SNUPPS

Standardized Nuclear Unit
Power Plant System

5 Choke Cherry Road
Rockville, Maryland 20850
(301) 869-8010

Nicholas A. Petrick
Executive Director

September 1, 1981

SLNRC 81-82 FILE: 0541
SUBJ: ICSB Review

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: STN 50-482, STN 50-483, and STN 50-486

Reference: 1. SLNRC 81-68, dated August 14, 1981, Same subject
2. SLNRC 81-67, dated August 14, 1981, ICSB Positions

Dear Mr. Denton:

The referenced letters provided information concerning, and the status of, the agenda items and positions involved in the Instrumentation and Control Systems Branch review of the SNUPPS FSAR. This letter provides the information to close all review matters which have not been closed previously by meetings, correspondence, and FSAR changes.

1. ICSB position #6 concerned reactor coolant temperature indication at the auxiliary shutdown panel. Reference 2 provided a response to the position. Enclosure A to this letter is a revised response to position #6.

2. ICSB position #8 concerned balance-of-plant instrumentation and controls for safety functions. As a clarification to the response to position #8 (see reference 2), the following information is provided. The instrument loops discussed in Table 8-1 can be tested without interfering with normal plant operations and without lifting instrument leads or using jury rigs. For Table 8-1, item D, an alarm will be provided to indicate that the setpoint of the safety function has been reached.

3. Agenda item #45 concerned postulated boron dilution events. The SNUPPS design will incorporate the same design changes that were reviewed by the NRC on the Comanche Peak docket for mitigation of inadvertent boron dilution. These changes include seismic and environmental qualification of the source range and intermediate range nuclear instrumentation.

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4. Agenda item #50 concerned setpoint methodology. Reference 1 provided a response to the item for the balance-of-plant. The following clarification is provided for item 50, part "e". Safety related BOP

instrumentation that could be subject to hostile environments will undergo equipment qualification for that environment. The design goal and the objective of qualification testing is that the instrumentation will remain operable for one year following a design basis event.

5. Agenda item #12 concerned sensors and circuits in non-seismically designed structures. An FSAR change was provided in Reference 1 and Revision 6 to the SNUPPS FSAR. A further clarification concerning cable routing and sensor qualification is provided with the FSAR changes included herein as Enclosure B. Enclosure B will be incorporated in Revision 7 to the SNUPPS FSAR.

6. Agenda items #33 and #35 concerned display instrumentation. Included in the Enclosure B FSAR changes are text and table additions that close these agenda items.

7. Agenda item #39 concerned low temperature overpressure protection. Included in the Enclosure B FSAR changes are revised descriptions of the SNUPPS design. The NRC questioned the failure modes in this system. The pressurizer PORV block valves are normally open and are designed to close upon loss of opening signal. The pressurizer PORVs are normally closed and are designed to utilize energize-to-open logic. The PORVs are designed to close upon loss of energy or loss of opening signal. Therefore, loss of opening signal will result in both PORV and PORV block valve closure, and accidental RCS depressurization is precluded. Since each PORV and its associated block valve are on a separate power supply from the other PORV and its block valve, the pressurizer pressure relief system is designed such that no single failure can prevent opening of at least one flow path to the pressurizer relief tank and except for postulated second random failures, all flow paths to the pressurizer relief tank can be closed.

8. Agenda item #51 concerned steam generator water level reference leg heatup and the resulting errors in indicated level. Enclosure C is a report describing this matter for the SNUPPS design.

9. Agenda item #52 concerned the interface criteria given in WCAP-8584. Included in the Enclosure B FSAR changes is a change that addresses this matter.

10. Agenda item #60 concerned the potential effects of a steamline rupture on the automatic rod control system. More specifically, the postulated scenario is as follows. Following an intermediate steamline rupture outside containment, the automatic rod control system exhibits a consequential failure due to an adverse environment which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower ΔT. This scenario results in a lower DNB ratio than presently presented in the SNUPPS FSAR. However, as discussed below, a typical bounding analysis has been performed which calculated that no fuel damage would occur.

