John D. O'Toole
Vice President

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December 20, 1985

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

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

ATTN: Mr. Hugh L. Thompson, Jr., Director
Division of PWR Licensing-A

Dear Mr. Thompson:

This letter provides our response to Generic Letter 85-12 pertaining to the implementation of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps." Attachment A provides the plant specific information requested in Section IV of the Safety Evaluation Report issued with Generic Letter 85-12.

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

Very truly yours,

for J. O'Toole

attach.

cc: Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

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ADD.

AD - J. KNIGHT (ltr Only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
RSB (BERLINGER)
FOB (BENAROYA)

ATTACHMENT A

INDIAN POINT UNIT NO. 2

RESPONSE TO GENERIC LETTER 85-12
TMI ACTION ITEM II.K.3.5
"AUTOMATIC TRIP OF REACTOR COOLANT PUMP"

A. Determination of RCP Trip Criteria

1. Identify the instrumentation to be used to determine the RCP trip set point, including the degree of redundancy of each parameter signal needed for the criterion chosen.

Response

As indicated in our August 1, 1984 letter, Consolidated Edison Company of New York, Inc. (Con Edison) is currently using a symptom based low Reactor Coolant System (RCS) pressure Reactor Coolant Pump (RCP) manual trip criterion. This criterion is in accordance with the Westinghouse Owners Group (WOG) justification for manual RCP trip for small break LOCA events submitted to NRC on March 9, 1984 and the evaluation of alternate RCP trip criteria provided in the WOG developed Emergency Response Guidelines (ERG's), Revision 1 (LP).

At Indian Point Unit No. 2 (IP-2) the RCS pressure is sensed and transmitted by two independent and environmentally qualified wide range (0-3000 psig) pressure transmitters designated PT-402 and PT-403 (one of the two associated recorders does not record full range and will be replaced to meet the required range as part of our ongoing NUREG-0737 Supplement 1 -Regulatory Guide 1.97 upgrade program as discussed in our letter of August 30, 1985).

2. Identify the instrument uncertainties for both normal and adverse containment conditions. Describe the basis for the selection of the adverse containment parameters. Address, as appropriate, local conditions such as fluid jets or pipe whip which might influence the instrumentation reliability.

Response

The associated instrument uncertainty for both the normal and adverse containment condition is as follows:

Normal uncertainty = 3% of span (3000 psi) or 90 psi

Adverse uncertainty = 13% of span (3000 psi) or 390 psi
(Includes normal uncertainty)

The selection of adverse containment parameters is based on the calculated values used to support our plant specific implementation of Rev. 1 of the WOG ERGs and considers the effects of instrument accuracy and adverse containment conditions. The methodology employed, to select the appropriate RCP setpoint for adverse containment conditions, is contained in the Executive Volume and background volumes of the WOG ERGs Rev. 1. In the IP-2 Emergency Operating Procedures (EOPs) both the normal and adverse containment RCP trip parameter values are given. The operator decides which of the two values to use by observing the containment pressure and radiation values for determining when to apply adverse containment values.

If either containment pressure exceeds 4 psig or the containment radiation exceeds 10^5 R/hr, the operator would implement the procedures using the adverse containment (post-accident) RCP trip parameter. Alternatively, if containment pressure is less than 4 psig and containment radiation is less than 10^5 R/hr, the operator would use the normal containment value. The RCS wide range pressure transmitters PT 402 and PT 403 are located inside of containment but outside of the crane wall and are not influenced by jet impingement and pipe whip.

3. In addressing the selection of the criterion, consideration to uncertainties associated with the WOG supplied analyses values must be provided. These uncertainties include both uncertainties in the computer program results and uncertainties resulting from plant specific features not representative of the generic data group.

