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John C. Brons
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December 13, 1985
IPN-85-63

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

Attention: Mr. Hugh L. Thompson, Director
Division of Licensing

Subject: Indian Point Unit 3
Docket No. 50-286
Plant-Specific Information Requested by Generic Letter No. 85-12, "Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pump." (TAC# M49687)

Dear Sir:

The purpose of this letter is to inform you of the reactor coolant pump (RCP) manual trip criteria selected by the Authority and to provide the necessary plant-specific information to justify its implementation at Indian Point 3.

The Safety Evaluation Report (SER) accompanying the subject generic letter (GL 85-12) approved the Westinghouse Owners Group (WOG) methodology for justifying manual RCP trip in lieu of automatic trip. The three alternative trip criteria employed by WOG were found consistent with the original RCP trip guidelines of Generic Letters 83-10c and d. The Authority endorses the WOG methodology. We have selected Reactor Coolant Subcooling as the alternate criteria for determining when to trip RCPs. The appropriate manual trip setpoints will be incorporated as necessary into the plant's Emergency Operating Procedures.

The enclosed attachments provide the plant-specific information in a question and answer format corresponding to the information requests of Section IV of the GL 85-12 SER. This addresses the plant-specific questions not answered by the WOG generic responses to Generic Letter 83-10c and d.

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ADD: AD - J. KNIGHT (ltr only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
PSB (BERLINGER)
FOB (BENAROYA)

We trust this submittal will enable you to complete the plant-specific SER for implementation of TMI Action Item II.K.3.5. Should you or your staff have any questions regarding this response please contact Mr. P. Kokolakis of my staff.

Very truly yours,

J.C. Brons
Senior Vice President
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Attachments

cc: Resident Inspector's Office
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Attachments to IPN-85-63

RESPONSE TO GENERIC LETTER NO. 85-12

IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

ATTACHMENT A

IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5

A. DETERMINATION OF RCP TRIP CRITERIA

1. REQUEST

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

The Reactor Coolant System (RCS) Loop Subcooling Margin is monitored to determine when RCPs should be manually tripped. This saturation margin is calculated by the plant computer using inputs from the hot (T_H) and cold (T_C) leg RTDs of each RCS loop and the Loop 1 and Loop 4 wide range RCS pressure instruments. The program compares the lowest RCS pressure to the saturation pressure corresponding to the highest T_H or T_C RTD signal. The difference in these two parameters is displayed in psi and degrees (F) subcooling by the Emergency Response Facility Data Acquisition and Display System (ERFDADS). In addition to loop subcooling, saturation margin is also calculated using pressurizer pressure instruments, loop pressure instruments and core-exit thermocouples. (Reference 1).

The ERFDADS is comprised of a Qualified Safety Parameter Display System (QSPDS) and a Critical Function Monitoring System (CFMS). There are two (2) QSPDS and four (4) CFMS displays in the Control Room which can provide subcooling values to the operator. QSPDS is a redundant computer system qualified to Class 1E standards, including seismic and single failure proof design. It provides overall plant safety status from a minimum set of variable inputs whose instrumentation requirements are based on Regulatory Guide 1.97 criteria.CFMS is the plant's principle SPDS which in addition to processing the qualified inputs of QSPDS also incorporates an expanded signal list (non-1E) for monitoring the EOP Critical Safety Functions (Reference 2).

Besides the redundant plant computer system described above, RCS loop subcooling is also provided on continuous strip chart readout by the Saturation Recorder as required by plant Technical Specifications. This device monitors the two (2) wide range loop pressure instruments and four (4) T_H RTDs independently of ERFDADS. An alarm occurs when the difference between RCS pressure and calculated saturation pressure approach saturated conditions (i.e. 300 psi and also when RCS is at saturation). (Reference 3).

2. REQUEST

Identify the instrumentation 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 subcooling margin setpoints for manual RCP trip are as follows:

Normal Containment Conditions: 29°F subcooled.