This scenario is considered to be very unlikely because, among other reasons, the break must occur at certain power levels, the break must be in a certain intermediate size range, the adverse environment must affect the turbine impulse transmitter, and the transmitter must cause a spurious low power signal without causing a reactor trip.

A typical bounding analysis of the intermediate steamline rupture was performed to calculate the extent of fuel damage due to rod control system withdrawal prior to reactor trip. Based upon the reduction in radial peaking factor with burn-up and conservative end-of-life physics parameters, no fuel damage was calculated to occur following the intermediate steamline rupture with a consequential rod control system failure.

A plant-specific analysis of this scenario is being performed for SNUPPS. The results of this analysis (scheduled to be available by the end of 1981) are expected to confirm the results of the typical bounding analysis discussed above.

11. Agenda item #69 concerned FSAR terminology used in describing extended hot shutdown from outside the control room. The local actions that are referred to do not involve lifting instrument leads. Appropriate wording changes are included in the Enclosure B FSAR changes.

12. Agenda item #13 concerned the analysis for determining if the pressurizer PORV would lift following a turbine trip from below 50% power. Included with the Enclosure B FSAR changes are additions to the analysis results that consider the low vacuum turbine trip case.

Very truly yours,

NA Petrick

Nicholas A. Petrick

RLS/dck/3a9

Enclosures A. Revised response to Position #6
B. FSAR Changes
C. Report on Steam Generator water level errors.

cc: J. K. Bryan UE
G. L. Koester KGE
D. T. McPhee KCPL
W. A. Hansen NRC/Cal
T. E. Vandel NRC/WC

6. Concern: Information provided by the applicants indicates that the reactor coolant wide range temperature indicators to be provided on the auxiliary shutdown panel will not meet all criteria applicable to safety related displays (such as being provided power from separate Class IE busses).

Position: The staff position is that reactor coolant system temperature is required parameter for maintaining the plant in a safe condition. Indicators meeting criteria applicable to safety related displays should be provided for reactor coolant temperature on the auxiliary shutdown panel.

Response: As indicated in Section 7.4.3 of the SNUPPS FSAR, the reactor coolant wide range temperature indicators are not essential for maintaining safe hot shutdown (hot standby). Safe hot shutdown can be maintained from the auxiliary shutdown panel through the use of the essential short-term monitoring indicators and controls listed in FSAR Section 7.4.3.1.1. These indicators and controls meet the criteria applicable to safety grade equipment (see FSAR Section 7.4.3.1.4).

The reactor coolant wide range temperature indicators (one per RCS cold leg) located on the auxiliary shutdown panel provide a highly reliable indication of reactor coolant temperature. These instrument loops are powered from protection sets I and II (loops 1 and 2 from protection set II, loops 3 and 4 from protection set I), isolated at the protection set cabinet, and routed to the auxiliary shutdown panel via separation groups five and six. The indicators are the same model number as the PAMS indicators provided for the same function on the main control board.

RLS/dck/3a12

ENCLOSURE B TO SLNRC 81-92
FSAR CHANGES

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reactor trip. Faults on the first stage turbine pressure circuits would result in upscale, conservative output for open circuits and a sustained current, limited by circuit resistance, for short circuits. Multiple failures imposed on these redundant circuits could potentially disable the P-13 interlock. In this event, the nuclear instrumentation power range signals would provide the P-7 safety interlock. Refer to functional diagram, Sheet 4 of Figure 7.2-1. The sensors for the P-13 interlock are seismically qualified.

Evaluations provided in Section 7.6.1 for the trip fluid pressure transmitter loops indicate that credible electrical faults would not degrade the functional performance of the safety-related BOP instrumentation.

In addition, the following measures will be taken to ensure the integrity of the cabling to the reactor protection system (RPS):

1. Inputs from the turbine steam stop valves will originate from four separate limit switches (one per valve), each of which is dedicated to providing an input to one channel of the RPS. Cables carrying these signals will be routed in individual conduits. The four circuits will be separated from one another, from non-Class IE circuits, and identified according to the criteria imposed on Class IE circuits from their source up to their terminations with the RPS cabinets.

2. Inputs from the emergency trip oil pressure and P-13 interlock instrumentation will be routed in a similar manner as are the turbine stop valve inputs.

The logic for this trip is shown on Figure 7.2-1 (Sheet 16).

action by manual or automatic means, the standard does not specifically preclude the sharing of initiated circuitry logic between automatic and manual functions. It is true that the manual safety injection initiation functions associated with one actuation train (e.g., train A) share portions of the automatic initiation circuitry logic of the same logic train; however, a single failure in shared functions does not defeat the protective action of the redundant actuation train (e.g., train B). A single failure in shared functions does not defeat the protective action of the safety function. It is further noted that the sharing of the logic by manual and automatic initiation is consistent with the system level action requirements of the IEEE Standard 279-1971, Section 4.17 and consistent with the minimization of complexity.

c. Conformance to regulatory guides and associated IEEE standards

Conformance to regulatory guides and associated IEEE standards is provided in Sections 7.1.2.5 and 7.1.2.6.

d. Failure mode and effects analyses

Failure mode and effects analyses have been performed on the engineered safety feature systems' equipment, and the results are provided in Reference 3. The interface criteria provided in Appendices B and C of Reference 3 have been met in the SNUPPS design.

In addition to the consideration given in this reference a loss of instrument air or loss of component cooling water to vital equipment has been considered. Neither the loss of instrument air nor the loss of cooling water (assuming no other accident conditions) can cause safety limits, as given in Chapter 16.0, to be exceeded. Likewise, loss of either of the two will not adversely affect the core or the reactor coolant system nor will it prevent an orderly shutdown if this is necessary. Furthermore, all pneumatically operated valves and controls will assume a preferred operating position upon loss of instrument air. It is also noted that, for conservatism during the accident analysis (Chapter 15.0), credit is not taken for the instrument air systems nor for any control system benefit.

The design does not provide any circuitry which will directly trip the reactor coolant pumps on a loss of component cooling water. Normally, indication in the control room is provided whenever component cooling

6. Suction pressure for each auxiliary feedwater pump (5)

7. Auxiliary feedwater pump turbine speed (rpm)

8. Discharge pressure for each auxiliary feedwater pump

9. Auxiliary feedwater flow to each steam generator

10. Condensate storage tank level

11. Reactor coolant (cold leg) wide range temperature

12. Source range nuclear power indicators

13. Intermediate range nuclear power indicators

14. Indicating lights (on-off/open-closed) for all power-operated equipment listed in a. above.

An equipment list for the auxiliary shutdown panel is contained in Table 7.4-1.

7.4.3.1.2 Controls at Switchgear Motor Control Centers, and Other Locations

In addition to the controls and monitoring indicators listed above, the following essential short-term controls are provided outside of the control room with a communication network between these control locations and the auxiliary shutdown control panel:

1. Reactor trip capability at the reactor trip switchgear.

2. START/STOP controls for both centrifugal charging pumps. Location: Charging pump switchgear.

3. START/STOP controls for the component cooling water pumps. Location: Component cooling water pumps switchgear.

4. START/STOP controls for the containment fan cooler units. Location: Cooler fan motor control centers.

5. START/STOP controls for the control room air-conditioning units. Location: At the equipment.

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6. START/STOP controls for the diesel generators. Location: Each diesel generator local control panel.

7. START/STOP controls for the essential service water pumps. Location: Essential service pump switchgear.

7.4.3.1.3 Controls for Extended Hot Standby

In order to maintain an extended hot standby (greater than 24 hours), additional negative reactivity must be added to the RCS. This can be accomplished by manual control of the normal charging and letdown systems via controls at the auxiliary shutdown panel, motor control centers, switchgears, and control of individual equipment at the device location.

In addition to the normal charging and letdown systems, the systems discussed in Appendix 5.4A may be used to maintain an extended hot standby by local actions outside the control room.

7.4.3.1.4 Design Bases Information

In accordance with NRC General Design Criterion 19, the capability of establishing a hot standby condition and maintaining the station in a safe status in that mode is considered an essential function. To ensure the availability of the auxiliary shutdown control panel and essential short-term control and indications after control room evacuation, the following design features have been utilized:

a. The auxiliary shutdown control panel, including all essential short-term instrumentation mounted on it, is designed to withstand earthquakes with no loss of essential functions. The essential short-term local control stations are also designed to withstand earthquakes with no loss of essential functions.

b. The essential short-term local stations and the auxiliary shutdown control panel, including essential short-term controls and indicators, are designed to comply with applicable portions of IEEE Standard 279-1971.

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TABLE 7.4-1

AUXILIARY SHUTDOWN PANEL EQUIPMENT LIST

Instrument No.	Unit No.	Service	Sep. Group
BB PI-455B	All	Pressurizer Pressure	NV
BB LI-450B	All	Pressurizer Level	1
BB LI-460B	All	Pressurizer Level	4
BB PI-483Z	All	RCS Pressure (wide range)	4
BB PI-485Z	All	RCS Pressure (wide range)	1
BB HIS-51B	All	Pzr Htrs Backup GP A	NV
BB HIS-52B	All	Pzr Htrs Backup Gp B	NV
AB PI-516B	All	SG A Pressure	4
AB PI-524B	All	SG B Pressure	1
AB PI-535B	All	SG C Pressure	4
AB PI-544B	All	SG D Pressure	1
AE-LI-581A	All	SG A Level (wide range)	1
AE-LI-502A	All	SG B Level (wide range)	4
AE-LI-583A	All	SG C Level (wide range)	1
AE-LI-504A	All	SG D Level (wide range)	4

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TABLE 7.4-1 (Sheet 2)

Instrument No.	Unit No.	Service	Sep. Group
AB PIC-1B	All	SG A Stm Dump to Atmos Ctrl	1
AB PIC-2B	All	SG B Stm Dump to Atmos Ctrl	2
AB PIC-3B	All	SG C Stm Dump to Atmos Ctrl	3
AB PIC-4B	All	SG D Stm Dump to Atmos Ctrl	4
AB HS-1	All	SG A Stm Dump Ctrl Xfr Sw	1
AB HS-2	All	SG B Stm Dump Ctrl XFR Sw	2
AB HS-3	All	SG C Stm Dump Ctrl Xfr Sw	3
AB HS-4	All	SG D Stm Dump Ctrl Xfr Sw	4
AB ZL-1B	All	SG A Stm Dump to Atmos Vlv Posn	1
AB ZL-2B	All	SG B Stm Dump to Atmos Vlv Posn	2
AB ZL-3B	All	SG C Stm Dump to Atmos Vlv Posn	3
AB ZL-4B	All	SG D Stm Dump to Atmos Vlv Posn	4
BG HIS-8149AB	All	Letdown Orifice A Isol Vlv	NV
BG HIS-8149BB	All	Letdown Orifice B Isol Vlv	NV

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TABLE 7.4-1 (Sheet 3)

BG HIS-8149CB	All	Letdown Orifice C Isol Vlv	NV
BG HIS-8152A	All	Letdown Ctmt Isol Vlv	4
BG HIS-8188A	All	Letdown Ctmt Isol Vlv	1
AL HK-5B	All	SG D Aux Fw Ctrl Vlv Md Pmp B	4
AL HS-5	All	SG D Aux Fw Ctrl Vlv Xfr Sw	4
AL ZL-5B	All	SG D Aux Fw Ctrl Vlv Posn	4
AL HK-8B	All	SG D Aux Fw Ctrl Vlv to Pmp	1
AL HS-6	All	SG D Aux Fw Ctrl Vlv Xfr Sw	1
AL ZL-8B	All	SG D Aux Fw Ctrl Vlv Posn	1
AL HK-7B	All	SG A Aux Fw Ctrl Vlv MD Pmp B	4
AL HS-7	All	SG A Aux Fw Ctrl Vlv Xfr Sw	4
AL ZL-7B	All	SG A Aux Fw Ctrl Vlv Posn	4
AL HK-8B	All	SG A Aux Fw Ctrl Vlv to Pmp	1
AL HS-8	All	SG A Aux Fw Ctrl Vlv Xfr Sw	1
AL ZL-8B	All	SG A Aux Fw Ctrl Vlv Posn	1
AL HK-9B	All	SG B Aux Fw Ctrl Vlv Md Pmp A	1

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TABLE 7.4-1 (Sheet 4)