   If a licensee determines that the WOG alternative criteria are marginal for preventing unneeded RCP trip, it is recommended that a more discriminating plant-specific procedure be developed. For example, use of the NRC-required inadequate-core-cooling instrumentation may be useful to indicate the need for RCP trip. Licensees should take credit for all equipment (instrumentation) available to the operators for which the licensee has suffficient confidence that it will be operable during the expected conditions

Response

The Westinghouse LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA events were simulated in these analyses. LOFTRAN is a Westinghouse licensed code used for FSAR SGTR and non-LOCA analyses. The code has been validated against the January 1982 SGTR event at the Ginna plant. The results of this validation show that LOFTRAN can accurately predict RCS pressure, RCS temperature and secondary pressures, especially in the first ten minutes of the transient. This is the critical time period when minimum pressure and subcooling are determined.

The major causes of uncertainties and conservatism in the computer program results, assuming no changes in the initial plant conditions(i.e. full power, pressurizer level, all SI and AFW pumps run), are due to either models or inputs to LOFTRAN. The following are considered to have the most impact on the determination of the RCP trip criteria:

1. Break flow
2. SI flow
3. Decay heat
4. Auxiliary feedwater flow

The following sections provide a generic assessment of the uncertainties associated with each of these items and a plant-specific impact.

To conservatively simulate a double ended tube rupture in the safety analyses, the break flow model used in LOFTRAN includes a substantial amount of conservatism (i.e., predicts higher break flow than actually expected). Westinghouse has performed analyses and developed a more realistic break flow model that has been validated against the Ginna SGTR tube rupture data. The break flow model used in the WOG safety analyses has been shown to be approximately 30% conservative when compared to the more realistic model. The consequence of the higher predicted break flow is a lower than predicted minimum RCS pressure.

The SI flow inputs used were derived from best estimate calculations, assuming all SI trains operating. An evaluation of the calculational methodology shows that these inputs have a maximum uncertainty of ± 10%.

The decay heat model used in the WOG analyses was based on the 1971 ANS 5.1 standard. When compared with the more recent 1979 ANS 5.1 decay heat inputs, the values used in the WOG safety analyses are higher by about 5%.

To determine the effect of the uncertainty due to the decay heat model, a sensitivity study was conducted for SGTR. The results of this study show that a 20% decrease in decay heat resulted in only a 1% decrease in RCS pressure for the first 10 minutes of the transient. Since RCS temperature is controlled by the steam dump, it is not affected by the decay heat model uncertainty.

The AFW flow rate input used in the WOG safety analyses are best estimate values, assuming that all auxiliary feed pumps are running, minimum pump start delay, and no throttling. To evaluate the uncertainties with AFW flow rate, a sensitivity study was performed. Results from the two loop plant show that a 64% increase in AFW flow resulted in only an 8% decrease in minimum RCS pressure, a 3% decrease in minimum subcooling, and a 8% decrease in minimum pressure differential. Results from the 3 loop plant study show that a 27% increase in AFW flow resulted in only a 3% decrease in minimum RCS pressure, a 2% decrease in minimum RCS subcooling, and a 2% decrease in pressure differential.

The effects of all these uncertainties with the models and input parameters were evaluated and it was concluded that the contributions from the break flow conservatism and the SI uncertainty dominate. The calculated overall uncertainty in the WOG analyses as a result of these considerations for IP-2 is +20 to +100 psig for the minimum RCS pressure RCP trip setpoint. Due to the minimal effects from the decay heat model and AFW input, these results include only the effects of the uncertainties due to the break flow model and SI flow inputs.

As part of the WOG SGTR safety analyses, a minimum RCS pressure of 1190 psia was calculated for IP-2 using the LOFTRAN model without the improved break flow model. Applying the results of the Westinghouse uncertainty assessment described above, the expected minimum RCS pressure could be approximately 100 psi higher. Internal Con Edison calculations confirm this and, in fact, calculate a minimum RCS pressure of 1315 psia with an improved break flow model.

