Portland General Electric Company

March 30, 1990

Trojan Nuclear Plant
Docket 50-344
License NPF-1

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington DC 20555

Dear Sir:

Response Concerning the Review of the "Risk-Based Inspection Guide for the Trojan Nuclear Plant" (TAC No. 72894)

In a letter dated December 22, 1989, the Nuclear Regulatory Commission requested Portland General Electric Company to provide additional information on six items provided as part of the "Risk-Based Inspection Guide (RIG) for the Trojan Nuclear Plant", and to review and comment on the draft version of the RIG. The attachments to this letter provide the requested information and comments on the draft RIG.

Sincerely,

T. D. Walt
Acting Vice President, Nuclear

Attachments

c: Mr. John B. Martin
Regional Administrator, Region V
U.S. Nuclear Regulatory Commission

Mr. David Stewart-Smith
State of Oregon
Department of Energy

Mr. R. C. Barr
NRC Resident Inspector
Trojan Nuclear Plant

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PORTLAND GENERAL ELECTRIC COMPANY
RESPONSE CONCERNING THE REVIEW OF THE "RISK-BASED
INSPECTION GUIDE FOR THE TROJAN NUCLEAR PLANT" (TAC NO. 72894)

BACKGROUND

In a letter dated April 1, 1989, Portland General Electric Company (PGE) provided Trojan plant specific information to the Nuclear Regulatory Commission (NRC) to support the "Risk-Based Inspection Guide for the Trojan Nuclear Plant". Brookhaven National Laboratory has performed the analysis and the NRC provided the Draft Report to PGE (in a letter dated December 22, 1989) for comments and a request for additional information on six items. The PGE response for each item of additional information is provided.

NRC REQUEST NO. 1

Whether the air supply to the pressurizer power-operated relief valves (PORVs) is backed by nitrogen bottles or air accumulators.

PGE Response

The PORVs are backed by four Seismic Category I air accumulators (T-233A, T233B, T-233C, T-233D). These are shown on Final Safety Analysis Report (FSAR) Figure 5.1-1 B, "Reactor Coolant System", which is attached (Attachment 3).

NRC REQUEST NO. 2

The number of PORV actuations that the backup supply is expected to support.

PGE Response

The two air accumulators to each pressurizer PORV are capable of a combined total of 32 cycles. The basis for this is attached (Attachment 3) and is from the Reactor Coolant System Design Basis Document.

NRC REQUEST NO. 3

The number of PORVs that would be required to open to achieve sufficient core cooling in the "feed and bleed" mode.

PGE Response

Two pressurizer PORVs are required during the "feed and bleed" mode. Emergency Instruction (EI)-0, "Reactor Trip, Safety Injection, and Diagnosis", Step 12 on Page 7 of 18, references Functional Restoration (FR)-H.1, "Response to Loss of Secondary Heat Sink". Step 14 of FR-H.1 on Page 11 of 22 references operation of both PORVs. Partial copies of both EI-0 and FR-H.1 are attached (Attachment 3).

NRC REQUEST NO. 4

The success criteria for the PORVs/Safety Relief Valves (SRVs) during anticipated transient without scram (ATWS) events.

PGE Response

The success criteria during an ATWS is all five pressurizer relief valves, two out of two PORVs and three out of three SRVs. This is shown in FSAR Table 15.8-1, "ATWT Analysis Model Summaries", and is attached (Attachment 3).

NRC REQUEST NO. 5

The specific locations (e.g., Containment elevations) of the SRVs, PORVs, and block valves.

PGE Response

These valves are shown in an isometric view in the attached (Attachment 3) FSAR Figure 3.6-22, "Pressurizer Safety and Relief Lines (RC-2501R-8 and RC-2501R-12)". The Containment elevations are as follows:

SRVs	Elevation (MSL)*
PSV-8010A	119 ft 3 inches
PSV-8010B	119 ft 3 inches
PSV-8010C	119 ft 3 inches
PORVs	
PCV-455A	127 ft 0 inches
PCV-456	127 ft 0 inches
Block Valves	
MO-8000A	127 ft 0 inches
MO-8000B	127 ft 0 inches

* Mean Sea Level

NRC REQUEST NO. 6

The PORV control power supply breaker designations and locations.

PGE Response

The electrical drawing for PCV-455A is Drawing No. E-546, "Schematic Diagram Reactor Coolant System Valves", and is attached (Attachment 3). Power supply is 125 VDC, Panel D10, Breaker 72-1016, Control Building Train A switchgear room.

The electrical drawing for PCV-456 is Drawing No. E-548A and is attached. Power supply is 125 VDC, Panel D20, Breaker 72-2016, Control Building Train B switchgear room.

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PORTLAND GENERAL ELECTRIC COMPANY
COMMENTS ON "RISK-BASED INSPECTION GUIDE FOR THE TROJAN
NUCLEAR PLANT", DRAFT OF TR-A-3875-T2B, REV. 0, DECEMBER 1989

General Comments

1. There are systems in the document that are NOT Trojan specific. For example, Trojan does not identify a "High Pressure Service Water" system as shown in Table 2, Table A.1-1, and Figure A.1-1. The "Emergency Safeguards System (ESS)" as shown in Table 2 for Trojan is the Emergency Safeguards Features Actuation System (ESFAS). Trojan does not identify a "Power Conversion System (PCS)" as shown in Table 2. Also, the high head injection at Trojan is the centrifugal charging pumps (CCPs), the intermediate head injection is from safety injection (SI) pumps and accumulators, and the low head injection is the residual heat removal (RHR) pumps. This document uses just high head and low head injection modes.

2. The introduction states that the inspection guidance will be periodically revised, but does not identify who will perform this revision.

Editorial Comments

1. On Page 2 and other places throughout the document, the acronyms LHR and LPR are used for the same thing; likewise for HHR and HPR. These should be clarified as to their meaning. Also, these are not Trojan specific acronyms.

