T/C

Portland General Electric Company

Charles Goodwin, Jr. Assistant Vice President

August 29, 1979

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Mr. R. H. Engelken, Director
Nuclear Regulatory Commission
Region V
Suite 202, Walnut Creek Plaza
1990 N. California Blvd.
Walnut Creek, CA 94596

Dear Sir:

IE Bulletin 79-06C, dated July 26, 1979, was transmitted to Portland General Electric Company (PGE) for action concerning operation of reactor coolant pumps (RCPs) after a LOCA. PGE has reviewed the content of this Bulletin and evaluated its schedules and implications on the Trojan Nuclear Plant.

Attached is PGE's response to the subject Bulletin which has been prepared in cooperation with the Westinghouse Owner's Group. As we indicated in our July 31, 1979 letter relating to IE Bulletin 79-06C, this response satisfies both IE Bulletins 79-06A (Items 4 and 7.c) and 79-06C.

Sincerely,

C. Goodwin

C. Goodwin, Jr.
Assistant Vice President
Thermal Plant Operation and
Maintenance

CG/KM/4sb4A3
Attachment

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation

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PGE RESPONSE TO IE BULLETIN 79-06C

Short-Term Action

1. In the interim, until the design change required by the long-term action of this Bulletin have been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action, other immediate and follow-up actions required during such an occurrence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

   PGE Response

   A. Immediate Operator Actions in the existing Trojan Emergency Instruction EI-1, "Loss of Reactor Coolant" (copy attached) requires the operator to stop all reactor coolant pumps immediately upon verification of (1) reactor trip, (2) turbine trip, and (3) initiation of safety injection (including initiation of HPI by low reactor coolant system pressure). Therefore, this item is already being implemented at Trojan.

   B. Since the receipt of the subject Bulletin at the Trojan Nuclear Plant (August 1, 1979), two licensed operators have been provided in the control room at all times during operation to ensure the immediate trip of RCPs and associated follow-up actions required during such an occurrence. Appropriate documentation for this change has been completed at Trojan.

   Independent of IE Bulletin 79-06C, Administrative Procedures at Trojan already require a second licensed operator to be in the control room following a reactor trip until stable conditions are reached in one of the operating modes. This would include any reactor trips caused by safety injection. Although we are complying with this short-term action, it is not considered necessary to have two licensed operators just to ensure tripping of RCPs. Therefore, we consider the assignment of two licensed operators to be an interim measure.

Short-Term Action

2. Perform and submit a report of LOCA analyses for your plants for a range of small-break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters,

determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200°F is identified.

PGE Response

2. A series of Loss-of-Coolant Accident (LOCA) analyses for a range of break sizes and a range of time lapses between initiation of break and pump trip applicable to the two-, three- and four-loop plants has been performed by Westinghouse for the Westinghouse Owner's Group. A proprietary report summarizing the results of these analyses will be submitted to Mr. D. F. Ross (NRC) by Mr. Cordell Reed (Chairman, Westinghouse Owner's Group) on August 31, 1979. This report includes the maximum peak cladding temperatures (PCTs) for each break size and pump shutoff time considered. It is concluded that if the reactor coolant pumps are tripped prior to the reactor coolant system pressure reaching 1250 psia, the resulting PCTs are less than or equal to those reported in the FSAR. As we stated in the response to Item 1.A above, the current Emergency Instruction at Trojan requires immediate stopping of the RCPs upon HPI initiation, which is conservative relative to the 1250 psia criterion.

   The Westinghouse analyses show that there is a finite range of break sizes and RCP trip times, in all cases 10 min or longer, which will result in PCTs in excess of 2200°F as calculated with conservative Appendix K models. For these cases, the operator would have at least 10 min to trip the RCPs following the break, especially in light of the conservatism in the calculations. This is appropriate for manual rather than automatic action, based on the guidelines for termination of RCP operation presented in WCAP-9600, "Small Break Analysis for Westinghouse NSSS Systems".

Short-Term Action

3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For B&W-designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a high level, following RCP trip, to promote natural circulation flow.

PGE Response

3. The guidelines developed by the Westinghouse Owner's Group were submitted to the NRC in Section 6 and Appendix A of WCAP-9600. The analyses provided as the response to Item 2

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above are consistent with these guidelines. Hence, no changes to the guidelines for operator action are needed for either LOCA or non-LOCA transients.

Short-Term Actions

4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidelines developed under Item 3 above.

PGE Response

4. The action taken in response to Item 1 above is sufficient as an interim measure and there is no immediate need for changing Trojan emergency procedures. However, the Westinghouse Owner's Group has a continuing effort to review the necessity for revising emergency procedures, including such considerations as operation of the reactor coolant pumps. The following is the expected schedule for revising the LOCA, steamline break and steam generator tube rupture emergency procedures if changes are decided to be necessary:

Mid-October 1979:	Guidelines which have been reviewed by the NRC will be provided to each member utility. Utility personnel responsible for writing procedures will meet with the Owner's Group Subcommittee on Procedures and Westinghouse to provide the background for revising specific plant emergency procedures, if appropriate.
Mid-December 1979:	Plant-specific proeedures will be revised, if appropriate.
Mid-February 1980:	Revised procedures will be implemented and operators trained, if appropriate.

Short-Term Action

5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

PGE Response

5. Analyses related to inadequate core cooling and definition of conditions under which a restart of the RCPs should be

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attempted will be performed by the Westinghouse Owner's Group. Resolution of the requirements for the analyses, an acceptable schedule for providing the analyses, and appropriate guidelines and procedures resulting from the analyses will be arrived at between the Westinghouse Owner's Group and the NRC Staff.

Long-Term Action

1. Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

   PGE Response

   1. As discussed in the response to Short-Term Item 2, we do not believe that automatic tripping of the RCPs is required based on the analyses that have been performed and the guidelines that have been developed for manual RCP tripping. We propose that further discussion of this item be delayed until the NRC Staff has completed its review of the Owner's Group submittal of August 31, 1979.

PORTLAND GENERAL ELECTRIC COMPANY

TROJAN NUCLEAR PLANT

July 3, 1979

Revision 7*

POOR ORIGINAL

SAFETY-RELATED

*List of Effective Pages:

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EMERGENCY INSTRUCTION EI-1
LOSS OF REACTOR COOLANT

APPROVED BY [signature] DATE 7/3/79

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NOT UPDATED

A. SYMPTOMS

Listed below are the symptoms which may indicate a large leak in the reactor coolant system which will result in a loss of reactor coolant:

1. Pressurizer low pressure.
2. Pressurizer low level.
3. High containment pressure.
4. High containment humidity.
5. High containment recirculation sump level.
6. High containment radiation alarm.

B. AUTOMATIC ACTIONS

1. Reactor trip.
2. Turbine trip.
3. Safety injection is initiated.
4. Containment spray may be initiated.

C. IMMEDIATE OPERATOR ACTIONS

1. Verify reactor trip, turbine trip, and safety injection has occurred.

   CAUTION: If pressurizer pressure drops to 1765 psig and there is no automatic safety injection, manually start safety injection.

2. STOP reactor coolant pumps.
3. Verify all engineered safeguards valves and equipment are aligned and operating with status lamp panel.
4. Verify safety injection flow when pressure is below pump's shutoff head.

5. Verify spray initiated if the containment pressure reaches the high-high set point.

6. Verify the ECCS is keeping the RCS pressure above saturation. The incore thermocouples may provide better indication of core temperatures than the RTD bypass loops after the RCP's are turned off. If it is not possible to keep RCS pressure above saturation, see subsequent action for response.

7. When $50^{o}F$ subcooling can be maintained, AND if continued operation of the charging pumps would result in unsafe plant conditions, stop the charging pumps.

   CAUTION: A. Do not stop the safety injection or residual heat removal pumps for at least 20 minutes unless continued operation is likely to result in unsafe plant conditions.

   B. If it becomes necessary to reset containment isolation the attached list must be verified before resetting.

   C. Maintaining pressurizer level alone may not prevent excessive boiling in the RCS and resultant voids that may compromise the core cooling capability and natural circulation. Keeping pressure within the limits of the pressure temperature curves of figure 3.2 of the CROCTRM ensures saturation is not reached.

8. If both RHR pumps are running, manually isolate trains by closing RHR cross-connect valves MO-8716A/B.

D. SUBSEQUENT OPERATOR ACTIONS

1. If there is an increasing pressurizer relief tank level, pressure and/or temperature, along with a high relief line temperature and the pressurizer pressure is below 1765 psig, isolate the POR's to see if one is stuck open.

2. If pressurizer pressure and/or level are decreasing and Tave is remaining constant, a loss of coolant accident is indicated. It may further be distinguished from a loss of secondary coolant or S/G tube rupture as follows:

   a. An increase in containment pressure, a containment high radiation alarm, and rising sump water level indicates a loss of coolant accident.

   b. An increasing pressurizer relief tank level, pressure, and/or temperature with possibly a high relief line temperature after both POR's are isolated indicates a loss of coolant accident due to a stuck open safety valve.

   c. A condenser air removal equipment radiation alarm or a steam generator blowdown radiation alarm indicates a steam generator tube rupture.

   d. Abnormally low pressure in one or more steam generators, coincident with low pressurizer pressure and level and decreasing Tave indicate a main steam line break or feed line break.

CAUTION: Do not override automatic actions of engineered safety features without careful review of plant conditions and only then if continued ESF operation will result in unsafe plant conditions.

Do not make operat-...al decisions based on a single plant paramater or indication when a confirmatory indication is available, for example, pressurizer level without confirming with pressurizer pressure.

3. If it is determined by the above descriptions that the accident is a loss of reactor coolant, proceed to step 4. If the accident is not a loss of reactor coolant, proceed to the appropriate Emergency Instruction.

4. If plant conditions require a planned evolution, i.e. stopping unneeded RHR pumps, the safety injection signal may be reset after 20 minutes.

   CAUTION: In the event of a loss of off-site power following manual blocking an automatic SI, the only loads that will re-sequence onto the diesel generator are those initiated by the shut-down sequencer. All other ESF loads required to be in operation as a result of the initial safety injection, must be manually re-started by the operator.

5. Implement the Emergency Plan.

6. If the RCS has spent a period of time below saturation or RCS samples show cladding damage or a buildup of hydrogen gas in the RCS, start the containment hydrogen recombiners and mixing fans and periodically vent the pressurizer to the pressurizer relief tank. If WGDT's are full, it may be necessary to allow the PRT rupture disc to blow, venting gases to the containment. If RCP's are available, maximize pressurizer sprays to aid in degassing RCS and dissolving any voids which may now exist in the vessel head area.

7. When the RWST LO LEVEL annunciator actuates, start aligning the safety injection system to take suction from the containment recirculation sump as follows:

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   NOTE: Ensure the residual heat removal (RHR) pumps tripped automatically on RWST LO LEVEL signal.

   a. Open RHR heat exchanger (Hx) component cooling water (CCW) inlet valves MO-3210A and MO-3210B.

   b. Close RHR pump suction valves MO-8700A, MO-8700B and MO-8812 from the RWST.

   c. Open RHR pump suction valves MO-8811A and MO-8811B from recirculation sump.

      NOTE: These valves are interlocked such that MO-8700A/B must be closed before MO-8711A/B can be opened.

   d. Verify the RHR Hx outlet cross-connect valves MO-8716A and MO-8716B are closed to provide train separation if both RHR pumps are operable. If both pumps are not operable leave the valves open.

e. START west/east RHR pumps A/B.

f. Flow to the vessel through the two cold leg injection lines can be checked by using west RHR Hx "A" outlet flow FI-971A and FI-971B, and east RHR Hx "B" outlet flow FI-970A and FI-970B.

g. Close safety injection pump miniflow line block valves MO-8813 and MO-8814.

h. Open RHR pump discharge valve isolation valve MO-8804B to the safety injection pump suction.

NOTE: Valve MO-8804B is interlocked such that the reactor coolant system to RHR system isolation valves MO-8701 or MO-8702 must be closed, safety injection pump miniflow block valves MO-8813 or MO-8814 must be closed, and recirculation sump isolation valve MO-8811B must be open before MO-8804B can be opened.

i. Open RHR pump discharge isolation valve MO-8804A to the charging pump suction.