For normal containment conditions, as would be expected for a design basis SGTR, our EOP manual RCP trip setpoint is 1250 psig. When comparing this value with the expected minimum value of 1300 psig (1315 psia), the RCPs are not expected to be tripped using the low RCS pressure criterion at IP-2. Since all three RCP trip criteria (low RCS pressure, RCS subcooling, and RCS/Secondary differential pressure) are valid for IP-2, low RCS pressure is simplest of the three to use from a human factors engineering standpoint, and since RCS pressure is a common input to determine RCS subcooling and RCS/Secondary differential pressure, we have maintained useage of the low RCS pressure criterion for manual RCP trip at IP-2.

B. Potential Reactor Coolant Pump Problems

1. Assure that containment isolation, including inadvertent isolation, will not cause problems if it occurs for non-LOCA transients and accidents.

   a. Demonstrate that, if water services needed for RCP operations are terminated, they can be restored fast enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

   b. Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

Response

As indicated in our April 15, 1983 letter, the RCPs at IP-2 are provided with two means of seal cooling: (1) direct seal water injection via the Chemical and Volume Control System (CVCS) and (2) thermal barrier cooling via the Component Cooling System (CCS). Both of these flow paths are identified as being "essential".

In general, containment isolation at IP-2 initiates on either a Phase A or Phase B isolation signal. A Phase A isolation signal isolates only non-essential process lines; thus, water services needed to prevent RCP seal damage are not terminated. A Phase B isolation signal trips the automatic isolation valves in the "essential" process lines penetrating the containment. Accordingly, the contaiment isolation valves in the CCS lines for RCP thermal barrier and motor lube oil cooling and the CVCS seal return line isolation valve receive a Phase B automatic isolation signal. The CVCS seal injection supply line valves are open, remote manual, thus this line would remain in service. A Phase B automatic containment isolation signal is only generated if the containment internal pressure reaches the high-high setpoint. (i.e., Tech Spec limit 30 psig, actually

set around 28 psig). Thus, large breaks in either the main steam system inside containment or in the reactor coolant system are the only credible design basis accidents that could, by conservative FSAR analysis, create containment pressures in that range.

Accordingly, except for the above mentioned two specific and unlikely accident initiators, contaiment isolation signals should not result in the termination of the cooling systems essential to RCP operation, especially for those hypothetical accidents where RCP operation is desirable

Nonetheless, if Phase B isolation occurs during RCP operation, the IP-2 EOPs permit the operator up to two minutes in the absence of all RCP seal cooling before RCP trip is required. This time limit is intended to preclude subsequent seal or pump damage or failure. Since CVCS seal injection lines remain in service, seal damage is even more remote. In addition, two minutes provides the required time to permit the operator to reset the Phase B isolation signal and reestablish normal RCP services if continued RCP operation is necessary or desirable and feasible.

In the event of prolonged loss of seal cooling with the RCPs tripped (e.g., station blackout), the EOPs provide the steps required to reestablish component cooling and seal injection for the RCPs in a controlled manner to prevent thermal shocking of the RCPs and subsequent potential pump shaft bowing and/or seal damage.

2. Identify the components required to trip the RCPs, including relays, power supplies and breakers. Assure that RCP trip, when determined to be necessary, will occur. If necessary, as a result of the location of any critical component, include the effects of adverse containment conditions on RCP trip reliability. Describe the basis for the adverse containment parameters selected.

Response

Each RCP is manually tripped by moving its associated breaker control switch to the trip position. Each switch is mounted on the Supervisory Panel SA located in the central control room. In the trip position, the breaker control switch will energize a RCP breaker trip coil (powered by a 125VDC power source) located in the switchgear assembly and causing the RCP 6.9Kv circuit breakers to open thus removing power from the RCP motors.

All of the necessary components to trip the RCPs are located outside containment and, thus, are not influenced by adverse containment conditions.