2. Page 10 uses the acronym USAR. This has been changed to be FSAR (Final Safety Analysis Report). This acronym needs to be identified in the text.

3. On Page 11, there is reference to a motor-driven auxiliary feedwater (AFW) pump. Trojan does have a motor-driven AFW pump, but it is not safety-related. Trojan has two safety-related AFW pumps, one turbine-driven and one diesel-driven. It should be clarified as to what is intended here.

4. On Page 12, there is no mention of charging pumps in Footnote 5 or 6.

5. On Page 13, Table AX-1 should be clarified to indicate that the "X" is the number listed in Table 2 on Page 12.

6. On Page A-16, the "Normal/Emergency AC Power System" is misplaced and not consistent with the other title blocks.

7. On Page A-67, what do the "?" marks mean?

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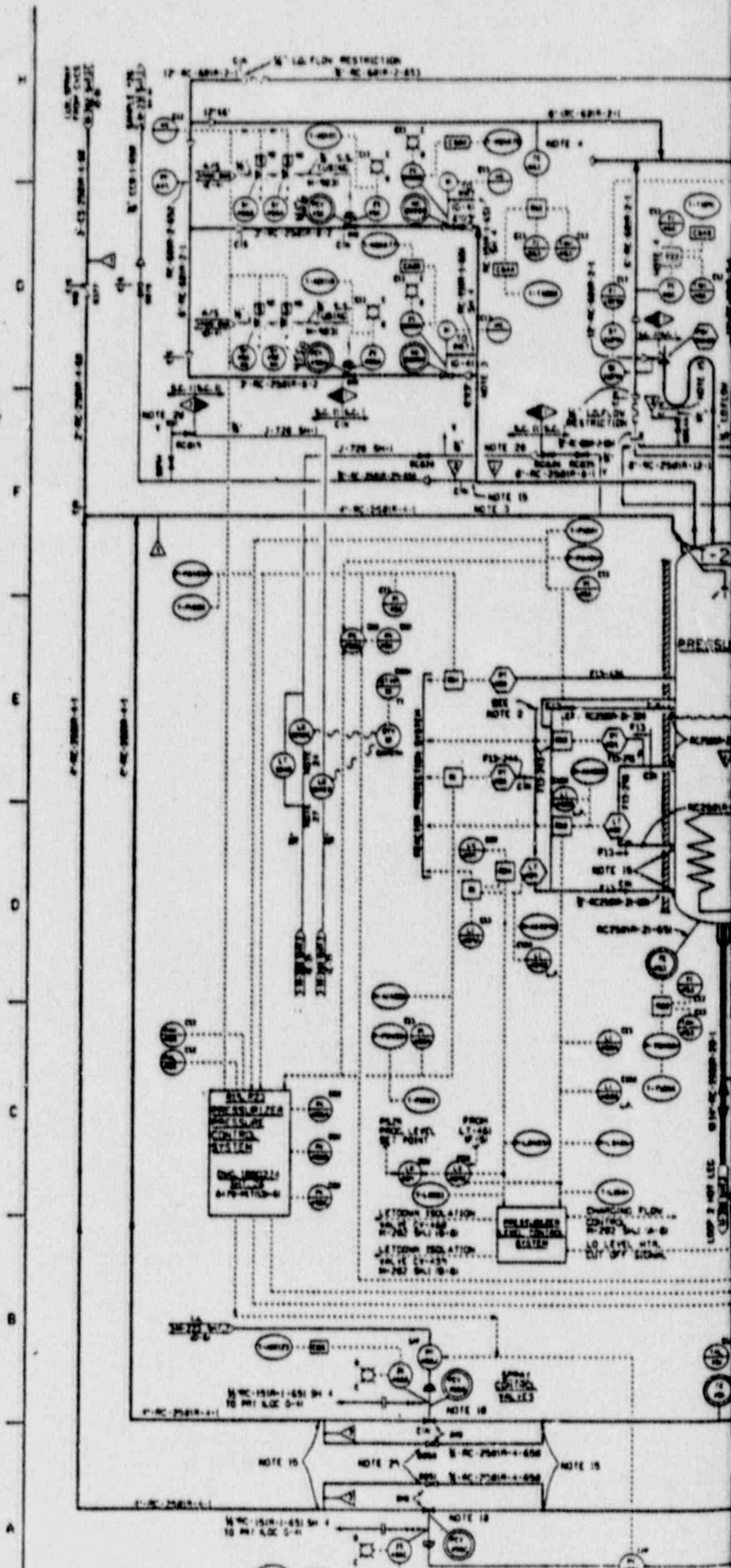
Response to #1