Instrument No.	Unit No.	Service	Sep. Group
AL HS-9	All	SG B Aux Fw Ctrl Vlv Xfr Sw	1
AL ZL-9B	All	SG B Aux Fw Ctrl Vlv Posn	1
AL HK-10B	All	SG B Aux Fw Ctrl Vlv to Pmp	4
AL HS-10	All	SG B Aux Fw Ctrl Vlv Xfr Sw	4
AL ZL-10B	All	SG B Aux Fw Ctrl Vlv Posn	4
AL HK-11B	All	SG C Aux Fw Ctrl Vlv Md Pmp A	1
AL HS-11	All	SG C Aux Fw Ctrl Vlv Xfr Sw	1
AL ZL-11B	All	SG C Aux Fw Ctrl Vlv Posn	1
AL HK-12B	All	SG C Aux Fw Ctrl Vlv to Pmp	4
AL HS-12	All	SG C Aux Fw Ctrl Vlv Xfr Sw	4
AL ZL-12B	All	SG C Aux Fw Ctrl Vlv Posn	4
AL FI-1B	All	SG D Aux Fw Flow	4
AL FI-2B	All	SG A Aux Fw Flow	1
AL FI-3B	All	SG B Aux Fw Flow	2
AL FI-4B	All	SG C Aux Fw Flow	3

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TABLE 7.4-1 (Sheet 5)

Instrument No.	Unit No.	Service	Sep. Group
AL PI-15B	All	Md Aux Fw Pmp B Disch Press	NV
AL PI-18B	All	Md Aux Fw Pmp A Disch Press	NV
AL PI-21B	All	Turb Driven Aux Fw Pmp Disch Press	NV
AL PI-25B	All	Md Aux Fw Pmp A Suct Press	1
AL PI-24B	All	Md Aux Fw Pmp B Suct Press	4
AL PI-28B	All	Turb Drive Aux Fw Pmp Suct Press	2
AL HIS-22B	All	Md Aux Fw Pmp B	4
AL HIS-23B	All	Md Aux Fw Pmp A	1
FC-ZL-312AD, AE, AF	All	Afpt Trip & Throt Vlv Posn	2
FC HIS-312B	All	Turb Driven Aux Fw Pmp Trip A Throt Vlv	2
FC HIK-313B	All	Aux Fp Turb Speed Gov Ctrl	2
AB HIS-5B	All	Turb Drvn Aux Fw Pmp Stm Isol Vlv	2
AB HIS-8B	All	Turb Drvn Aux Fw Pmp Stm Isol Vlv	2
AP LI-4B	All	Cond Stor Tank Level	NV

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TABLE 7.4-1 (Sheet 6)

Instrument No.	Unit No.	Service	Sep. Group
AL HIS-39B	All	ESW to Md Aux Fw Pmp B	4
AL HIS-31B	All	ESW to Md Aux Fw Pmp A	1
AL HIS-32B	All	ESW to Turb Driven Aux Fw Pmp	1
AL HIS-33B	All	ESW to Turb Driven Aux Fw Pmp	4
AL HIS-34B	All	Cst to Md Aux Fw Pmp B	4
AL HIS-35B	All	Cst to Md Aux Fw Pmp A	1
AL HIS-36B	All	Cst to Turb Driven Aux Fw Pmp	1
BB-TI-413X	All	RCS Cold Leg Temp Loop 1	NV
BB-TI-423X	All	RCS Cold Leg Temp Loop 2	NV
BB-TI-433X	All	RCS Cold Leg Temp Loop 3	NV
BB-TI-443X	All	RCS Cold Leg Temp Loop 4	NV
SE-NI-31C	All	Source Range Nuclear Inst	NV
SE-NI-32C	All	Source Range Nuclear Inst	NV
AE-LI-517X	All	S.G. A Level (narrow range)	4
AE-LI-528X	All	S.G. B Level (narrow range)	1

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TABLE 7.4-1 (Sheet 7)

Instrument No.	Unit No.	Service	Sep. Group
AE-LI-537X	All	S.G. C Level (narrow range)	4
AE-LI-548X	All	S.G. D Level (narrow range)	1
BG HIS-459B	All	RCS Letdown to Regen Hx	NV
BG HIS-460B	All	RCS Letdown to Regen Hx	NV
FC-HS-313	All	Afpt Gov Ctrl Sel Sw	2
SE-NI-35C	All	Intermediate Range Nuclear Inst	NV
SE-NI-36C	All	Intermediate Range Nuclear Inst	NV
FC-ZL-315B, 317B	All	AFPT Gov Vlv Position	2
FC-ZL-312DB	All	Afpt Throttle Vlv Trip Mech Pos	2

NV - NON-VITAL

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7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

The information necessary to monitor the nuclear steam supply systems, the containment systems, and the balance of plant is displayed on the operator's console and the various control boards located within the control room. These indications include the information to control and operate the unit through all operating conditions, including anticipated operational occurrences and accident and post-accident conditions. Hot shutdown information is also displayed on the auxiliary shutdown control panel located outside the control room (refer to Section 7.4.3). This section is limited to the discussion of those display instruments which provide information to enable the operator to assess reactor status, the onset and severity of accident conditions, and engineered safety feature system (ESFS) status and performance, or to enable the operator to intelligently perform vital manual actions such as safe shutdown and initiation of manual ESFSs. Reactivity control is monitored by sampling of the reactor coolant for boron.

The surveillance instrumentation, which includes indicators, annunciators, recorders, and lights, consists of specific instrumentation for the following functions:

a. Reactor trip
b. Engineered safety features
c. Safe shutdown

This section discusses instrumentation that is required for safety as well as instrumentation that is only indirectly related to safety. The safety-related display instrumentation provided in the control room is listed in Table 7.5-4 and 7.5-5.

This section also furnishes a summary of important display instrumentation provided to monitor system status and performance. The bypassed status indication is treated separately to establish a clear definition of the system of bypass indication. The display instrumentation defined for bypass, status, and performance monitoring is not safety related (refer to Table 7.1-2, Sheet 2) since failure in no way degrades the operation of safety systems and poses no threat to public health and safety.

Refer to Section 1.7 for drawings associated with auxiliary shutdown panel, safety-related display instrumentation, and main control board layouts and ESFS logic diagrams.

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TABLE 7.5-4

SAFETY-RELATED DISPLAY INSTRUMENTATION

LOCATED ON THE CONTROL BOARD - (NSSS SCOPE OF SUPPLY)

Parameter	Indicator Tag No.	Notes 1 and 2 PAMS Separation Group I	II
WIDE RANGE T HOT LEG	TI 413A	X	
WIDE RANGE TO HOT LEG	TI 423A	X	
WIDE RANGE T COLD LEG	TI 413B		X
WIDE RANGE T COLD LEG	TI 423B		X
PRESSURIZER WATER LEVEL	LI 459A	X	
PRESSURIZER WATER LEVEL	LI 460A		X
PRESSURIZER WATER LEVEL	LI 461A	X	
STEMA GEN. LOOP 3 PRESSURE	PI 534A	X	
STEAM GEN. LOOP 1 PRESSURE	PI 514A	X	
STEAM GEN. LOOP 2 PRESSURE	PI 524A	X	
STEAM GEN. LOOP 4 PRESSURE	PI 544A	X	
STEAM GEN. LOOP 1 PRESSURE	PI 515A		X
STEAM GEN. LOOP 2 PRESSURE	PI 525A		X
STEAM GEN. LOOP 4 PRESSURE	PI 545A		X
STEAM GEN. LOOP 3 PRESSURE	PI 535A		X
STEAM GEN. LOOP 1 PRESSURE	PI 516A		X
STEAM GEN. LOOP 4 PRESSURE	PI 546A		X
STEAM GEN. LOOP 2 PRESSURE	PI 526A	X	
STEAM GEN. LOOP 3 PRESSURE	PI 536A	X	
STEAM GEN. LOOP 2 WATER LEVEL	LI 529	X	
STEAM GEN. LOOP 3 WATER LEVEL	LI 539	X	
STEAM GEN. LOOP 1 WATER LEVEL	LI 519		X
STEAM GEN. LOOP 4 WATER LEVEL	LI 549		X
STEAM GEN. LOOP 1 WATER LEVEL	LI 518	X	
STEAM GEN. LOOP 2 WATER LEVEL	LI 528	X	
STEAM GEN. LOOP 3 WATER LEVEL	LI 538	X	
STEAM GEN. LOOP 4 WATER LEVEL	LI 548	X	
STEAM GEN. LOOP 1 WATER LEVEL	LI 517		X
STEMA GEN. LOOP 2 WATER LEVEL	LI 527		X
STEAM GEN. LOOP 3 WATER LEVEL	LI 537		X
STEAM GEN. LOOP 4 WATER LEVEL	LI 547		X
CONTAINMENT PRESSURE N. R.	PI 934		X
CONTAINMENT PRESSURE N. R.	PI 935	X	
CONTAINMENT PRESSURE N. R.	PI 936		X
CONTAINMENT PRESSURE N. R.	PI 937	X	
STEAM GEN. W. R. WATER LEVEL	LI 501	X	
STEAM GEN. W. R. WATER LEVEL	LI 502		X
STEAM GEN. W. R. WATER LEVEL	LI 503	X	
STEAM GEN. W. R. WATER LEVEL	LI 504		X

TABLE 7.5-4 (Sheet 2)

Parameter	Indcator Tag No.	Notes 1 and 2 PAMS Separation Group	
		I	II
R. C. S. W. R. PRESSURE	PI 405	X	
R. C. S. W. R. PRESSURE	PI 403		X
BORIC ACID WATER LEVEL	LI 102	X	
R. W. S. T. WATER LEVEL	LI 930	X	
R. W. S. T. WATER LEVEL	LI 931		X
R. W. S. T. WATER LEVEL	LI 932	X	
R. W. S. T. WATER LEVEL	LI 933		X
SAFETY INJECTION FLOW	FI 917A	X	
SAFETY INJECTION FLOW	FI 918B		X
CONTAINMENT PRESSURE W. R.	PI 938	X	
CONTAINMENT PRESSURE W. R.	PI 939		X
R. C. S. EXCESS LETDOWN HEAT EXCHANGER	TI 137A	X	
R. C. S. EXCESS LETDOWN HEAT EXCHANGER	TI 137B		X
R. C. S. EXCESS LETDOWN HEAT EXCHANGER	TI 138A	X	
R. C. S. EXCESS LETDOWN HEAT EXCHANGER	TI 138B		X
BORIC ACID WATER LEVEL	LI 104		X
BORIC ACID WATER LEVEL	LI 105	X	
BORIC ACID WATER LEVEL	LI 106		X

NOTES:

1. PAM I routed as Separation Group 1. PAM II routed as Separation Group 4.

2. See Westinghouse process control block diagrams for the applicable protection set.

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TABLE 7.5-5

SAFETY-RELATED DISPLAY INSTRUMENTATION

LOCATED ON THE CONTROL BOARD - (BOP SCOPE OF SUPPLY)