C. Operator Training and Procedures (RCP Trip)

1. Describe the operator training program for RCP trip. Include the general philosophy regarding the need to trip pumps versus the desire to keep pumps running.

Response

The IP-2 reactor operator/senior reactor operator training program and requalification program were recently revised to incorporate the procedure upgrade (EOPs) required by NUREG-0737, Supplement 1. The RCP trip is now an integral part of many upgraded EOPs (refer to response to Item C.2) and the upgraded EOP training program. In general the IP-2 EOP training program consisted of classroom instruction, directed self-study, and simulator exercises. The classroom lectures are the principal mechanism to accomplish operator EOP training. A description of the IP-2 EOP training program was provided in our letter dated June 4, 1984 in response to NUREG-0737, Supplement 1. The EOP training program presented the following:

- a discussion of plant conditions (e. g., normal operation, anticipated transients, and accident conditions) where RCP trip or restart may be required and the performance of RCPs under these conditions

- a detailed description of the RCP trip criteria including an evaluation of alternate parameters, selection of the RCP low RCS pressure trip parameter, and the calculation of the low RCS pressure manual trip setpoint.

- a discussion of RCS pressure instrumentation and the various contributors to instrument channel accuracy.

The above RCP trip topics were discussed during the operator EOP training program and are part of the requalification training program. In addition, the general philosophy regarding the need to trip the RCPs versus the desire to keep the RCPs running was discussed during the EOP training program. Basically, this philosophy emphasizes the need to trip the RCPs for small break LOCAs where it is considered necessary, while ensuring continued RCP operation for steam generator tube rupture events (up to the design basis double-ended tube rupture) and other non-LOCA transients where forced circulation is desirable (e.g., steam line breaks equal to or smaller than 1 stuck open secondary side PORV).

2. Identify those procedures which include RCP trip related operations:

a. RCP trip using WOG alternate criteria
b. RCP restart
c. Decay heat removal by natural circulation
d. Primary system void removal
e. Use of steam generators with and without RCPs operating
f. RCP trip for other reasons

Response

The IP-2 Emergency Operating Procedures which contain the RCP trip related instructions are as follows:

a. RCP trip using WOG alternate criteria

E-0	REACTOR TRIP OR SAFETY INJECTION
ES-0.4	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL(WITHOUT RVLIS)
E-1	LOSS OF REACTOR OR SECONDARY COOLANT
E-3	STEAM GENERATOR TUBE RUPTURE
ECA-2.1	UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATOR

b. RCP Restart

ES-0.1	REACTOR TRIP RESPONSE
ES-0.2	NATURAL CIRCULATION COOLDOWN
ES-0.3	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)
ES-0.4	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)
ES-1.2	POST LOCA COOLDOWN AND DEPRESSURIZATION
ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED
ECA-3.2	SGTR WITH LOSS OF REACTOR COOLANT-SATURATED RECOVERY DESIRED

FR-C.1	RESPONSE TO INADEQUATE CORE COOLING
FR-I.3	RESPONSE TO VOIDS IN REACTOR VESSEL

c. Decay Heat Removal by Natural Circulation

E-0	REACTOR TRIP OR SAFETY INJECTION
ES-0.2	NATURAL CIRCULATION COOLDOWN
ES-0.3	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)
ES-0.4	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)

d. Primary System Void Removal

E-0	REACTOR TRIP OR SAFETY INJECTION
ES-0.3	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)
ES-0.4	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)
FR-I.3	RESPONSE TO VOIDS IN REACTOR VESSEL

e. Use of Steam Generators With and Without RCPs Operating

E-0	REACTOR TRIP OR SAFETY INJECTION
ES-0.2	NATURAL CIRCULATION COOLDOWN
ES-0.3	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)
ES-0.4	NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)
ECA-0.0	LOSS OF ALL AC POWER

f. RCP Trip for Other Reasons

FR-C.2 RESPONSE TO DEGRADED CORE COOLING

FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK