T/C

Portland General Electric Company

Charles Goodwin, Jr. Assistant Vice President

June 25, 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

RECEIVED JUN 29 1979 REGION V INSPECTION & ENFORCEMENT

Dear Sir:

Subsequent to the Three Mile Island accident, a series of IE Bulletins were issued by the NRC: IE Bulletin 79-06 dated 4/11/79, IE Bulletin 79-06A dated 4/14/79, and Revision 1 to IE Bulletin 79-06A dated 4/18/79. Portland General Electric Company had reviewed these Bulletins and responded accordingly. Our original response was submitted on April 24, 1979 and updated on May 4, 1979. The 30-day response to Bulletin Item 13 was submitted on May 18, 1979.

A May 22 letter from A. Schwencer of the NRC to C. Goodwin, Jr. of PGE notified us of a completion of a preliminary review of licensee responses to IE Bulletin 79-06A, including Revision 1, and of a meeting to be held in Bethesda, Maryland, on May 30, 1979. At that meeting, it was decided that the draft SER prepared by the NRC Staff and any requests for specific information would be transmitted to each licensee prior to formal publication.

We have reviewed the working paper transmitted by the NRC Staff, dated June 7, 1979, and the Request for Additional Information. Enclosed is our response to these documents, identified by the applicable Bulletin item number.

Sincerely,

C. Goodwin, Jr.

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

CG/KM/4sb5A7
Attachments

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

Office of Management Information and Program Control

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Bulletin Item 2

Revise your response after a thorough review of all transient and accident conditions based on insight gained from TMI-2 to (a) assure that the action steps specifically warn of potential for voiding, with a description of all instrumentation which might provide indication of potential or actual voiding, (b) specifically address operator actions, based on operational modes and instrument indications discussed above, for terminating conditions tending to lead to void formation, and (c) provide operators with guidance for enhancing core cooling, given the unexpected condition of actual voiding in the primary system.

Summarize the results of this review, including the revisions to procedures. Identify all instrumentation which might be utilized in void recognition and to summarize the review results and actions taken with regard to the natural circulation mode of operation.

PGE Response

a. Emergency Operating Instruction EI-1 "Loss of Reactor Coolant" was modified as shown in Attachment 1, and the "CAUTION" statement in Immediate Action Step 6 specifically warns operators of the potential for voiding as follows:

"B. 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 [Control Room Operating Curves and Tables Reference Manual] ensures that saturation is not reached."

Figure 3.2 referred to in the "CAUTION" statement has been revised to ensure that 50 degrees sub-cooling is maintained, per the requirements of Bulletin 79-06A (see Attachment 2). Furthermore, we are preparing a procedure which provides a backup means of indication for some primary Plant parameters for long-term monitoring. Attachment 3 is a list of backup means of indication and will be a part of the new procedure. The discussion associated with pressurizer level describes Plant parameter behavior under voiding conditions. This procedure, which will be in effect by June 30, 1979, along with Figure 3.2, will be valuable aids to the operator to guide him in maintaining safe Plant conditions at all times, including adverse conditions.

b. A specific operator action is to ensure that the ECCS systems are maintaining RCS pressure above saturation, per Immediate Operation Action Step 6 of the attached procedure, EI-1 (Attachment 1), preventing void formation.

"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."

A subsequent action mentioned in Step 6 above is shown in EI-1 (Attachment 1).

c. Guidance for enhancing core cooling in the unlikely event of actual voiding in the primary system has been provided in the Immediate Operation Action Step 6 of the attached procedure, EI-1. Any steam voids would be eliminated by maintaining or relieving pressure above saturation by at least 50 degrees. These measures are to be taken when the potential for voiding exists, and enhance core cooling by providing a flow of as much cold injection water as necessary to maintain pressure above saturation, or as high as possible if the leak is large. The discussion in Attachment 3, under Steam Generator Level (Loop T_H and T_C), describes how to ensure that natural circulation loop flow is being maintained.

Bulletin Item 3

Provide a commitment to maintain all pressurizer low-level bistables in a tripped condition whenever a safety injection signal from coincident pressurizer pressure and level is required to be operable.

PGE Response

The Trojan plant was temporarily modified by placing the pressurizer water level bistables in the tripped position on April 25, 1979, while in Mode 1. This action removed the water level coincidence requirement for safety injection actuation, thus providing the temporary protection required.

Since that time, the Plant has remained in Mode 5 for annual maintenance. Included in the schedule of modifications and repairs for the Plant during this time is a change to pressurizer safety injection logic. This change removes the pressurizer water level coincidence signals and substitutes a two-out-of-three low pressurizer pressure logic to initiate safety injection. These modifications are the subject of License Change Application 54 submitted on June 6, 1979 to the Director of Nuclear Reactor Regulation. This License Change Application will be approved and the necessary modifications will be made before the Plant returns to power operation.

After these modifications, the pressurizer water level will no longer be involved in the development of a safety injection signal. The pressurizer water level signals will provide control and indication only. Thus the requirement of this Bulletin item is no longer applicable to Trojan.

Bulletin Item 4

Discuss the procedures that would be necessary in order to allow reactor coolant pumps to be operated under containment isolation conditions. Discuss whether or not these procedures have been written and incorporated into Plant procedures, or your schedule for doing so.

PGE Response

As discussed in our original response to Bulletin Item 7.c, in the event of a Safety Injection Signal (SIS) or Containment Isolation Signal (CIS), the reactor coolant pump cooling water supplies, Component Cooling Water System (CCWS), and seal water return lines, Chemical Volume and Control System (CVCS), would automatically be isolated. This would preclude continued operation of the reactor coolant pumps. Therefore, new procedures to allow reactor coolant pumps to be operated under containment isolation conditions are not possible at this time.

A design review of the scope of the modifications necessary to operate reactor coolant pumps under safety injection and/or containment isolation conditions is underway and is expected to be completed by July 31, 1979. There are eight valves involved in the cooling water and seal water lines for which the logic must be changed to allow them to remain open or be opened after an SIS or CIS. These valves are as follows:

CCW Supply to RCP Motors	MO 3296, MO 3294
CCW Return from RCP Motors	MO 3320, MO 3300
CCW Supply to Seal Water Heat Exchanger	MO 3295
CCW Return from Seal Water Heat Exchanger	MO 3319
CVCS Seal Water Return to Volume Control Tank	MO 8100, MO 8112

A comprehensive safety evaluation still has to be performed to determine the radiological, seismic, and emergency power effects of this modification. This safety analysis is expected to be completed by July 31, 1979. At that time, we plan to provide a description of the design modifications, a description of the necessary operating procedures, and a schedule for implementation of modifications to allow the reactor coolant pumps to continue to operate after SIS and CIS.

Bulletin Item 7.a

You do indicate that your review and revision of training documents has been performed, but you do not indicate that you have completed the required review of operating instructions. In addition, you do not indicate that the Bulletin guidelines are being incorporated, since you appear to address only reactor vessel integrity and safety injectien. Therefore, it is requested that you provide assurance that operating procedures and training instructions will be reviewed to ensure that operators will not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe Plant conditions. Provide a schedule for completion of the review of operating procedures and training instructions, incorporating such modifications as are necessary to comply with item 7.a of the Bulletin.

PGE Response

The Operating Instructions were reviewed and the Bulletin guidelines implemented by a modification to Subsequent Action Instuctions in EI-1 (Attachment 1). The following "CAUTION" statement was added to EI-1:

"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 operational decisions based on a single plant parameter or indication when a confirmatory indication is available, for example, pressurizer level without confirming with pressurizer pressure."

The basis for this "CAUTION" statement is that "continued ESF operation" may result in unsafe conditions if allowed to continue unchecked in the event of a small leak which can be overcome by our centrifugal charging pumps. The shutoff head of the centrifugal charging pumps is about 200 psi higher than safety valve settings, and a continual popping of these safety valves could be expected to result in valve seat leakage which cannot be isolated.

Bulletin Item 7.b

Your response includes criteria for HPI combinations which do not comply with those specified in the Bulletin. Therefore, you must provide assurance that operating procedures will be modified to keep high-pressure injection and charging pumps in operation in accordance with the criteria specified in Item 7.b of the Bulletin. Provide a schedule for completion of the review of operating procedures incorporating such modifications as are necessary to comply with Item 7.b of the Bulletin.

PGE Response

Plant procedures have been revised to caution the operators not to override the automatic function of Engineered Safety Features (ESF) as discussed in above response to Bulletin Item 7.a. At Trojan, the ESF Systems consist of Centrifugal Charging Pumps (HPI), Safety Injection Pumps (LPI) and Residual Heat Removal Pumps (LPI). The termination of either High Pressure Injection (HPI) or Low Pressure Injection (LPI) pumps is considered to involve the overriding of the automatic function of ESF.

The HPI pumps at Trojan have a shutoff head approximately 200 psig higher than the pressurizer safety valve set point. In the event of a small break or an inadvertent Safety Injection Signal (SIS), the Reactor Coolant System (RCS) pressure may reach the safety valve setpoint in less than 20 minutes. Therefore, Plant procedures are being revised to allow the HPI pumps to be shut off when RCS pressure can be maintained above the saturation limit as shown in Figure 3.2 (Attachment 2). The LPI pumps will be kept running for at least 20 minutes unless continued operation is likely to result in unsafe Plant conditions. If a safety injection signal has been generated and its initiation is known to be inadvertent, and if Plant parameters have reached stable conditions with Reactor Coolant System temperature being maintained at least 50 degrees below the saturation temperature, both HPI and LPI pumps may be shut off.

The Emergency Instructions must be maintained free of confusion and ambiguity in order for the operators to take adequate and necessary corrective actions. Although we realize the seriousness and consequences of terminating the HPI and LPI pumps prematurely without careful review of Plant conditions, the IE Bulletin guidance for allowing the pumps to operate for more than 20 minutes or until unsafe Plant conditions exist is subject to different interpretations by different operators and inspectors. Our revisions to the Emergency Instructions regarding operation and overriding of HPI and LPI pumps have attempted to take this concern into consideration.

Bulletin Item 7.c

Your response stated that a design review of the necessary modifications required to operate reactor coolant pumps (RCP) under safety injection conditions would be complete as of July 31, 1979. At that time, we require a schedule for implementation of any design modifications required to keep the RCP's operating. In addition, we will require a schedule for completion of the review of operating procedures incorporating such modifications as are necessary to comply with Item 7.c of the Bulletin.

PGE Response

Please refer to PGE's response to Bulletin Item 4 which covers both Bulletin Items 4 and 7.c.

Bulletin Item 7.d

Identify those specific parameters other than pressurizer level identified for operator use in evaluating Plant conditions, and verify that these parameters have been included in appropriate operating procedures.

PGE Response

Specific parameters for evaluating Plant conditions are identified under "Symptoms" at the beginning of each applicable emergency procedure, as is the case in the attached EI-1. The bodies of our procedures contain discussions in more detail of how to use these parameters as a trouble-shooting guide once the necessary immediate actions have been carried out. The immediate actions ensure that the ECCS subsystems are functioning properly and contain the precautions necessary to prevent voiding. Please refer to the attached EI-1 for identification of those specific parameters other than pressurizer water level used by the operators in evaluating Plant conditions.

Bulletin Item 8

Please submit a summary of the results of the reviews of alignment requirements and procedures controlling manipulation of safety-related valves and any revisions necessary within 2 weeks after completion of the review. (LTM: Determine whether the Technical Specifications require periodic surveillance of locked valves. If not, add the following request for information.)

Also review Plant procedures and revise them as necessary to ensure that locked safety-related valves are subjected to periodic surveillance. Submit a summary of the results of the review.

PGE Response

All ESF flow paths were identified on the Plant Piping and Instrument Drawings. All valves in these flow paths were checked for correct positioning requirement. Each valve was checked against Plant administrative control procedures (Plant operating tests, locked valve lists, Plant engineering tests, and valve line-ups) for proper positioning and controls. Further, valves not directly in the ESF flow path that might affect ESF performance were similarly checked, i.e., valves which might allow bypass flow or dilution of borated water sources. ESF systems that were checked were:

Containment Isolation System	FSAR Fig. 6.2-1 - 6.2-47
Safety Injection System	6.3-1
Centrifugal Charging Pumps	6.3-1
Boron Injection System	6.3-1
Residual Heat Removal System	5.5-7
Containment Spray System	6.4-1
Hydrogen Vent System	6.2-48
Hydrogen Sampling System	6.2-48
Component Cooling Water System	9.2-4
Service Water System	9.2-1
Emergency Diesel Generators	9.5-3
Containment Coolers	9.2-4

All valve position requirements were found to be correct. Valves in vital flow paths are either locked in position or verified to be in proper position monthly per Technical Specification requirements. Two valves were added to the locked valve list. These were:

Valve 8735 - RHR Common Discharge to RWST (FSAR Fig. 5.5-7)
Valve MD050 - Auxiliary Feedwater Pumps Suction from Condensate Storage Tank (FSAR Fig. 10.4-2).

Control of locked valves is described in Administrative Order AO-3-13. Locked valves must be controlled in Modes 1, 2, 3 & 4. Position changes of locked valves must be approved by the Shift Supervisor. All position changes are maintained on a special list by the Control Operator (the locked valve list). When the Plant is in Modes 5 and 6, permission of the Shift Supervisor must still be obtained to reposition a locked valve. However, the locked valve list does not have to be maintained in Modes 5 and 6. Prior to Plant heatup to Mode 4, the locked valve list is required to be updated.

Bulletin Item 10.c

Please identify the level of authority required for removing and returning systems to service, and describe the method used for transferring information about the status of safety-related systems of shift change.

PGE Response

In general, the Shift Supervisor is the level of authority required for removing and returning systems to service. Removal and return of safety-related equipment from/to service for periodic surveillance in accordance with test procedures reviewed and approved by the Plant Review Board require authorization by the Control Operator. For the removal of equipment from service for maintenance, the Shift Supervisor's approval with the Operations Supervisor's concurrence must be received under the following circumstances:

a. When removal from service is not by PRB's previously approved procedure, and

b. When operability is required in the mode that the Plant is in or may be in by the time the equipment is returned to service.

Authorization for removal of safety-related equipment from service for maintenance is documented on outage worksheets. At shift change, the oncoming Control Operator and Shift Supervisor are informed of tests in progress, and outage worksheets are reviewed at the beginning of each shift.

Bulletin Item 12

Confirm your review of operating modes and procedures for removal of hydrogen from the Containment to assure that such removal can be accomplished. Briefly discuss various methods for dealing with hydrogen in the Containment and provide your schedule for completing procedures incorporating these methods.

PGE Response

We have reviewed modes and procedures for removal of hydrogen from the Containment. We have revised applicable procedures in the manner shown in Subsequent Action Step 6 of the attached EI-1 (Attachment 1). This recognizes the insight we have gained based on TMI-2 that hydrogen can be evolved earlier in post-LOCA conditions under certain circumstances. If these circumstances have existed, hydrogen removal is commenced with the equipment which is already installed and operable during power operations. A detailed discussion of the various methods for dealing with hydrogen in Containment (sampling, mixing, recombiners, and venting) is provided in FSAR Section 6.2.5. The Trojan hydrogen recombiners are tested monthly per Technical Specifications to verify operability.

PORTLAND GENERAL ELECTRIC COMPANY

TROJAN NUCLEAR PLANT

April 30, 1979

Revision 6

SAFETY-RELATED

EMERGENCY INSTRUCTION EI-1
LOSS OF REACTOR COOLANT

APPROVED BY BDWithers DATE 4/30/79

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, m... lly 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.

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

   B. 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.

7. 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 operational 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.

   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 restarted 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:

   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.

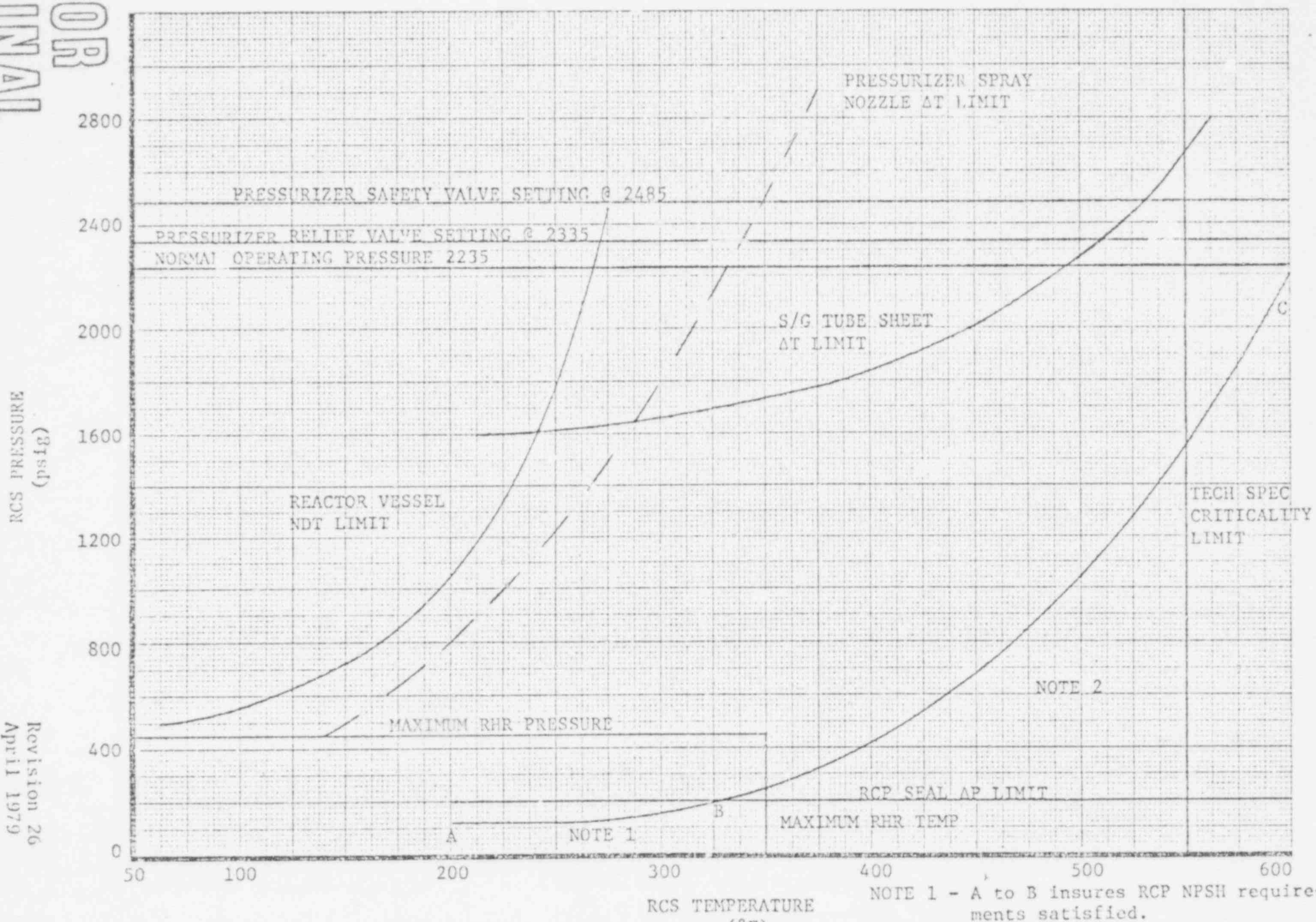
LOSS OF REACTOR COOLANT

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	

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



NOTE 1 - A to B insures RCP NPSH requirements satisfied.

