August 16, 1990

Docket No. 50-352

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Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: REPLACEMENT OF HCU ISOLATION BOUNDARIES (TSCR NO. 90-01-1), LIMERICK GENERATING STATION, UNIT 1 (TAC NO. 76960)

The Commission has issued the enclosed Amendment No. 42 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 14, 1990.

This amendment revises the TSs to add new isolation valves on each common Control Rod Drive header to the table of primary containment isolation valves that must be operable and to delete the existing individual Hydraulic Control Unit isolation valves from the TSs.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 42 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

CP1

OFC	:PDI-2/LA	:PDI-2/PE	:PDI-2/PM	:OGC	:PDI-2/D	SRXB
NAME	:MO'Brien	:SDembek	:RClark	: E Holler	:WButler	: RJones
DATE	: 7/10/90	: 7/10/90	:07/09/90	: 7/16/90	:8/14/90	: 8/1/90

SPLB
CMcCracken
8/1/90

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 16, 1990

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Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
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2. Safety Evaluation

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See next page

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Philadelphia Electric Company

Limerick Generating Station
Units 1 & 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-39

1. The Nuclear Regulatory Commission (the Commission) has found that

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated June 14, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 42 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective September 30, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

PDI-2/LA	PDI-2/PE	PDI-2/PM	OGC	PDI-2/D	SRXB/C
MO'Brien	SDembek	RClark	EHoller	WButler	RJones
7/10/90	7/10/90	07/09/90	7/14/90	8/15/90	8/1/90
					SPLB
					CMcCracken
					8/10/90

3. This license amendment is effective September 30, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 6-23	3/4 6-23
3/4 6-24	3/4 6-24*
3/4 6-41	3/4 6-41
3/4 6-42	3/4 6-42*

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
028B	DRYWELL H2/02 SAMPLE	SV57-133		5	B,H,R,S	11	57
			SV57-143	5	B,H,R,S	11	
			SV57-195	5	B,H,R,S	11	
030B-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147A	45		10	42
035B	TIP PURGE	59-1056(CK) (DOUBLE "O" RING)		NA			59
			HV59-131	7	B,H,S	16	
035C-G	TIP DRIVES	XV59-141A-E (DOUBLE "O" RING)		NA	B,H	11,16,21	59
			XV59-140A-E	NA		11,16	
037A-D	CRD INSERT LINES	BALL CHECK		NA		12	47
			46-1101	NA		12,22	
			46-1102	NA		12,22	
			46-1108	NA		12,22	
			46-1109	NA		12,22	
038A-D	CRD WITHDRAW LINES SDV VENTS & DRAINS		46-1115	NA		12,22	47
			46-1116	NA		12,22	
			46-1122	NA		12,22	
			46-1123	NA		12,22	
			XV47-1F010	25		30	
			XV47-1F180	30		30	
			XV47-1F011	25		30	
			XV47-1F181	30		30	
039A(B)	DRYWELL SPRAY	HV51-1F021A(B)		160		4,11	51
			HV51-1F016A(B)	160		11	
040E	DRYWELL PRESSURE INSTRUMENTATION		HV42-147D	45		10	42
040F-2	CONTAINMENT INSTRUMENT GAS -SUCTION	HV59-101		45	C,H,S	5	59
			HV59-102	7	C,H,S		

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX. ISOL. TIME. IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040G-1	ILRT DATA ACQUISITION	60-1057		NA		11	60
			60-1058	NA		11	
040G-2	ILRT DATA ACQUISITION	60-1071		NA		11	60
			60-1070	NA		11	
040H-1	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'A'	59-1005A(CK)		NA			59
			HV59-129A	7	C,H,S		
042	STANDBY LIQUID CONTROL	48-1F007(CK) (X-116)		NA			48
			HV48-1F006A	60		29	
043B	MAIN STEAM SAMPLE	HV41-1F084		10	B,D		41
			HV41-1F085	10	B,D		
044	RWCU ALTERNATE RETURN	41-1017		NA		5,31	41
			41-1016(X-9A, X-9B)	NA			
			PSV41-112	NA			
045A(B,C,D)	LPCI INJECTION 'A'(B,C,D)	HV51-1F041A(B,C, D)(CK)	NA		9,22	51	
		HV51-142A(B,C, D)		7		9,22	
			HV51-1F017A (B,C,D)	38			
050A-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147B	45		10	42
053	DRYWELL CHILLED WATER SUPPLY - LOOP 'A'	HV87-128		60	C,H	11	87
			HV87-120A	60	C,H	11	
			''V87-125A	60	C,H	11	

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES

1. Instrumentation line isolation provisions consist of an orifice and excess flow-check valve or remote manual isolation valve. The excess flow-check valve is subjected to operability testing, but no Type C test is performed or required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

2. Penetration is sealed by a blind flange or door with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings.

3. Inboard butterfly valve tested in the reverse direction.

4. Inboard gate valve tested in the reverse direction.

5. Inboard globe valve tested in the reverse direction.

6. The MSIVs and this penetration are tested by pressurizing between the valves. Testing of the inboard valve in the reverse direction tends to unseat the valve and is therefore conservative. The valves are Type C tested at a test pressure of 22 psig.

7. Gate valve tested in the reverse direction.

8. Electrical penetrations are tested by pressurizing between the seals.

9. The isolation provisions for this penetration consist of two isolation valves and a closed system outside containment. Because a water seal is maintained in these lines by the safeguard piping fill system, the inboard valve may be tested with water. The outboard valve will be pneumatically tested.

10. The valve does not receive an isolation signal but remains open to measure containment conditions post-LOCA. Leaktightness of the penetration is verified during the Type A test. Type C test is not required.

11. All isolation barriers are located outside containment.

12. Leakage monitoring of the control rod drive insert and withdraw line is provided by Type A leakage rate test. The outboard isolation provisions for the control rod insert and withdraw lines consists of two redundant Type C tested simple check valves located on each main water header (i.e., charging, cooling, drive and exhaust). Type C test is not required for the ball check valve.

13. The motor operators on HV-13-109 and HV-13-110 are not connected to any power supply.

14. Valve is provided with a separate testable seal assembly, with double concentric O-ring seals installed between the pipe flange and valve flange facing primary containment. Leakage through these seals is included within the Type C leakage rate for this penetration.

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

15. Check valve used instead of flow orifice.

16. Penetration is sealed by a flange with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings. Both the TIP Purge Supply (Penetration 35B) and the TIP Drive Tubes (Penetrations 35 C thru G)are welded to their respective flanges. Leakage through these seals is included in the Type C leakage rate total for this penetration. The ball valves (XV-141A thru E) are Type C tested. It is not practicable to leak test the shear valves (XV-140A thru E) because squib firing is required for closure. Shear valves (XV-140A thru E) are normally open.

17. Instrument line isolation provisions consist of an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The excess flow check valves are subject to operability testing, but no Type C test is performed nor required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

18. In addition to double "O" ring seals, this penetration is tested by pressurizing volume between doors per Specification 4.6.1.3.

19. The RHR system safety pressure relief valves which are flanged to facilitate removal will be equipped with double O-ring seal assemblies on the flange closest to primary containment. These seals will be leak rate tested by pressurizing between the O-rings, and the results added into the Type C total for this penetration.

20. See Specification 3.3.2, Table 3.3.2-1, for a description of the PCRVICS isolation signal(s) that initiate closure of each automatic isolation valve. In addition, the following non-PCRVICS isolation signals also initiate closure of selected valves:

    EA Main steam line high pressure, high steam line leakage flow, low MSIV-LCS dilution air flow

    LFHP With HPCI pumps running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFRC With RCIC pump running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFCH With CSS pump running, opens on low flow in associated pipe, closes when flow is above setpoint

    LFCC Steam supply valve fully closed or RCIC turbine stop valve fully closed

All power operated isolation valves may be opened or closed remote manually.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-39

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated June 14, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. The proposed amendment would revise the Technical Specifications (TSs) to add new isolation valves on each common Control Rod Drive header to the table of primary containment isolation valves that must be operable and to delete the existing individual Hydraulic Control Unit isolation valves from the TSs. Limerick, Unit 1, is scheduled to shutdown for the third refueling outage on September 8, 1990. During the refueling outage, the licensee plans to install eight new check valves (four pairs of valves) on the control rod drive (CRD) supply headers to the hydraulic control units (HCUs). These new valves will constitute a new isolation boundary for the Integrated Leak Rate Test (ILRT), replacing the existing HCU isolation boundary valves. The proposed changes to the TS are to include the new valves in Table 3.6.3-1, "Part A - Primary Containment Isolation Valves" and to remove the current numbers for the HCU boundaries.

2.0 DISCUSSION

The Integrated Leak Rate Test (ILRT) is a pressurization of primary containment and measurement of total leakage from all isolation boundaries. The current method of testing the isolation boundaries of the CRD system is to collect leakage through the HCUs at the vent valves on each of four supply headers during an ILRT. If the total leakage exceeds specified limits, approximately 1300 individual check valves or solenoid valves must be examined to find and repair leak paths.

To minimize critical path outage time, new check valves will be installed in the CRD supply headers in four locations, effectively extending the isolation boundary from the HCUs to these new valves. These eight new valves (4 pairs) will reduce the number of testable CRD penetrations from approximately 1300 to four.

The licensee included a sketch of the proposed arrangement which is enclosed (Figure 1). As shown, the licensee proposes to install two lift check valves, a manually controlled block valve and two test connections on each of the four CRD headers to the HCUs (drive, cooling, charging and exhaust). The general process diagram for the CRD hydraulic system is shown in figure 4.6.7 in the Limerick Final Safety Analysis Report (FSAR) which is also enclosed. The new valves will be installed between the main control station and the existing vent valve shown in Figure 1.

In a BWR 4 such as Limerick, Units 1 and 2, there are 185 control rods, each of which has a HCU. As shown on Figure 4.6.8 in the FSAR (enclosure 3), and in Figure 1, there are 3 check valves and 4 directional control valves on each HCU for a total of 1295 valves. The Integrated Leak Rate Test (ILRT) is a pressurization of primary containment and measurement of total leakage from all isolation boundaries. With the present arrangement, the 1295 valves constitute the isolation boundary. The current method of testing the isolation boundary of the CRD system is to place a bucket under the vent valves shown in Figure 1 and to collect leakage through the HCUs at the vent valves on each of the four supply headers during an ILRT. If the total leakage exceeds specified limits, approximately 1300 individual check valves or solenoid valves must be examined to find and repair leak paths. Experience has shown that the total leakage is due to a drop or two of water leakage from many of the ball check valves rather than failure of one or two valves. To avoid these problems, the licensee proposed to relocate the isolation boundary on the headers to and from the HCUs rather than on the HCUs.

Although a TS change is not required prior to installation of the new valves, a TS change is required to take credit for these new isolation boundaries, and also to remove the current valve numbers for the HCU boundaries from the TS. Therefore, the licensee is proposing that TS Table 3.3-1, "Part A-Primary Containment Isolation Valves," be revised to remove the existing HCU isolation boundary valves and replace them with the newly installed isolation boundary valves. Note 12 of that table also has to be revised to reflect the addition of the new valves. Also since the affected CRD lines are water filled and would remain water filled for a minimum of thirty days after a Loss of Coolant Accident (LOCA), Note 22 applies to these isolation valves.

3.0 EVALUATION

The proposed TS change will take credit for the new valves installed in each of the CRD headers to the HCUs (drive, cooling, charging, exhaust) between the main control station and the vent valve. These valves constitute a new isolation boundary. Each check valve station consists of two check valves, a block valve and two test connections. This enables each check valve to be tested individually instead of during the critical path ILRT. Each check valve station is accessible from an existing platform near the existing vent valves. The change will move the isolation boundary out on the CRD headers.

The licensee has provided analyses to demonstrate that the new design does not change the design criteria previously approved in the staff's Safety Evaluation Report (SER), NUREG-0991, Section 6.2.4.1. The present method of leakage monitoring was accepted by the staff in Section 6.2.6.3 of the SER.

The piping to be included within the new isolation boundary complies with the same standards and specifications as the original boundary. The number of active components making up the boundary will be reduced from approximately 1300 to four. The current CRD isolation boundary includes the insert and withdraw lines, the scram discharge volume and the HCUs. The relocation of the boundary will add some of the supply header piping but will not affect the existing equipment. The added piping is small diameter (2" or less) comparable to the previously analyzed scram discharge drain line. The consequences of a pipe failure inside the isolation boundary remain within the envelope analyzed in NUREG 0803.

An analysis has been performed on the piping being upgraded for inclusion in the extended ILRT boundary. The piping and related pipe supports are designed to meet the criteria of Seismic Category I and ASME Code Section III, Class 2 or 3 as appropriate. Analysis has shown that the existing piping and the modified piping is within the ASME Code allowables. Piping supports have been evaluated and modified as necessary to accommodate the newly analyzed loads.

The licensee has evaluated the hydraulic effect of the new valves. The additional pressure drop will not introduce significant line loss and is well within the CRD pump capacity. The calculations show that there will be adequate flow in each of the headers to meet required design flows. The performance of the CRD system is well within the system capability for normal operation, and control rod scram performance is unaffected.

The testing of the new valves uses techniques and criteria accepted for other similar applications as documented in the Limerick Final Safety Analysis Report (FSAR) Table 6.2-25, "Containment Penetrations - Compliance with 10 CFR, Part 50, Appendix J," Note 14.

The staff has evaluated the potential if the lift check valves fail closed, blocking one of the headers. The new check valves will be similar to the present check valves shown in Figure 1 on the cooling, charging and drive headers, but located farther upstream. The licensee plans to "blue" the seats for a polished surface. Experience has demonstrated that unless the lift mechanism is distorted by temperature, becomes severely corroded or is coated by gummy organics in the line, the valve is highly unlikely to fail closed. In any case, even if one of the check valves were to fail closed, it would not affect the ability to scram the reactor.

As discussed in the FSAR, at system pressure above 600 psig, reactor pressure provides adequate energy to insert the control rods without the assistance of the accummulators. At low or zero reactor pressure, the accumulators provide the energy to scram the rods. A typical curve of scram time vs reactor pressure is shown in enclosure 4. This was verified as part of the Limerick Unit 2 startup test program as shown in enclosure 5 (letter from G.M. Leitch to NRC dated April 2, 1990 transmitting Limerick Unit 2 startup test report.) If a valve failed closed on the charging water header, the pressure in one or more accumulators might bleed down if one could postulate a leakage path. If the pressure decreased to 955 psig, the alarm setpoint, this is annunciated in the control room. When a HCU accumulator alarm condition occurs, the Main Control Room (MCR) reactor operators receive a flashing accumulator trouble alarm indication on the Full Core Display panel in the MCR for the specific HCU (panel *00600). The reactor operator must examine the Full Core Display to identify the specific HCU accumulator that is in alarm. Any alarmed accumulator trouble alarm on the Full Core Display will flash until the reactor operator acknowledges the alarm on a specific accumulator trouble alarm acknowledge button on the reactor console (panel *00603). This alarm condition is accompanied by an audible and flashing annunciator alarm, "Accumulator Trouble," in the MCR. The Reactor Operator must acknowledge the alarm on a general annunciator acknowledge button to silence the alarm noise and stop the flashing alarm window. If a second HCU accumulator alarm is received after the first alarm is acknowledged, the annunciator re-alarms and the operator must again acknowledge the alarm to silence the alarm noise and stop the flashing alarm window. The second HCU accumulator trouble alarm also flashes on the Full Core Display. This sequence is the same for multiple HCU accumulator alarms. Therefore, adequate MCR indication exists for the operator to be alerted to multiple HCU accumulator trouble alarms. If more than one accumulator is inoperable and if reactor pressure is less than 900 psig, the TSs require that the reactor mode switch be placed in the shutdown position. Thus, if a check valve on the charging water line failed closed and this somehow allowed the pressure in one or more HCU accumulators to bleed down, the operators would have ample warning of the condition and the plant would be shutdown while there would be sufficient reactor pressure to scram the rods.

If a check valve failed closed on the cooling water header, the lack of flow would be alarmed in the control room, followed by high temperature alarms. The high temperatures in the drives would require plant shutdown but would not have any effect on scram capability.

The drive and exhaust headers are only used to move control rods. During a scram, the water is discharged to the scram discharge header, not the CRD exhaust line. If either of these lines were blocked by a check valve failing closed, the operators would not be able to perform the weekly exercise test. Every Sunday, the operators move each control rod in and out one notch (6") to demonstrate operability. This would not effect the ability to scram the rods.

In summary, if a check valve on one of the CRD headers failed closed, it would not effect the ability to scram the rods.

The staff has reviewed the licensee's analyses and found them satisfactory. The proposed TS changes to relocate the isolation boundary are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22.(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 26290) on June 27, 1990 and consulted with the State of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

The staff has concluded, based on the consideration discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the healthy and safety of the public.

Principal Contributor: Dick Clark

Dated: August 16, 1990

Enclosure 1

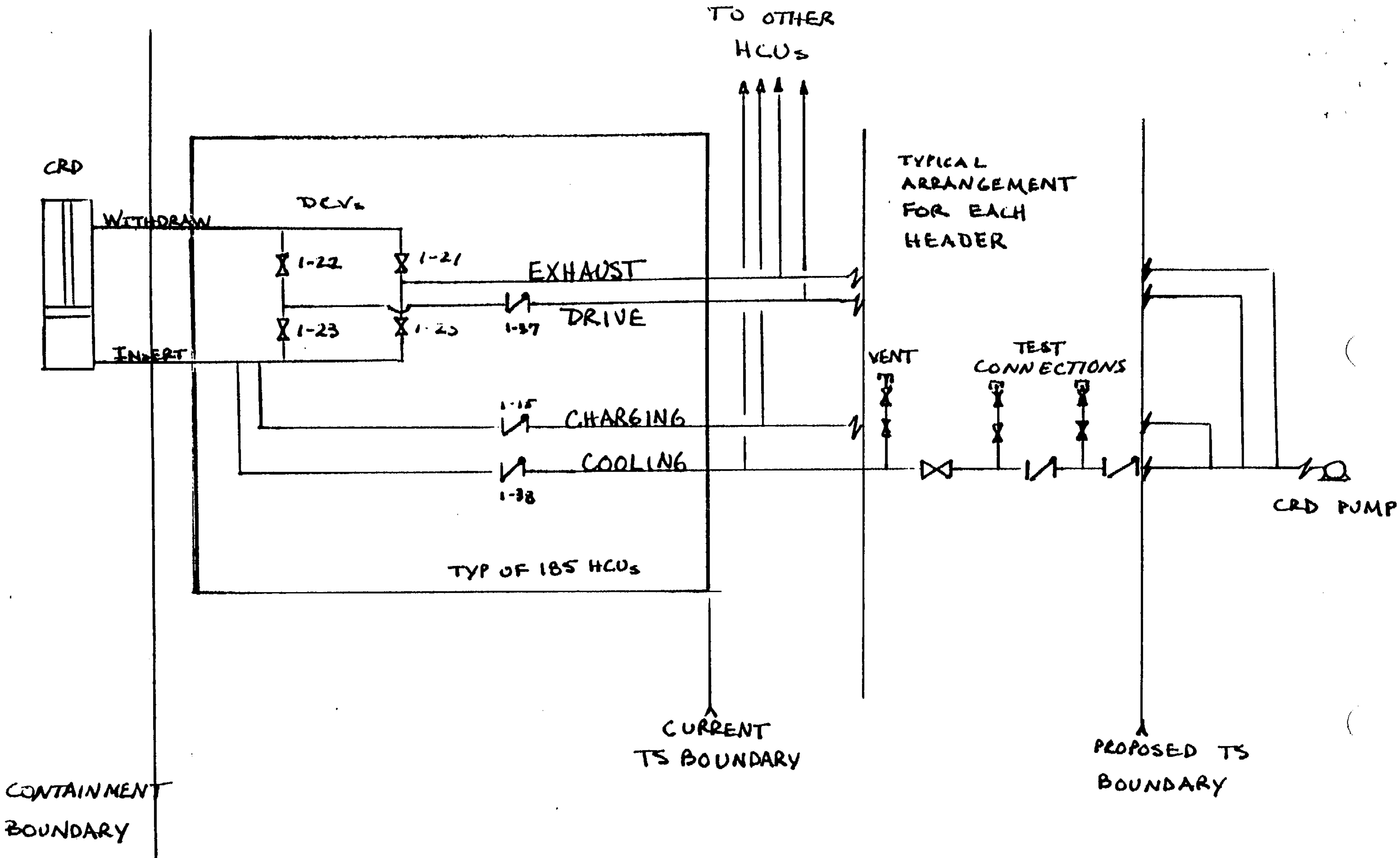


FIGURE 1

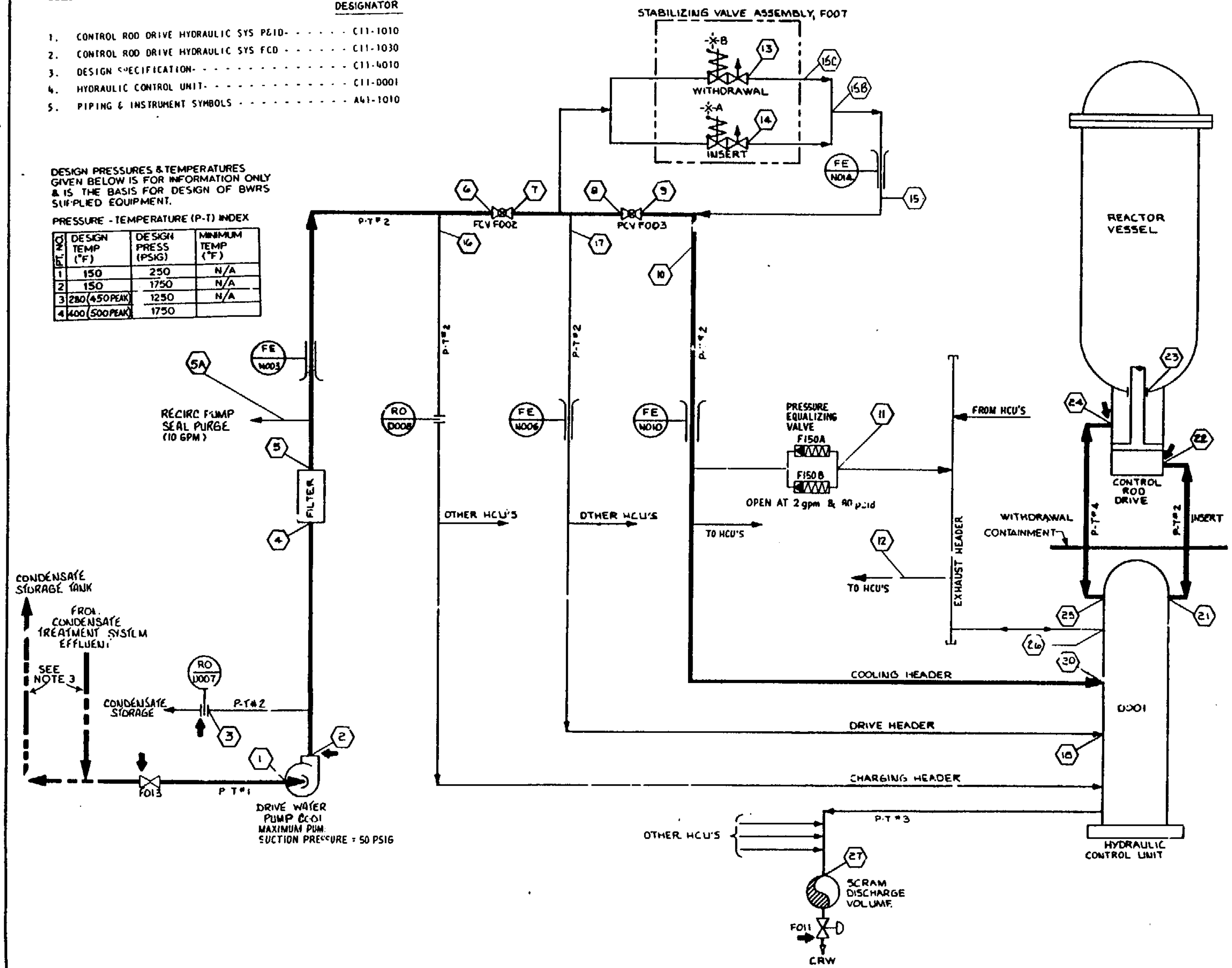
SUPPLEMENTAL DOCUMENTS UNDER THE FOLLOWING IDENTITIES ARE TO BE USED IN CONJUNCTION WITH THIS DRAWING:

		REFERENCE DESIGNATOR
1.	CONTROL ROD DRIVE HYDRAULIC SYS P&ID	C11-1010
2.	CONTROL ROD DRIVE HYDRAULIC SYS FCD	C11-1030
3.	DESIGN SPECIFICATION	C11-4010
4.	HYDRAULIC CONTROL UNIT	C11-D001
5.	PIPING & INSTRUMENT SYMBOLS	A41-1010

DESIGN PRESSURES & TEMPERATURES GIVEN BELOW IS FOR INFORMATION ONLY & IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT.

PRESSURE - TEMPERATURE (P-T) INDEX

PT. NO.	DESIGN TEMP (°F)	DESIGN PRESS (PSIG)	MINIMUM TEMP (°F)
1	150	250	N/A
2	150	1750	N/A
3	280 (450 PEAK)	1250	N/A
4	400 (500 PEAK)	1750	



NOTES:

1. DELETED
2. DELETED
3. THESE LINES SHALL BE SIZED SO THAT A RATE OF 100 GPM MINIMUM CAN BE MAINTAINED TO THE CONDENSATE STORAGE TANK IN ADDITION TO THE CRD SYSTEM NORMAL FLOW REQUIREMENTS SPECIFIED IN THE DATA SHEET FOR MODE A. THIS FLOW SHALL BE MAINTAINED DURING PLANT OPERATION WHEN THE CONDENSATE SYSTEM IS OPERATING.
4. DELETED
5. DELETED
6. DEFINITION OF SYMBOLS
   PR INDICATES PRESSURE OF THE REACTOR
7. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURES WILL NOT EXCEED 150 DEG. F. WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAX. TEMP. (DEG. F.)
MODE A -	23	200
MODE A - (LEAKING SCRAM DISCHARGE VALVE)	23	546
	24	500
	25	500
	27	280
MODE D -	23	475
	24	475
	25	475
	27	450

TEMPERATURES ABOVE 280°F FOR POSITIONS 24, 25 AND 27 MAY BE ASSUMED TO OCCUR LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM.

8. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL. ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 580 PSIG MAXIMUM, AT 70°F.
   B. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR+15 CONDITIONS INDICATED.
   C. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM PER DRIVE.
9. MODE B -
   A. LOCATIONS 13 & 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL, WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR+250 PSIG FOR THE CONDITIONS INDICATED.

105D5584, REV. 1

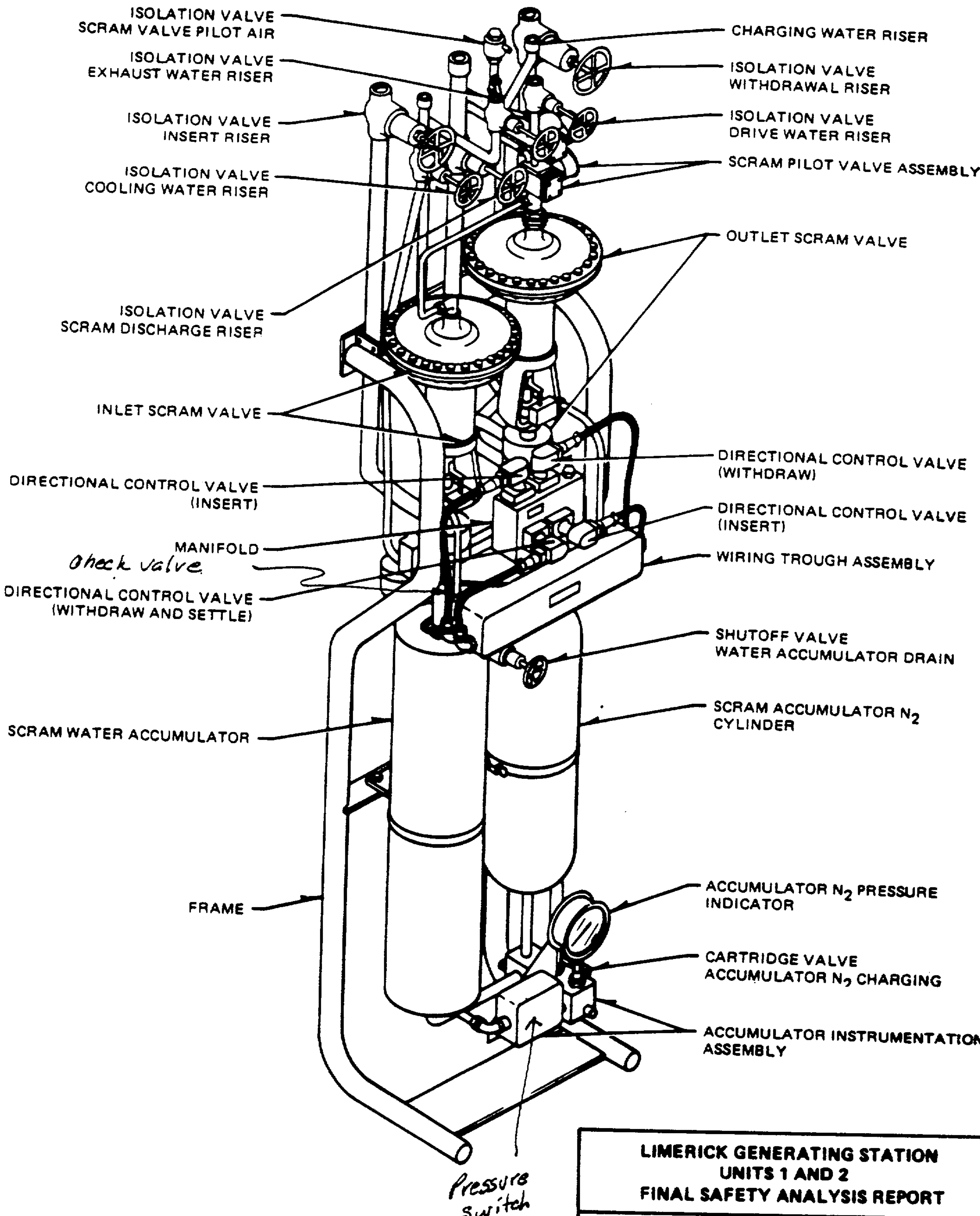
LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD DRIVE HYDRAULIC SYSTEM PROCESS DIAGRAM (SHEET 1 OF 3)

FIGURE 4.6-7

REV. 52, 06/88

Enclosure 2



LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CONTROL ROD DRIVE
HYDRAULIC CONTROL UNIT

FIGURE 4.6-8

Enclosure 3

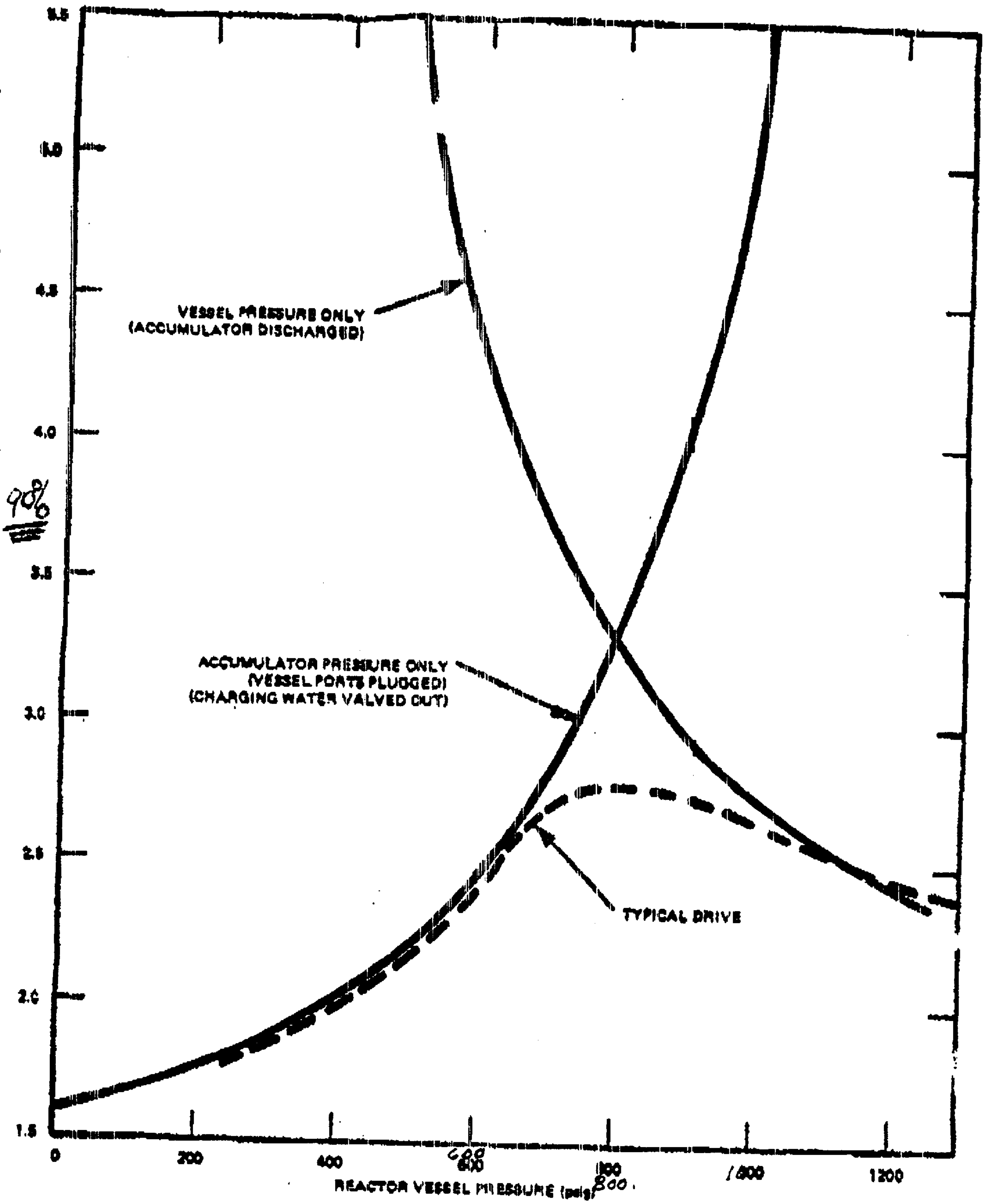


Figure 9

T-47
Enclosure 4

2STP-5.4, Scram Testing of Selected Rods

From the results of previous CRD testing, the four rods with the slowest scram times to position 05 or with unusual operating characteristics-were selected for further testing.

This subtest was performed at the following test conditions: at zero reactor pressure with accumulator pressure just above the low pressure alarm point, at 600 psig reactor pressure with normal accumulator pressure, and at 800 psig reactor pressure with normal accumulator pressure. Each control rod was scrammed three times at every test condition. There were no acceptance criteria verified in this subtest, but each selected rod scram time was verified against Technical Specifications (TS) when performing this subtest during Test Condition Heatup.

The scram times of selected control rods are as follows:

	Measured Time to Position 05 (sec)			
Selected Rod	0 psig (TC OV)	600 psig (TC HU)	800 psig (TC HU)	TS Limit (sec)
22-11	1.78	2.33	2.47	≤ 7
	1.55	2.56	2.71	≤ 7
	1.56	2.32	2.58	≤ 7
26-11	1.65			≤ 7
	1.69			≤ 7
	1.66			≤ 7
18-39	1.71			≤ 7
	1.69			≤ 7
	1.71			≤ 7
26-47	1.65			≤ 7
	1.69			≤ 7
	1.69			≤ 7
26-23		2.65	2.64	≤ 7
		2.52	2.79	≤ 7
		2.53	2.75	≤ 7
30-15		2.36	2.51	≤ 7
		2.34	2.32	≤ 7
		2.38	2.39	≤ 7
34-07		2.32	2.32	≤ 7
		2.30	2.39	≤ 7
		2.28	2.33	≤ 7

Enclosure 5