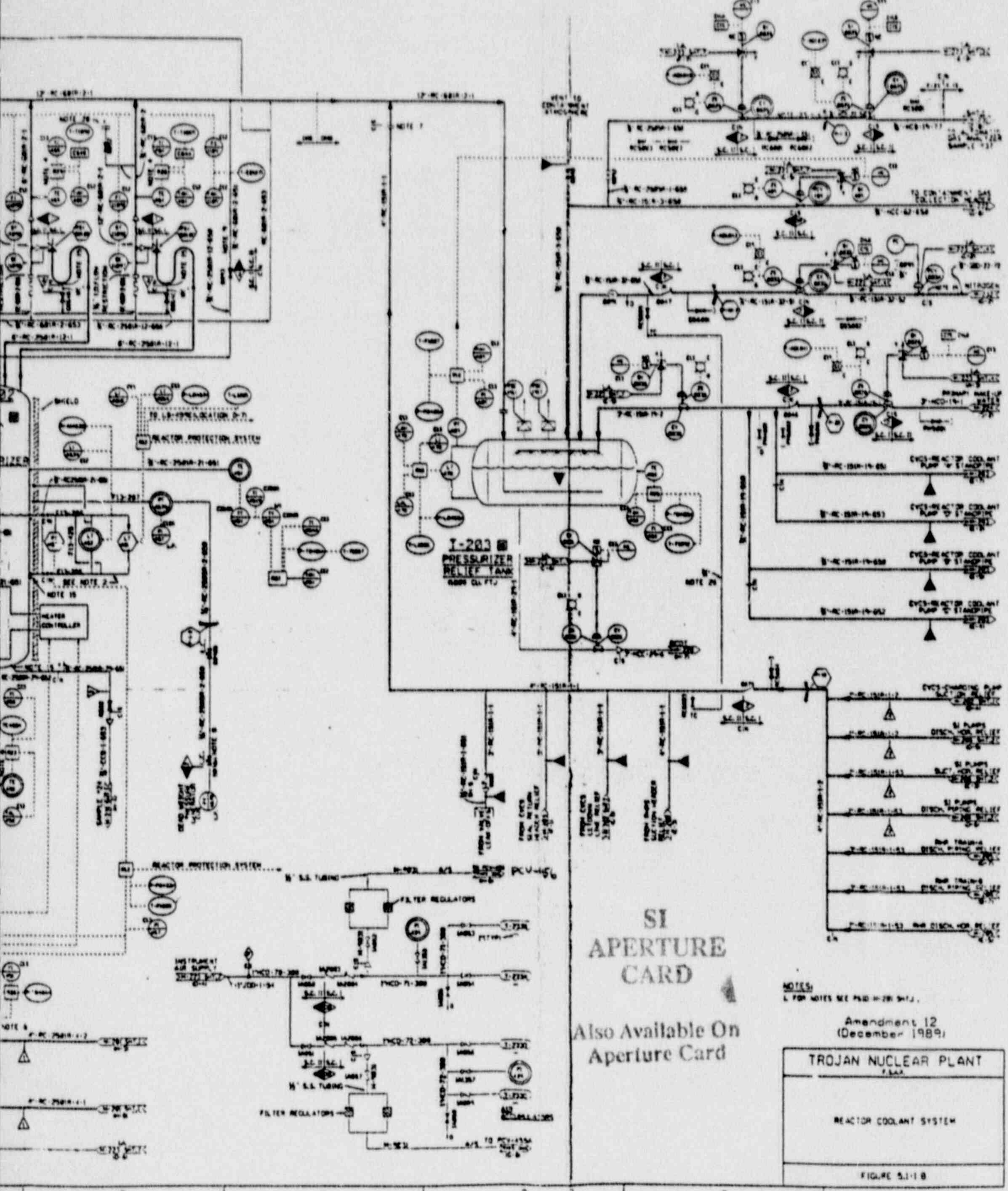
COLOR LEGEND

QUALITY GROUP 1
QUALITY GROUP 2
QUALITY GROUP [illegible]
QUALITY GROUP [illegible]

NOTES:
1. FOR FLUID SYSTEM SYMBOLS AND LEGEND SEE FIGURE [illegible].
2. FOR QUALITY GROUP APPLICATION AND CODE CLASSIFICATION FOR TROJAN PLANT SEE TABLE [illegible].
3. INSTRUMENT CONNECTION ROOT VALVES ARE NOT SHOWN. CLASSIFICATION OF INSTRUMENT CONNECTION PIPING FROM PROCESS LINE UP TO ROOT VALVE IS SHOWN WHERE APPLICABLE BY SAME COLOR LEGEND.
4. FIGURE [illegible] IS TAKEN FROM P&ID [illegible] SHT.2 REV. [illegible].
5. BY CONVENTION SOME INSTRUMENT TIE LINES SUCH AS THOSE FOR PRESSURE POINTS ARE SHOWN AS BLACK. IN SOME CASES A MINIMAL LENGTH OF TUBING OR SMALL PIPE NOT DEPICTED ON THE P&ID MAY EXIST. THIS CONVENTION IS NOT MEANT TO IMPLY THAT THIS TUBING OR PIPING IS QUALITY CATEGORY [illegible]. ANY TUBING OR PIPING BRANCHING OFF THE PROCESS PIPE MUST MEET THE QUALITY CATEGORY GUIDELINES AS DISCUSSED IN CHAPTER 3.

REFERENCE DRAWING	P&ID NO.	FIG. NO.
1. REACTOR COOLANT SYSTEM	[illegible] SHT.[illegible]	[illegible]
2. CHEMICAL & VOLUME CONTROL SYSTEM SHEET 1	[illegible] SHT.[illegible]	[illegible]
3. CHEMICAL & VOLUME CONTROL SYSTEM SHEET 2	[illegible]	[illegible]
4. RESIDUAL HEAT REMOVAL SYSTEM	[illegible]	[illegible]
5. SAFETY INJECTION SYSTEM	[illegible] SHT.[illegible]	[illegible]
6. CLEAN RADIOACTIVE WASTE TREATMENT SYSTEM	[illegible]	[illegible]
7. RADIOACTIVE GASEOUS WASTE SYSTEM	[illegible]	[illegible]
8. INSTRUMENT AND SERVICE AIR SYSTEM	[illegible] SHT.[illegible] [illegible] SHT.[illegible] [illegible] SHT.[illegible]	[illegible] [illegible] [illegible]
9. PRIMARY MAKE-UP WATER SYSTEM	[illegible]	[illegible]
10. PROCESS SAMPLING SYSTEMS	[illegible] SHT.[illegible]	[illegible]
11. MISCELLANEOUS GAS SUPPLY SYSTEMS	[illegible]	[illegible]





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Response to #2

4.9.5.6 YT-471A/B, YT-472A/B, YT-473A/B, and YT-474A/B Continuous Vibration Monitors

Each RCP is equipped with two vibration pickups mounted at the top of the motor support stand to measure radial vibrations of the pump. The signals are taken to a multi-point selector switch mounted outside the reactor Containment. A portable vibration meter is connected to this selector switch, and so the signal from any one pickup may be monitored at one time. It is recommended that the vibration levels of the RCPs be monitored periodically or whenever an abnormal condition is suspected. There is no requirement for continuous monitoring or recording.

For further details about the continuous vibration monitors, see Subsection 4.3.2.

4.10 OVERPRESSURE MITIGATING SYSTEM

Section 4.10 provides the design bases and configuration for the Overpressure Mitigating System (OMS). This information was obtained from the following correspondence:

(1) Westinghouse Letter to PGE, POR-87-506, Low Temperature Overpressure Protection System (LTOPS) Reanalysis, January 23, 1987.

(2) Westinghouse Letter to PGE, POR-87-516, Low Temperature Overpressure Protection System (LTOPS) Setpoint Analysis Under High Mass Injection Conditions, February 12, 1987.

(4) PGE Letter to the Nuclear Regulatory Commission (NRC), C. Goodwin Jr. to A. Schwencer, Overpressure Mitigating System, April 8, 1977.

(3) PGE Letter to the NRC, C. Goodwin Jr. to A. Schwencer, July 21, 1977.

4.10.1 DESIGN REQUIREMENTS

The OMS must be designed to prevent the RCS pressure from exceeding the limits stated in 10 CFR 50 Appendix G. It also must be designed to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) 279, as follows:

(1) Separate power sources must be provided for each low-pressure circuit.

(2) The low-pressure circuits must be independent and redundant, to satisfy the single-failure criteria.

(3) Equipment quality and qualification must be consistent with that of the original RCS components.

(4) System inputs must be derived from signals that are a direct measure of the desired variable.

(5) The system must be capable of being tested and calibrated.

Three Trojan specific mass input overpressure analyses were performed by means of the RETRAN code to determine the specific PORV setpoints and design criteria for the OMS (see letter mentioned above, PGE to the NRC, July 21, 1977). A Trojan SI pump startup incident was analyzed to determine the maximum predicted RCS pressure overshoot for mass input incidents, since this pump has the highest flow delivery. An incident of a Trojan centrifugal pump charging with letdown isolation was analyzed to determine the criteria for PORV cycling that was used in the design of the air accumulators. This pump has the highest flow delivery for situations in which a loss of instrument air causes both letdown isolation and overpressure. The Trojan positive displacement pump charging with letdown isolation was analyzed to determine the maximum predicted RCS pressure undershoot for mass input incidents, since this pump has the lowest flow delivery.

The two Trojan PORVs were determined to have different opening and closing characteristics. PCV-456 is considered the fastest acting of the two PORVs; conversely, PCV-455A is considered the slowest acting. In the three RETRAN analyses, the appropriate PORV was modeled to find out which would yield the most conservative results. The analyses determined that the setpoint for PCV-455A (slow PORV) should be 440 psig, and that the setpoint for PCV-456 (fast PORV) should be 490 psig.

For the Trojan centrifugal pump charging with letdown isolation incident, the fastest acting PORV (PCV-456) was assumed to be actuated. This action results in a minimum PORV cycle time of 29.1 seconds, and this information forms the basis for sizing the air accumulators in the OMS. The air accumulators were designed to assure that 32 PORV cycles can be accomplished during the 10 minutes during which credit for operator action cannot be taken. Calculation TM-011 has determined that a minimum volume of 7.52 ft^3 is necessary to meet these requirements.

4.10.2 CONFIGURATION

The OMS is designed as two independent, redundant, channels that actuate the relief valves. Each channel consists of the following:

(1) A pressure transmitter

(2) A pressure alarm bistable and annunciator

(3) A pressure actuating bistable

(4) A mode selector switch and indication lights

(5) A circuitry activation annunciator

(6) Valve actuation circuitry

(7) A relief valve

Seismic Class I air accumulators have been installed for each PORV operator to ensure that the PORVs have an air supply for proper operation after a seismic event. The accumulators were fabricated from 4-ft sections of 12-in. Schedule 40S pipe with two 12-in. pipe caps. Thus, two accumulators (3.93 ft^3 each) are used for each PORV, and the total volume provided is 7.68 ft^3.

Response to #3 See Step 12

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	NOTE Steps 1 through 11 are IMMEDIATE ACTION steps.	
	IMMEDIATE ACTION STEPS	
1	Verify Reactor Trip: • Reactor trip and bypass breakers - OPEN • Rod bottom lights - LIT • Neutron flux - DECREASING	Manually trip reactor. IF reactor is NOT subcritical, THEN go to FR-S.1, RESPONSE TO POWER GENERATION/ATWS, Step 1.
2	Verify Turbine Trip:	
	a. Verify the following valves - CLOSED: • Stop valves • Control valves • Intercept valves • Reheat stop valves	a. Manually trip the turbine.
	b. Bleeder trip valves - CLOSED	b. Manually close valves.
	c. Extraction line drains - OPEN	c. Manually open drains.
3	Verify Generator Trip:	
	a. Generator output breakers - OPEN	a. Manually open output breakers 30 seconds after turbine trip.
	b. Generator exciter field breaker - OPEN	b. Manually open breaker.

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	IMMEDIATE ACTION STEPS	
4	Verify Power to AC Emergency Buses:	
	a. A1 or A2 emergency buses - AT LEAST ONE ENERGIZED	a. Try to restore power to at least one emergency bus by manually starting and loading the EDG from the control room. IF power CANNOT be restored to at least one emergency bus, THEN go to ECA-0.0, LOSS OF ALL AC POWER, Step 1.
	b. A1 and A2 emergency buses - BOTH ENERGIZED	b. Try to restore power to deenergized bus per ONI-32, 12.47-kV and 4.16-kV System Faults, while continuing with this procedure.
5	Check If SI Is Actuated:	Check if SI is required. IF SI is required, THEN manually actuate. IF SI is NOT required, THEN go to ES-0.1, REACTOR TRIP RESPONSE, Step 1.
	a. Sequencer tree lights - LIT	a. Manually start applicable equipment.
	b. SI status lights - LIT	b. Manually align applicable equipment.

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	<u>IMMEDIATE ACTION STEPS</u>	
6	<u>Verify Containment Isolation, Phase A</u>:	
	a. Containment Phase 'A' isolation status lights - LIT	a. Manually initiate CIS. Manually close valves that DO NOT have status light lit.
7	<u>Verify Feedwater Isolation</u>: • FWIVs - CLOSED • FWIV bypasses - CLOSED • FWRVs - CLOSED • FWRV bypasses - CLOSED	Manually close valves.
8	<u>Verify AFW Pumps Running</u>:	
	a. Turbine-driven AFP - STARTED: • Steam supply valves - OPEN • Steam stop valve - OPEN • Trip and throttle valve - OPEN	a. Manually open valves.
	b. Diesel AFP - STARTED	b. Manually start pump.

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	IMMEDIATE ACTION STEPS	
9	Verify ECCS Flows:	
	a. CCP BIT flow - FLOW on FI-917	a. Manually start pumps and align valves.
	b. RCS pressure - < 1,520 psig (< 1,800 psig for adverse containment)	b. Go to Step 10.
	c. SI FLOW on FI-918 and FI-922	c. Manually start pumps and align valves.
	d. RCS pressure - < 200 psig (< 480 psig for adverse containment)	d. Go to Step 10.
	e. RHR FLOW on FI-970 and FI-971	e. Manually start pumps and align valves.
10	Check If Main Steamlines Should Be Isolated	
	a. Steam flow - HIGH COINCIDENT WITH: • Lo steam pressure - OR - • Lo-Lo TAVE	a. Go to Step 11.
	b. Verify MSIVs and bypasses - CLOSED	b. Manually close MSIVs and bypasses.

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	IMMEDIATE ACTION STEPS	
11	Check Containment Pressure:	
	a. Containment pressure - HAS EXCEEDED 30 psig	a. Go to Step 12.
	b. MSIVs and bypasses - CLOSED	b. Manually close MSIVs and bypasses.
	c. Containment spray eductor flow - FLOW on FI-2079A and FI-2079B	c. Perform the following: 1) Manually initiate a containment spray signal. 2) Manually start the containment spray pumps. 3) IF no flow on FI-2079A and FI-2079B, THEN manually align valves.
	d. CIS phase B valves - CLOSED • MO-3294 • MO-3296 • MO-3300 • MO-3320	d. Manually close Phase 'B' isolation valves.
	e. Stop all RCPs	
	END OF IMMEDIATE ACTION STEPS	

REACTOR TRIP, SAFETY INJECTION, AND DIAGNOSIS

Step	Action/Expected Response	Response Not Obtained
	NOTE RERP actions should be performed in parallel with this procedure.	
12	Verify total AFW flow > 495 gpm	IF S/G NR level > 5% (> 12% for adverse containment) in any S/G, THEN control feed flow to maintain NR level. IF NR level < 5% (< 12% for adverse containment) in all S/Gs, THEN manually start pumps and align valves as necessary. IF AFW flow > 495 gpm CANNOT be established, THEN go to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, step 1.
13	Verify RCS Heat Removal:	
	a. TAVE - DECREASING to 557°F	a. IF temperature > 557°F, THEN • Dump steam to condenser - OR - • Dump steam using S/G PORVs.
	b. TAVE - STABILIZES at 557°F	b. IF cooldown continues THEN: 1) Verify steam dumps and S/G PORVs CLOSED 2) Maintain total feed flow > 495 gpm until NR level > 5% (> 12% for adverse containment) 3) IF cooldown continues, THEN close MSIVs and bypasses.

Response to #3 See Step 14

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
	CAUTION If total feed flow is < 720 gpm due to operator action, this procedure should NOT be performed.	
	CAUTION If WR level in any 3 S/Gs < 25% (< 38% for adverse containment) <u>OR</u> PZR pressure > 2,335 psig due to loss of secondary heat sink, RCPs should be tripped and steps 10 through 15 should be immediately initiated for bleed and feed.	
	CAUTION Feed flow should not be established to any faulted S/C if an intact S/C is available.	
1	<u>Check If Secondary Heat Sink Is Required</u>:	
	a. RCS pressure - > ANY INTACT GENERATOR PRESSURE	a. Go to EI-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. RCS temperature - > 350°F	b. Try to place RHR system in service per OI-4-1, Residual Heat Removal, while continuing with this procedure. IF adequate cooling with RHR is established, THEN return to the procedure and step in effect.

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
	NOTE Refer to ONI-55, Operation of Electric AFP Supplied by EDG, for the steps necessary to start the electric AFP on an EDG if offsite power is not available	
2	Try to Establish AFW Flow to at Least One S/G:	
	a. Check control room indications for cause of AFW failure: • CST level • Turbine AFW pump steam supplies stop valve, trip and throttle valve • Diesel AFW pump fuel oil supply • AFW system valve alignment.	
	b. Try to restore AFW flow: • Start the diesel AFW pump • Start the turbine AFW pump	b. Start the electric AFW pump.
	c. Check total flow to S/G - > 720 gpm	c. Dispatch operator to locally restore AFW pumps and valve alignments. Go to Step 3.
	d. Return to procedure and step in effect	
3	Stop All RCPs	
4	Check CCP Status - AT LEAST ONE AVAILABLE	Go to Step 10

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
5	Try to Establish Main FW Flow To At Least One S/G:	
	a. Check condensate system - IN SERVICE	a. Try to place condensate system in service per OI-8-1, Condensate and Feedwater System. IF NOT, THEN go to Step 9.
	b. OPEN FWIVs to the selected S/G:	b. Open FWIVs and FRVs to the selected S/G per Appendix A.
	• 1) Rack out the reactor trip breakers and their bypass breakers	IF no FRV and FWIV can be opened, THEN go to Step 9.
	2) Reset SI	
	3) Reset FWIS	
	4) Open the FWIV, and FRV; or their bypasses to the selected S/G	
	- OR -	
	• 1) Locally override the applicable FRV open with the manual handwheel	
	2) OPEN and HOLD the applicable FWIV switch in OPEN at the FWI control panel	
	c. Defeat the MFP trip signals:	c. Lift leads specified in Appendix B.
	1) Rack out the reactor trip breakers and their bypasses	
	2) Reset SIS	
	d. Establish MFW flow per OI-8-1, Condensate and Feedwater	c. Go to Step 7.

