providing overpressure protection during solid system operation, the overpressurization transients will be analyzed assuming a mitigating system employing the pressurizer power operated relief valves.

G. A working subcommittee was formed to evaluate the possible overpressurization events for the purpose of defining the conditions and parameters to be included in the transient analysis. This subcommittee was directed to meet with Westinghouse and reach concurrence as to the number of events and conditions for each event to be analyzed. This meeting was held on September 28, 1976.

H. A second working subcommittee was also formed to review operating parameters such as chemistry requirements and temperature difference limits which affect implementing procedures to minimize solid system operation by use of a steam bubble at low reactor system average temperatures. This second working subcommittee also was directed to consider an action plan should Appendix G limits be exceeded. This committee met on September 29, 1976.

The meetings which have been held by the utilities and the working subcommittees have resulted in agreement as to a decisive course of action to resolve this overpressurization problem. This course of action includes transient analysis, which will consider mass input induced overpressurization and heat input induced overpressurization. The range of system and component physical parameters, performance characteristics and operating limits applicable to Westinghouse designed plants will be used to bound the analysis. Conservative assumptions will be employed to characterize the relief valve performance.

The single failure criteria presented in your letter of August 11, 1976, will be applied. That is, if the overpressurization transient is caused by an equipment failure or operator error, that failure or error will be considered the single failure event, and all resulting subsequent actions resulting from the failure or error which could reduce the effectiveness of the mitigating system will be considered and included in the analysis.

The preliminary evaluations performed indicate that the use of the pressurizer relief system for protection of a solid reactor coolant system from an inadvertently opened safety injection accumulator, which is charged to its design pressure, is not practical. It is our opinion that administrative controls similar to the controls employed for assuring that the accumulator valves are open during normal operation, which satisfy the single failure critiera, will provide adequate protection against overpressurization during solid system operation. We are revising our procedures to incorporate such controls and are continuing to investigate the feasibility of automatic systems to supplement these controls.

It should be noted that with the administrative controls and the necessary design modifications for pressure relief, the overpressurization of the reactor coolant system will either be avoided, or else the consequences of such an event will be greatly mitigated. However. in the unlikely event that the 10CFR50 Appendix G limits should be exceeded, an analysis would be performed to determine the long term consequences of the event and the impact on plant safety. The analysis would be similar to the analysis presented by Virginia Electric Power Company in Abnormal Occurrence Report AO-S1-73-01-10, dated February 13. 1973.

The analysis of the overpressurization transients will be a major activity with an estimated duration of six months. Following completion of the analysis appropriate modification to the Indian Point Unit No. 2 plant will be initiated consistent with analysis results. Following completion of the analysis, the results of the analysis and the schedule for any required modifications will be provided.

We have reviewed our operating procedures to determine if changes to the procedures are necessary to conform to the objective of limiting solid system operation to the minimum time practical and to increase the awareness of the operators to plant conditions that have the potential of initiating overpressurization transients. Attachment 1 includes a description of the review performed and the instrumentation installed at the Indian Point Unit No. 2 plant to provide a record of both pressure and temperature in the event of an overpressurization transient.

Very truly yours,

William J. Cahill, Jr.

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Vice President

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ATTACHMENT 1

SUMMARY OF CHANGES TO THE INDIAN POINT UNIT NO. 2 PLANT OPERATING PROCEDURES IN ORDER TO PRECLUDE REACTOR COOLANT SYSTEM (RCS) OVERPRESSURIZATION

The relevant procedures for the Indian Point Unit No. 2 plant were informally submitted to the Commission (NRC Project Manager, Peter Erickson). The salient sections of these procedures, which will reduce the likelihood of overpressurization of the Reactor Coolant System because of operator error, are summarized below:

"Low Pressurizer Operation Without a Steam Bubble"

"While the RCS is solid, extreme caution shall be exercised during evolutions since changing RCS temperature or charging and letdown flows could result in an uncontrolled RCS pressure spike."

"The charging and letdown sections of the Chemical and Volume Control System (CVCS) are aligned."

"The Residual Heat Removal System (RHRS) is operational as required...."

(Isolation of the RCS could result in overpressurization while charging to a water solid RCS; the CVCS and/or RHRS provide(s) a letdown path for the RCS, and this mode of operation will preclude an overpressurization event.)

"When the RCS has been placed in the cold shutdown condition, or is operated in the solid mode, automatic SI (Safety Injection) should be defeated by removing the lead on terminal 2 of the following relays."

"While the Reactor Coolant is solid, extreme caution shall be used when changing any parameter that could result in an uncontrolled RCS pressure spike. The following is a list of parameters and how they effect RCS pressure when changed:

A) Increased charging flow will increase RCS pressure.

B) Decreased charging flow will decrease RCS pressure.

C) Increased letdown flow will decrease RCS pressure.

D) Decreased letdown flow will increase RCS pressure.

E) Increased feedwater addition may decrease RCS pressure.

F) Increased component cooling flow while the RHR System is in service will decrease RCS pressure.

G) Increasing RCS temperature will result in an RCS pressure increase.

H) Starting an RHR Pump will cause a decrease in RCS pressure.