NOTE 2 - B to C insures temperature maintained 50°F below saturation temperature

Figure 3.2

Revision 26
April 1979

302063

POOR ORIGINAL

ATTACHMENT 3

TROJAN NUCLEAR PLANT
POST-ACCIDENT BACKUP MONITORING CAPABILITY

Normal Instrument Which may be Lost	Backup Instrument Outside Containment[a]	Use of Backup Instrument
(A) Pressurizer Water Level	(1) Differential Pressure Measured Outside Containment	Read directly from remote indication. PGE is presently in the process of adding a differential pressure transmitter utilizing existing pressurizer sample lines located outside Containment. If operation of the instrumentation is tested and found successful, specific test procedures (and operating procedures required in the event of an accident requiring long-term monitoring) can be made available before commercial operation is resumed, but are not available at this date (6/15/79).
	(2) Pressurizer Steam Temperature and Water Temperature RTDs	By use of pressurizer temperature indication from existing pressurizer RTDs, a comparison of pressurizer steam and water temperatures can be made by the operator. If the water temperature is approaching the steam temperature, the pressurizer may be empty. However, if the steam temperature is approaching the water temperature, then the pressurizer is going solid. [NOTE: The existing pressurizer RTDs contain teflon-insulated lead wires and may not be satisfactorily moisture tight for the post-LOCA environment. The operability of these RTDs in such an environment will be enhanced by a modification

ATTACHMENT 3

Normal Instrument Which may be Lost	Backup Instrument Outside Containment[a]	Use of Backup Instrument
		to the RTD wiring and encapsulation of the terminals with a high temperature sealing compound. This work will be completed prior to Plant operation.]
	(3) RCS Pressure and Charging Pump Flow	Read RCS pressure directly. The Reactor Coolant System is going solid if RCS pressure is rapidly increasing at the same time that the charging flow decreases.
	(4) RCP Ammeter	Read ammeter directly. A fluctuating or low reactor coolant pump ammeter reading indicates approach of two-phase (steam/water mixture) flow.
	(5) Nuclear Instrumentation	Read directly. An increasing count rate indicates a decrease in shielding as water level approaches core.
(B) Steam Generator Water Level	(1) Differential Pressure Measured Outside Containment	Read directly from remote indication. PGE is presently in the process of adding signal transmitters for each steam generator utilizing the existing steam generator blowdown lines and main steam lines located outside Containment. If operation of the instrumentation is tested and found successful, specific test procedures (and operating procedures required in the event of an accident requiring long-term monitoring) can be made available before commercial operation is resumed, but are not available at this date (6/15/79).

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ATTACHMENT 3

Normal Instrument Which may be Lost	Backup Instrument Outside Containment[a]	Use of Backup Instrument
	(2) Steam Generator Pressure	Read pressure directly. (a) The atmospheric relief valve can be periodically opened. If the steam generator is nearly dry, it will rapidly depressurize. (b) While using the condenser steam dump valves, the operator can listen for flow through the dump valve and confirm that the steam generator pressure is stable near the no load value.
	(3) RCS Loop T_{hot} and T_{cold}	Safe shutdown conditions should show an RCS loop ΔT (i.e., $T_{hot} - T_{cold}$) less than full load ΔT. Steam generator level has predictable effects on RCS loop ΔT characteristics.
	(4) Fill Steam Generator Solid	This method would be considered a last resort. Indication is (a) water flow in steam lines and (b) possible discharge of two-phase water through atmospheric dump valves.
(C) Reactor Coolant System Pressure	(1) Centrifugal Charging Pump (High Pressure) Discharge Pressure	Charging pump discharge pressure is available from Panel C12 in the control room and from the Plant computer.
	(2) Centrifugal Charging Pump Flow	Charging pump flow is available at Panel C12 in the control room and from the Plant computer. With the charging pump running, the pump curve can be examined to estimate discharge pressure.

ATTACHMENT 3

Normal Instrument Which may be Lost	Backup Instrument Outside Containment[a]	Use of Backup Instrument
	(3) Safety Injection Pump (Intermediate Pressure) Discharge Pressure	Safety injection pump discharge pressure is available at Panel C19 in the control room. This pressure is only accurate after the flow starts. If there is no safety injection flow, this indicates that the RCS pressure is greater than the safety injection pump discharge pressure.
	(4) RHR Pump (Low Pressure) Discharge Pressure	RHR pump discharge pressure is available at Panels C12 and C13 in the control room and from the Plant computer. This pressure is only accurate after flow starts. If there is no flow, this indicates that the RCS pressure is greater than the RHR pump discharge pressure.
	(5) Pressure Indicators in Various RCS Sample Lines	Manipulate sample valves as necessary and read locally at the sample stations.
(D) RCS Loops T_{cold}, T_{hot}, Incore Thermocouples	Steam Generator Pressure	Calculate RCS temperature by use of steam table.

[a] Listed in order of preference.