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
6	Check S/G Levels	
	a. NR level in at least one S/G > 5% (> 12% for adverse containment)	a. IF feed flow to at least one S/G verified, THEN maintain flow to restore NR level to > 5% (> 12% for adverse containment). IF NOT verified, THEN go to Step 7.
	b. Return to procedure and step in effect	

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
	CAUTION The following step calls for depressurizing at least one S/G. Since steamline delta-P SI signals CANNOT be blocked, automatic initiation may result. SI and CIS should be reset as soon as possible and charging, letdown and containment instrument air restored to normal lineup while performing this step	
	CAUTION Following block of automatic SI signals, or SI reset following automatic actuation, manual SI actuation will be required if conditions degrade	
7	<u>Try To Establish Feed Flow From The Condensate System</u>:	
	a. Depressurize RCS to < 1,865 psig:	
	1) Check letdown - IN SERVICE	1) Use one PZR PORV. IF PZR PORV CANNOT be used, THEN use auxiliary spray. Go to Step 7.b.
	2) Use auxiliary spray	2) Use one PZR PORV.
	b. Block SI signals which are blockable:	
	• High steam flow coincident with low steam pressure or low T_{AVE}	
	• Low PZR pressure	
	Continued next page	

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
7	Continued	
	c. Depressurize at least one S/G to < 400 psig:	
	1) Close selected S/G MSIV and MSIV bypass	
	2) Open the steamline PORV to the selected S/G and depressurize the S/G to < 400 psig	2) Go to Step 9.
	d. OPEN FWIVs and FWRVs to the selected S/G:	d. Open FWIVs and FRVs to the selected S/G per Appendix A.
	• 1) Rack out the reactor trip breakers and their bypass breakers	
	2) Reset SI	
	3) Reset FWIS	
	4) Open the FWIV and FRV, or their bypasses to the selected S/G	
	- OR	
	• 1) Locally override the applicable FRV open with the manual handwheel	
	2) OPEN and HOLD the applicable FWIV switch in open at the FWIV control panel	
	e. Establish a valve lineup to supply condensate flow to the selected S/G	e. Go to Step 9.
	f. Operate the selected S/G PORV and regulate the condensate flow to control RCS temperature until another means of cooling is available	f. Go to Step 9.
	g. Establish makeup to the hotwell from the CST (CST level > 9%)	g. IF level in the CST drops to 9%, THEN makeup to the hotwell using fire main supply through an open manway.

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
8	Check S/G Levels: a. NR level in at least one S/G - > 5% (> 12% for adverse containment)	a. IF feed flow to at least one S/G verified, THEN maintain flow to restore NR level > 5% (> 12% for adverse containment). IF NOT verified, THEN go to Step 9.
	b. Return to procedure and step in effect	
9	Check for Loss of Secondary Heat Sink: • WR S/G level - < 25% in any 3 S/Gs (< 38% for adverse containment) - OR - • PZR pressure - > 2,335 psig	Return to Step 1.

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
	CAUTION Steps 10 through 15 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.	
10	Actuate SI	
11	Verify ECCS Flow: a. Check flow indication: • CCP BIT flow - FLOW indicated on FI-917 - OR - • SI pumps 1) SI pumps - AT LEAST ONE RUNNING 2) RCS pressure - < 1,520 psig (< 1,800 psig for adverse containment) 3) Verify SI flow - FLOW indicated on FI-918 and FI-922	a. Manually start pumps and align valves as necessary to establish feed path. IF a feed path CANNOT be established, THEN continue attempts to establish feed flow. Return to Step 5.
12	Reset SI and CIS	

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
13	Establish Instrument Air to Containment:	
	a. Verify at least one instrument A/C - RUNNING	a. IF off-site power is available, THEN locally start one instrument A/C per OI-7-5, Instrument and Service Air. IF off-site power or bearing cooling water system are NOT available, THEN start the B Joy A/C as follows: 1) Attach hose to drain between sprinkler valves FP-043 and FP-044. Attach other end of hose to B Joy A/C BCW drain valve BC-042. 2) Close BCW supply valves BC-033 and BC-039 and return valve BC-045 on B Joy A/C. 3) Open BCW drain to floor by BC-045, then open fire maindrain and BCW supply BC-042. 4) Place B Joy A/C controller in off. 5) Reset lockout relays on the LCCs: • B6-B01 • B6-B02 • B6-B03 • B6-B04 6) Place B Joy A/C controllers in constant and select lead No. 2.
	b. Open - CV-4471, instrument air to containment isolation	

RESPONSE TO LOSS OF SECONDARY HEAT SINK

Step	Action/Expected Response	Response Not Obtained
	CAUTION The following step may result in the rupture of the PRT rupture disc. Abnormal containment conditions indicating a possible RCS leak or LOCA may occur due to a loss of PRT integrity.	
14	<u>Establish RCS Bleed Path</u>:	
	a. Verify PZR PORV block valves - ALL OPEN	a. Open block valves.
	b. Open ALL PZR PORVs	
15	<u>Verify Adequate RCS Bleed Path</u>:	
	a. PZR PORVs - BOTH OPEN	a. Perform the following: 1) Open reactor vessel head vents: • SV-1015A • SV-1015B • SV-1016A • SV-1016B 2) Depressurize at least one intact S/G to atmosphere using S/G PORV. 3) Align any available low pressure water source to the depressurized S/G.

Response to #4

TABLE 15.8-1

ATWT ANALYSIS MODEL SUMMARIES

Parameter	WCAP-8330 Model	Trojan Nuclear Plant	
Number of loops	4	4	
Core power, MWt	3,411	3,411	
Nominal pressurizer pressure	2,250	2,250	
Nominal coolant flow, gpm	354,000	354,000	
Nominal average coolant temperature, °F	584.65	584.7[a]	(3)
Nominal coolant no-load temperature, °F	557	557	
Total RCS volume including pressurizer, ft^3	12,600[b]	12,527[b,c]	(3)
Pressurizer volume, ft^3	1,843.7[b]	1,800	(3)
Steam capacity of power-operated relief valves, lb/hr, 2 @ 2,350 psia	210,000	210,000	
Steam capacity of safety valves, lb/hr	420,000[d]	435,120[e]	(3)
Best estimate rod worth of bank D at its full power insertion limit, %Δk/k	0.3	0.3	
Steam generator design pressure	1,200	1,200	
Steam generator nominal steam temperature, °F	533.3	533.3	
Nominal steam flow, lb/sec	4,192	4,186.1	
Nominal fluid mass in steam generator, lb	406,400	406,400	
Auxiliary feedwater temperature, °F	>130	>130	
Auxiliary feedwater available, gal	170,000	>190,000	
Capacity of auxiliary feedwater, gpm	1,760	>1,760	
Volume of line between auxiliary feedwater connection on feedline and steam generator inlet, total for all loops, ft^3	500	>500	

[a] Nominal T_{ave} at 100% power.

[b] Includes surge line.

[c] Does not include 3% thermal expansion factor. The RCS volume used in the large break LOCA analysis was 12,476 cubic feet. (3)

[d] Three at 2,500 psia.

[e] Three at 2,590 psia.