I) Stopping an RHR Pump will cause an increase in RCS pressure."

"Adjust RCS pressure, as necessary to remain within the allowable area of the appropriate pressure-temperature curve (refer to Graph RCS-1A or 1B) by manually adjusting charging pump speed and/or letdown flow. Letdown flow should be adjusted by changing the setpoint of the low pressure letdown backpressure controller (PCV-135) as required. When the RHR system is in service the RHR purification path letdown valve (HCV-133) should be sufficiently open so that PCV-135 is the controlling pressure regulating valve. Do not increase RCS pressure above 450 psig if the RHR loop is in service or decrease below 400 psig if an RCP is operating."

"All three letdown orifice isolation valves should remain open so that an unanticipated RCS pressure increase can be relieved through the orifices."

"To preclude Reactor Coolant System (RCS) pressure spikes which exceed the RCS pressure-temperature relationship requirements of the Technical Specifications (refer to Graph RCS-1A or 1B) do not start a RCP, if no other RCPs are running, unless one of the following conditions is met:

a) A gas bubble exists in the RCS. The bubble may be either air, nitrogen or steam. If the bubble is in the pressurizer (nitrogen or steam), the pressurizer level must be less than 95% of span. If the bubble exists in the steam generators due to filling drained loops (air), the pressurizer should be completely filled.

NOTE: Indicated pressurizer level <u>must</u> be corrected to take into account temperature as per Graph RCS-3B and must be at least 5% less than that indicated for a solid pressurizer.

OR

b) There is complete temperature equalization between the water in the Reactor Vessel and the water in the Steam Generators. A comparison between core thermocouples and steam generator shell thermocouples should be made to verify this."

"Reactor Coolant Pump Operation"

This procedure states that a RC pump can not be started unless one of the following is met:

A. A gas bubble exists in the RCS

B. There is complete temperature equalization between the water in the RCS and in the steam generators

"Plant Heatup from Cold Shutdown Condition"

This procedure states that the RC pumps must be started, operated and shutdown in accordance with procedure SOP-1.3 ("Reactor Coolant Pump Operation").

This procedure has the following "caution" - "When the RCS is solid and the RHR pumps are shutdown, a pressure spike will occur due to the decreased letdown flow through HCV-133. When the pump is tripped, HCV-133 will have to be opened further to re-establish the letdown flow to the previous value."

"Plant Cooldown from Zero Power Condition to Cold Shutdown Condition"

This procedure states that when the pressurizer pressure decreases below 1000 psig. energize and close the accumulator isolation valves; when the accumulator isolation valves are closed, de-energize their motor operator by opening their respective disconnect switches. (Thus the accumulators will be locked-out by double administrative controls, and pressure transients due to accumulator fluid flowing into the RCS is precluded.)

This procedure states that when the RCS pressure decreases below 450 psig and temperature decreases below 350 F place the RHR system in service.

This procedure states "Open all three letdown orifices 200A, 200B and 200C if not previously done."

This procedure states that when the RCS temperature is below 200^{o}F, defeat automatic safeguards actuation (thus a pressure transient due to safeguards fluid flowing into the RCS is precluded below a temperature of 200^{o}F).

"Filling and Venting the Reactor Coolant System"

"Align Chemical Volume Control System"

(Isolation of the RCS could result in overpressurization while charging to a water solid RCS; the Chemical Volume Control System provides a letdown path for the RCS, and this mode of operation will preclude an overpressurization event.)

This procedure establishes the RHR letdown and purification path (and as explained in the preceding paragraph a letdown mode of operation will preclude an overpressurization event.

This procedure has the following "Caution" - a) stay within the limits of the heatup curve, and b) starting or stopping an RHR pump will change the head on the letdown path by 150 psig which will affect the Reactor Coolant System pressure greatly when solid.

"Collapsing the Steam Bubble in the Pressurizer"

This procedure describes the method for collapsing the pressurizer steam bubble during RCS cooldown. It specifies a maximum allowable Δ T between the RCS and the pressurizer of 320^{o}F (to prevent thermal shocking of the surge line thermal sleeve), and a maximum allowable pressurizer cooldown rate of 200^{o}F per hour.

For Indian Point Unit No. 2 the following instrumentation allows us to verify adherence to the temperature/pressure limitations in our Technical Specification.

1. Resistance temperature detectors (RTDs) are installed in each loop cold leg. These RTDs provide wide range temperature indication during heatup and cooldown. The four temperatures are indicated on two continuous, two pen, recorders located on the control room flight panel (Loops 21 and 22 on one recorder and Loops 23 and 24 on the other).

2) A pressure transmitter is installed on the hot leg of No. 24 loops. The transmitter provides continuous full range (0-3000psig) pressure indication on a flight panel recorder in the control room.

   An additional pressure transmitter will be installed on the hot leg of No. 21 loop, and a second recorder will be installed to provide narrow range (0-750psig) pressure indication on the supervisory panel in the control room. Our Engineering Department is presently designing for the installation of a large flashing red light and an audible device which has a sound that will be readily distinguishable from existing audible alarms. This will provide the capability to warn the plant operator of an approaching overpressurization condition.