Adverse Containment Conditions ($>10^5$R/hr or >3psig):

RCS Pressure	Degrees Subcooled
>1900 psig	50 °F
1900 > 1000 psig	100 °F
< 1000 psig	225 °F

The above subcooling setpoint values are based on the combined maximum instrument and monitor uncertainties for the range of RCS pressures shown below:

RCS Pressure (psig)	Normal Error °F	Adverse Error °F
450	28.9	not available
585	26.5	225.5
785	24.7	123.9
985	23.7	93.8
1185	23.1	77.6
1385	22.7	67.2
1585	22.4	59.8
1785	22.2	54.4
1985	22.1	50.1
2185	22.0	46.6
2385	21.9	43.8
2485	21.8	42.6
	MAX.VAL.=28.9°F	MAX.VAL. = 225.5°F

The adverse containment paramemters (10^5R/hr and 3psig) were selected as part of a setpoint study conducted for the IP-3 EOPs. They are based upon containment conditions, below which environmental effects on instrument uncertainty are considered negligible when determining overall channel inaccuracies for EOP setpoint instruments. All instruments providing input into the loop subcooling calculation meet environmental requirements of Reg. Guide 1.97 except for Loop 1 pressure which will be rescaled to meet the new range criteria. The redundant QSPDS computer processors which provide these subcooling values to the operators through redundant printers/CRT displays, are designed to Seismic Category 1 and Class 1E criteria. Additionaly, local event conditions such as pipe whip or fluid jets are not expected to affect subcooling output values due to the redundancy of inputs available should one instrument be rendered inoperable.

3. REQUEST

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.

RESPONSE

A study of non-LOCA safety analysis events revealed that a steam generator tube rupture (SGTR) was the most limiting event for the Indian Point 3 plant type in terms of minimum pressures and temperatures approaching trip conditions of small break (SB) LOCA events. WOG employed the LOFTRAN computer code on a 4 loop/low pressure safety injection (SI) plant model, representative of Indian Point 3, to generate the minimum transient values expected for the three alternative RCP trip criteria. Inputs to the LOFTRAN code were conservatively estimated to bound plant specific flow rates and core behavior. A discussion of uncertainties associated with the code and input models follows.

LOFTRAN is a Westinghouse licensed code which has been validated against the Ginna SGTR event of January 1982. The major source of uncertainity in the computer program results are due either to the models or the inputs to the LOFTRAN code, assuming the following initial plant conditions remain unchanged:

- 100% Power
- Best estimate RCS flow
- Best estimate SI flow
- Total AFW flow
- Best estimate decay heat
- Best estimate reactivity coeffecients
- Steam dumps operable.

To bound the uncertainity range in the SGTR analysis results, an evaluation of uncertainties associated with these parameters was conducted by the WOG for all the models and plant types analyzed. Those input models having the most effect on program conservatisms are discussed below:

a. Break Flow: The break flow model used in LOFTRAN is 30% more conservative than realistic break flow calculations validated against the Ginna SGTR. Consequently break flows used in the plant models are much higher than can be expected for double ended tube ruptures for actual plant configuration.

b. SI Flow: The SI flow inputs used represent best estimates based on all SI trains operating. A review of the calculational methodology shows that variations in these inputs for plant types evaluated, have a maximum uncertainty range of -10% to +10%.

c. Decay Heat: The decay heat model used is based on the 1971 ANS 5.1 standard. This results in a 5% higher decay heat value when compared to the more recent 1979 ANS 5.1 standard. A sensitivity study for the SGTR analysis showed that a 20% decrease resulted in only a 1% decrease in RCS pressure values for the first 10 minutes of the transient. RCS temperature is not affected by decay heat uncertainty since it is assumed that steam dumps are available for temperature control.

d. AFW Flow: This input is based on all AFW pumps actuating with minimum start delay and no throttling. Sensitivity studies show that SGTR analysis results are relatively unaffected by changes in AFW flow.

CALCULATIONS:

A range of minimum transient values that bound the uncertainties discussed above are calculated below for the limiting SGTR event. The uncertainty range of each parameter below accounts only for the effects of break flow conservatism and SI flow uncertainty which are the dominant factors in LOFTRAN output. These minimum expected transient ranges are compared below to the RCP trip setpoints for each of the three alternative parameters to determine the ability of each trip criteria to discriminate between small break LOCA and SGTR events.

	RCS Pressure (psig)	Subcooling (°F)	$P_{RCS}-P_{S/G}$ (psi)
Non-LOCA SGTR lower transient limits	1216 to 1296	32 to 42	320 to 325
RCP Tripoint for SB LOCA	1260	29	171

RCS pressure does not meet the condition for distinguishing between a LOCA and a non-LOCA event. Subcooling and ($P_{RCS}-P_{S/G}$) both meet the criteria.