Trojan Nuclear Plant
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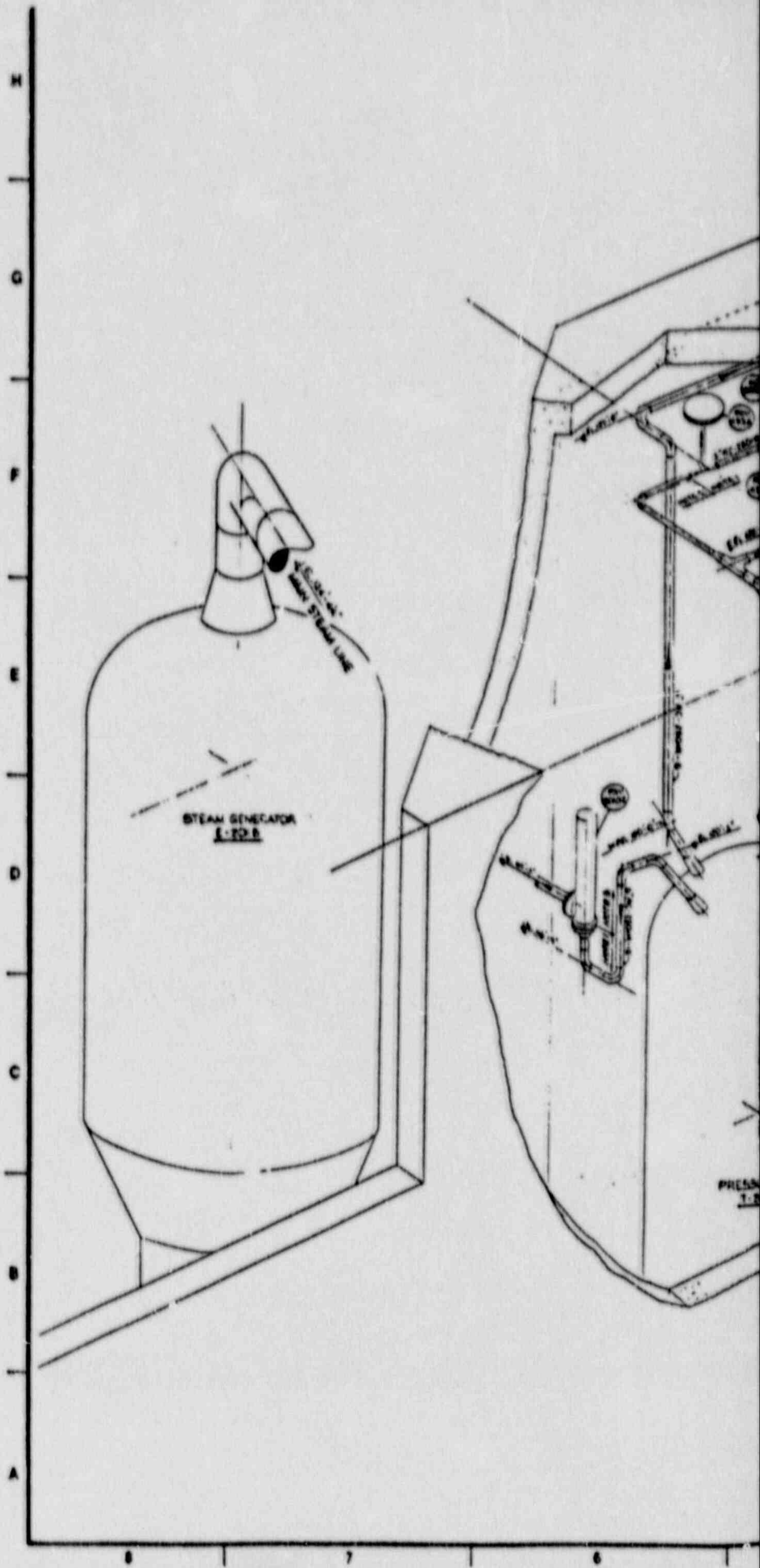
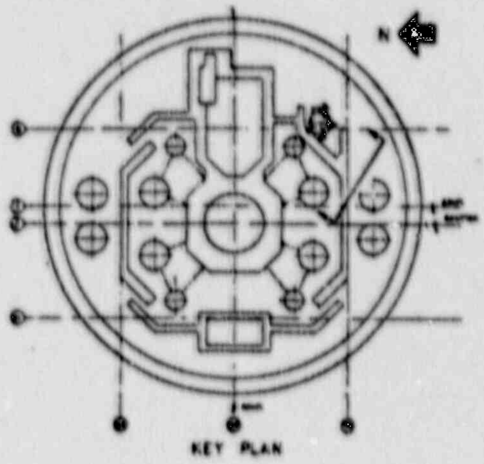
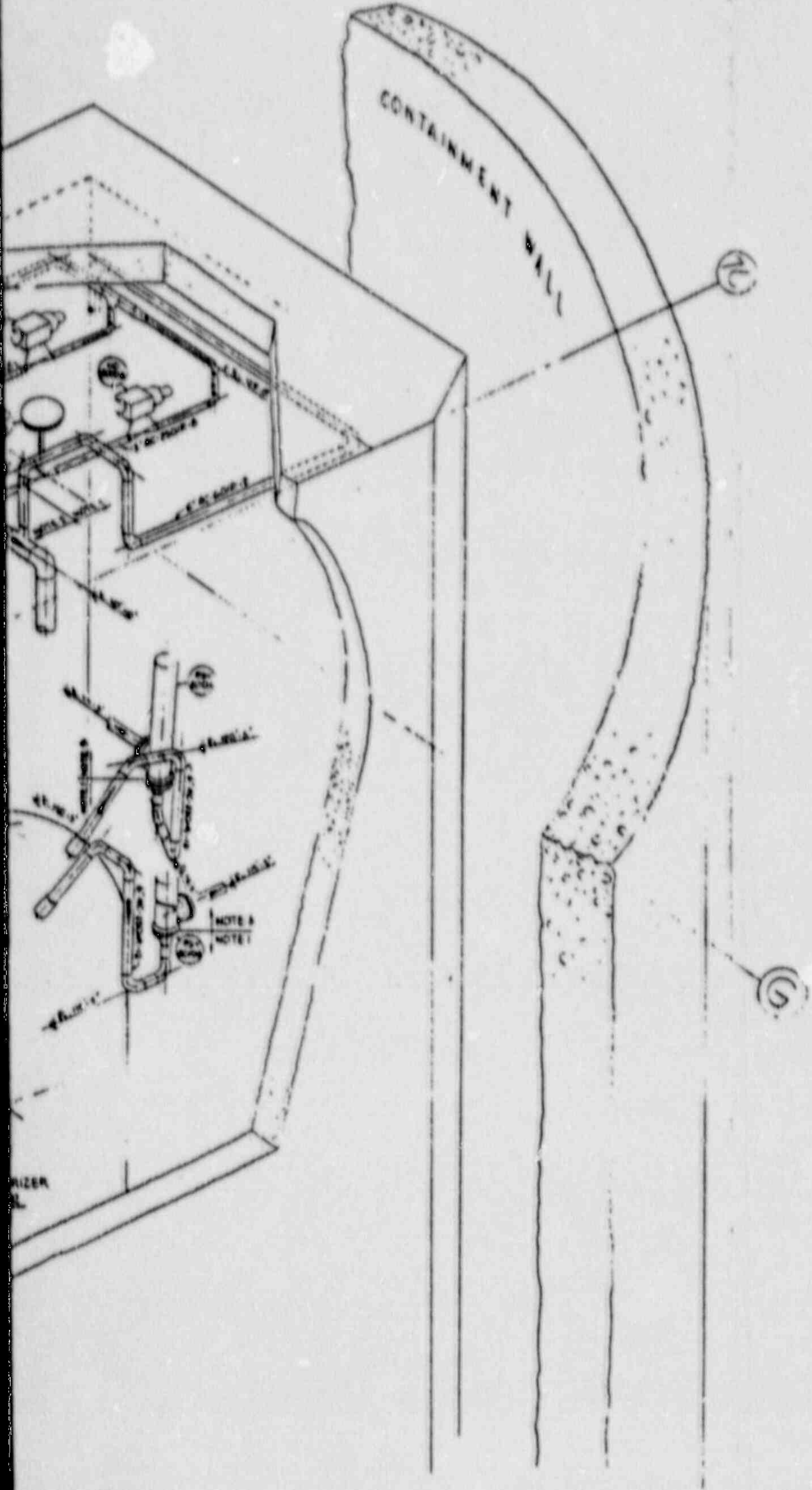
Information for #5







NOTES

1. THE CRITERIA OUTLINED IN SECTION 3.6.2.1.1, LARGE LINE RUPTURE RESULTING IN LOSS OF COOLANT, WAS USED TO ESTABLISH THE PROTECTION CRITERIA FOR THIS SECTION OF PIPE.
2. NO RESTRAINTS ARE NECESSARY BEYOND VALVES [illegible] AND [illegible]. SEE FIGURE 3.6-1, CASE 1.
3. NO RESTRAINTS ARE NECESSARY BEYOND VALVE [illegible]. SEE FIGURE 3.6-1 CASE 2.
4. FIGURE 3.6-22 IS ADAPTED FROM ISOMETRIC DWGS RC-[illegible], RC-2501R-8-1, RC-2501R-8-2, RC-2501R-12-1.

Amendment 11
(September 1989)

TROJAN NUCLEAR PLANT
FSAR
Pressurizer Safety and Relief Lines (RC-2501R-8 and RC 2501R-12)
FIGURE 3.6-22

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