NOTE: Valve MO-8804A is interlocked such that the reactor coolant system to RHR system isolation valves MO-8701 or MO-8702 must be closed, safety injection pump miniflow block valves MO-8813 or MO-8814 must be closed, and recirculation sump isolation valve MO-8811A must be open before MO-8804A can be opened.

j. Verify that east RHR pump "B" is supplying the safety injection pumps (increased safety injection pumps discharge pressure, PI-919, PI-923).

k. Open RHR discharge to safety injection pump suction valves MO-8807A and MO-8807B.

l. Close safety injection suction valve MO-8806 from RWST.

m. Close charging pump suction valves MO-112D and MO-112E from RWST.

8. Shift the suction on both spray pumps one at a time as follows:

a. STOP west/east containment spray pump A/B.

b. Close containment spray pump suction valve MO-2050 A/B from the RWST.

c. Open containment spray pump suction valve MO-2052 A/B from the recirculation sump.

d. START west/east containment spray pump A/B.

e. Repeat steps 1 through 5 and shift the other pump.

f. When the NaOH TANK LO LO LEVEL annunciator actuates, close spray additive valves MO-2056A and 2056B to prevent air binding of the spray pumps.

9. If the hydrogen recombiners and mixing fans have not already been started per 6 above, start them now.

10. Approximately 17 hours after the accident, depending on the recirculation sump boron concentration, align the safety injection system for hot leg/cold leg recirculation as follows:

   a. Open RHR to RCS hot legs isolation valve MO-8703.

   b. Open cross tie isolation valve MO-8716A.

   c. Close RHR Cold Leg Injection valves MO-8809A/B.

   d. Verify hot leg recirculation flow on FI-600.

   e. Open hot leg isolation valve MO-8802A.

   f. Verify flow to the reactor coolant system through the hot leg header on FI-918.

   g. Open hot leg isolation valve MO-8802B.

   h. Verify flow to the reactor coolant system through the hot leg header on FI-922.

   i. Close Cold Leg Safety injection valves MO-8821 A/B and MO-8835.

11. Sample the recirculated coolant to determine boron concentration as follows:

   a. Every 15 minutes for the first hour.

   b. Every hour for the next 3 hours.

   c. Every 4 hours after the first 4 hours.

   d. Maintain boron concentration greater than 2,000 ppm B.

   e. Use emergency borate mode to increase the boron concentration, as required.

EMERGENCY INSTRUCTION EI-1

LOSS OF REACTOR COOLANT

VALVE VERIFICATION BEFORE RESETTING CIS

Before resetting Containment Isolation, you must verify the following valves are in the indicated positions:

Valve	Description	Panel	Position	Verification
MO-4180	Containment Sump Discharge	C19	Pull to Lock	________
CV-4181	Containment Sump Discharge	C19	Pull to Lock	________
CV-5652	Accum Sample Isol	C17	Auto After Close	________
CV-5661	Reactor Coolant Drain Tk Sample	C17	Auto After Close	________
CV-4000	Reactor Coolant Drain Tk N_2 Supply	C17	Auto After Close	________
CV-4006	Reactor Coolant Drain Tk Outlet	C17	Auto After Close	________
CV-4301	Gas Collection Header Valve	C17	Auto After Close	________
CV-4471	Instrument Air to Containment	C17	Auto After Close	________
CV-4470	Service Air to Containment	C17	Auto After Close	________
CV-10001	Containment Purge Supply	C17	Auto After Close	________
CV-10004	Containment Exhaust	C17	Auto After Close	________
MO-10002	Containment Purge Isol	C17	Auto After Close	________
MO-10003	Containment Exhaust Isol	C17	Auto After Close	________
CV-10014	Chilled Water Return	C17	Auto After Close	________
CV-10015	Chilled Water Supply	C17	Auto After Close	________
MO-2810	A Steam Generator Blowdown Isol	C15	Auto After Close	________
MO-2813	B Steam Generator Blowdown Isol	C15	Auto After Close	________
MO-2812	C Steam Generator Blowdown Isol	C15	Auto After Close	________
MO-2808	D Steam Generator Blowdown Isol	C15	Auto After Close	________
CV-2811	A Steam Generator Blowdown Sample	C15	Auto After Close	________
CV-2880	B Steam Generator Blowdown Sample	C15	Auto After Close	________
CV-2814	C Steam Generator Blowdown Sample	C15	Auto After Close	________
CV-2809	D Steam Generator Blowdown Sample	C15	Auto After Close	________

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