The parameter chosen is loop subcooling margin because the operator only has to look at a single parameter. To use ($P_{RCS} - P_{S/G}$) the operator must look at two parameters. Additionally, subcooling is used as a conditional requirement for initiating steps or actions for other events covered by Emergency Response Guidelines (i.e. SI flow reduction, RCS depressurization for RHR initiation).

B. POTENTIAL REACTOR COOLANT PUMP PROBLEMS

1. REQUEST

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

a. Phase A Containment Isolation: trips the majority of automatic isolation valves coincident with automatic safety injection actuation. Phase A isolation will shut off "non-essential" process lines penetrating containment (i.e. not required for near term accident mitigation). Water services for RCP operation are considered "essential" process lines and are not automatically isolated. Thus, the majority of non-LOCA events (including SGTR) for which RCP operation is desired, continued RCP water services will not be hindered. (Reference 3).

b. Phase B Containment Isolation: trips the automatic isolation valves for the remaining or "essential" process penetration lines coincident with actuation of automatic containment spray. RCP seal water supply (manual isolation only) is not affected, provided one charging pump is running to maintain minimum seal flow and pressure differential. Component Cooling Water (CCW) supply to RCP seal thermal barriers and upper and lower bearing oil reservoirs is cutoff on Phase B isolation in which case RCP operation must be secured within two minutes or when high bearing temperature setpoints are reached (Reference 4). However, RCP operation is not desirable since containment isolation in this case (Phase B) is indicative of a LOCA event where RCPs are secured for accident mitigation and prevention of pump damage. (Reference 3).

When RCPs are stopped either seal injection or CCW supply must be continued. Numerous steps throughout the various procedures in the ERGs remind operators to check seal water or CCW supply to RCPs whenever they might have been affected. If RCP restart is desirable after Phase B isolation has occurred, CCW supply can be restored by remote or manual valve operation. RCP restart steps are included throughout ERG procedures whenever their operation is more desirable during event recovery. (Reference 5).

2. REQUEST

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

RCPs are powered by 75-DH500ACB 1200 amp circuit breakers from 6.9KV switchgears No. 31 & 32 located in the Turbine Building. These circuit breakers are tripped open by solenoids actuated from Control Room Panel SAF. Adverse containment conditions are not expected to affect RCP trip reliability as all of the trip circuitry is located in the Turbine and Control Buildings. (Reference 6).

C. OPERATOR TRAINING AND PROCEDURES (RCP TRIP)

1. REQUEST

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

Training for RCP trip was included in the EOP training program. This included classroom instruction on the logic behind the EOPs, the process used to develop the EOPs and the EOPs themselves along with supporting technical and human factors information. (Reference 7).

The general philosophy regarding the need to trip RCPs is consistent with the underlying concepts used to develop the EOPs. Prior to development of the EOPs, proper response to emergency transients rested heavily on the operators ability to correctly diagnose the cause of the transient. Due to the wide spectrum of potential plant responses to the various event transients which could occur, the EOPs were established to include a symptom-based diagnosis to guide the operator into selecting the optimal recovery strategy (either event-related or function-related) to place the plant in the optimal end state for the particular transient involved. Likewise, the operator is provided with symptom based criteria (i.e. RCS Loop Subcooling) in the EOPs for determining when RCP trip is necessary. The operator is not required to diagnose the exact nature of the emergency transient (i.e. whether or not a SBLOCA is occuring), but need only know that reactor coolant saturation conditions requiring RCP trip are occurring. In this way the probability of tripping RCPs is maximized for a SBLOCA and minimized for non-LOCA events. (Reference 2).

2. REQUEST

Identify those procedures which include RCP trip related operation:

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

Refer to Attachment B for titles to ERG procedure numbers. (Reference 5).

a. RCP trip using Subcooling Margin: E-0, ES-0.4, E-1, E-3 and ECA-2.1.

b. RCP restart: ES-0.1, ES-0.2, ES-0.3, ES-0.4, ES-1.1, ES-1.2, E-3, ECA-2.1, ECA-3.1, ECA-3.2, FR-C.1, FR-P.1, and FR-I.3.

c. Decay heat removal by natural circulation: ES-0.2, ES-0.3, and ES-0.4.

d. Primary system void removal: ES-0.3, ES-0.4, E-3 and FR-I.3.

e. Use of steam generators with and without RCPs operating: E-0, ES-0.1, ES-0.2, ES-0.3, ES-0.4, E-1, ES-1.1, ES-1.2, E-3, ES-3.1, ES-3.2, ES-3.3, ECA-0.0, ECA-1.1, ECA-2.1, ECA-3.1, ECA-3.2, ECA-3.3, FR-C.1, FR-C.2, FR-H.1, FR-H.2, and FR-I.3

f. RCP trip for other reasons:

(1) Phase B isolation required: E-0

(2) Loss of CCW flow to RCP: E-0

(3) RCP Number 1 seal leakoff or seal differential pressure less than minimum required: ES-1.2, ES-3.1, ES-3.2, ES-3.3, ECA-1.1, ECA-3.1, ECA-3.2 and ECA-3.3

(4) Stop all but one RCP to enchance cooldown: ES-1.2, E-3, ECA-3.1 and ECA -3.2

(5) Enhance recovery of Critical Safety Function: FR-C.1, FR-C.2 and FR-H.1

(6) The Standard Operating Procedure for the RCP also states other parameters which, when exceeded, require pump trip (e.g., vibration).

D. REFERENCES

1. Letter IPN-85-19 from Mr. J.P. Bayne (NYPA) to Mr. S.A. Varga (NRC) "Safety Parameter Display System - Safety Analysis Report," April 16, 1985.

2. Westinghouse Owners Group Emergency Response Guidelines (ERGs), "Executive Summary", Revision 1, transmitted to NRC via OG-111 dated 11/30/83.

3. Indian Point 3 FSAR, updated July, 1985.

4. Indian Point 3 Plant Manual Volume VI, "Precautions Limitations, Setpoints," revised by Westinghouse March, 1976.

5. Westinghouse ERGs, Rev. 1, "Optimal Recovery and Function Restoration Guidelines".

6. Westinghouse Elementary Wiring Diagrams - Indian Point 3 #31143 SH. 7 & 11; #31133, SH. 7 & 10.

7. Letter IPN-84-33, from Mr. J.P. Bayne (NYPA) to Mr. S.A. Varga (NRC), "Emergency Operating Procedures Upgrade Program," August 29, 1984.

ATTACHMENT B

EMERGENCY RESPONSE GUIDELINES
OPTIMAL RECOVERY GUIDELINES

E-0: Reactor Trip Or Safety Injection
- 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)

E-1: Loss Of Reactor Or Secondary Coolant
- ES-1.1: SI Termination
- ES-1.2: Post LOCA Cooldown and Depressurization
- ES-1.3: Transfer To Cold Leg Recirculation
- ES-1.4: Transfer To Hot Leg Recirculation

E-2: Faulted Steam Generator Isolation

E-3: Steam Generator Tube Rupture
- ES-3.1: Post-SGTR Cooldown Using Backfill
- ES-3.2: Post-SGTR Cooldown Using Blowdown
- ES-3.3: Post-SGTR Cooldown Using Steam Dump

ECA-0.0: Loss Of All AC Power
ECA-0.1: Loss Of All AC Power Recovery Without SI Required
ECA-0.2: Loss Of All AC Power Recovery With SI Required

ECA-1.1: Loss Of Emergency Coolant Recirculation
ECA-1.2: LOCA Outside Containment

ECA-2.1: Uncontrolled Depressurization Of All Steam Generators

ECA-3.1: SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired
ECA-3.2: SGTR With Loss Of Reactor Coolant-Saturated Recovery Desired
ECA-3.3: SGTR Without Pressurizer Pressure Control

EMERGENCY RESPONSE GUIDELINES

FUNCTION RESTORATION GUIDELINES

FR-S.1:	Response to Nuclear Power Generation/ATWS
FR-S.2:	Response to Loss of Core Shutdown
FR-C.1:	Response to Inadequate Core Cooling
FR-C.2:	Response to Degraded Core Cooling
FR-C.3:	Response to Saturated Core Cooling
FR-H.1:	Response to Loss of Secondary Heat Sink
FR-H.2:	Response to Steam Generator Overpressure
FR-H.3:	Response to Steam Generator High Level
FR-H.4:	Response to Loss of Normal Steam Release Capabilities
FR-H.5:	Response to Steam Generator Low Level
FR-P.1:	Response to Imminent Pressurized Thermal Shock Condition
FR-P.2:	Response to Anticipated Pressurized Thermal Shock Condition
FR-Z.1:	Response to High Containment Pressure
FR-Z.2:	Response to Containment Flooding
FR-Z.3:	Response to High Containment Radiation Level
FR-I.1:	Response to High Pressurizer Level
FR-I.2:	Response to Low Pressurizer Level
FR-I.3:	Response to Voids in Reactor Vessel