Parameter	Indicator Tag No.	Separation Group
AUXILIARY FEEDWATER-FLOW	AL-FI-1A	4
AUXILIARY FEEDWATER-FLOW	AL-FI-2A	1
AUXILIARY FEEDWATER-FLOW	AL-FI-3A	2
AUXILIARY FEEDWATER-FLOW	AL-FI-4A	3
CONDENSATE STORAGE TANK-PRESSURE	AL-PI-37	1
CONDENSATE STORAGE TANK-PRESSURE	AL-PI-38	2
CONDENSATE STORAGE TANK-PRESSURE	AL-PI-39	4
TURBINE DRIVEN AUXILIARY FEED PUMP-SUCTION PRESS.	AL-PI-26A	2
MOTOR DRIVEN AUXILIARY FEED PUMP A-SUCTION PRESS.	AL-PI-25A	1
MOTOR DRIVEN AUXILIARY FEED PUMP B-SUCTION PRESS.	AL-PI-24A	4
CONTROL ROOM AIR INTAKE-CHLORINE	GK-AI-2	4
CONTROL ROOM AIR INTAKE-CHLORINE	GK-AI-3	1
CONTROL ROOM AIR INTAKE-GASEOUS RADIOACTIVITY	GK-RI-4*	4
CONTROL ROOM AIR INTAKE-GASEOUS RADIOACTIVITY	GK-RI-5*	1
CONTAINMENT-GASEOUS RADIOACTIVITY	GT-RI-31*	4
CONTAINMENT-GASEOUS RADIOACTIVITY	GT-RI-32*	1
CONTAINMENT-HYDROGEN	GS-AI-10	4
CONTAINMENT-HYDROGEN	GS-AI-19	1
CONTAINMENT SUMP/CONTAINMENT LEVEL	LF-LI-10	4
CONTAINMENT SUMP/CONTAINMENT LEVEL	LF-LI-9	1
CONTAINMENT PURGE-GASEOUS RADIO-ACTIVITY	GT-RI-33*	4
CONTAINMENT PURGE-GASEOUS RADIO-ACTIVITY	GT-RI-22*	1
CONTAINMENT SPRAY ADDITIVE TANK-LEVEL	EN-LI-17	4
CONTAINMENT SPRAY ADDITIVE TANK-LEVEL	EN-LI-19	1
FUEL BUILDING-GASEOUS RADIOACTIVITY	GG-RI-28*	4
FUEL BUILDING-GASEOUS RADIOACTIVITY	GG-RI-27*	1
CONTAINMENT-AIR TEMPERATURE	GN-TI-61	4
CONTAINMENT-AIR TEMPERATURE	GN-TI-60	1
CONTAINMENT-AIR TEMPERATURE	GN-TI-63	4
CONTAINMENT-AIR TEMPERATURE	GN-TI-62	1
CONTAINMENT-POST-ACCIDENT RADIATION	GT-RI-60	4
CONTAINMENT-POST-ACCIDENT RADIATION	GT-RI-59	1
CONTROL BUILDING SUMP-LEVEL	LF-LI-125	4
CONTROL BUILDING SUMP-LEVEL	LF-LI-124	1
DIESEL GENERATOR BUILDING SUMP-LEVEL	LF-LI-106	4
DIESEL GENERATOR BUILDING SUMP-LEVEL	LF-LI-105	1

TABLE 7.5-5 (Sheet 2)

Parameter	Indicator Tag No.	Separation Group
RHR PUMP ROOM SUMP-LEVEL	LF-LI-101	4
RHR PUMP ROOM SUMP-LEVEL	LF-LI-102	1
AUXILIARY BUILDING SUMP-LEVEL	LF-LI-104	4
AUXILIARY BUILDING SUMP-LEVEL	LF-LI-103	1

*Digital display on radiation monitoring panel SP-067.

ICSB ITEM #39

is at a temperature below the reference nil ductility temperature (RNDT). The monitored system temperature signals are processed to generate the reference pressure limit program which is compared to the actual monitored RCS pressure. This comparison provides an actuation signal to an actuation device which will cause the PORV to automatically open, if necessary, to prevent pressure conditions from exceeding allowable limits. Refer to Figure 7.6-4 for the block diagram showing the interlocks for RCS pressure control during low temperature operation.

pressure and temperature

~~As shown on this figure,~~ the generating station variables required for this interlock are channelized as follows:

Replace with Insert (X)

~~a. Protection Set I~~

~~(1) Wide range RCS temperature from hot legs~~

~~(2) Wide range RCS pressure (PT 405)~~

~~b. Protection Set II~~

~~(1) Wide range RCS temperature from cold leg~~

~~c. Protection Set IV~~

~~(1) Wide range RCS pressure (PT 403)~~

~~The wide range temperature signals, as inputs to protection sets I and II, continuously monitor RCS temperature conditions whenever plant operation is at a temperature below the RNDT. In protection set I, the existing RCS hot leg wide range temperature channels will supply continuous analog input signals [illegible] to an auctioneering device, which is located in the process rack of control group 1.~~

INSERT (A)

The lowest reading is selected and input to a function generator which calculates the related reference pressure limit program, considering the plant's allowable pressure and temperature limits. Also available from protection set [illegible] is the wide range RCS pressure signal. ~~which is [illegible]~~ The reference pressure from the function generator is compared to the actual RCS pressure monitored by the wide range pressure channel. The error signal derived from the difference between the reference pressure and the actual measured pressure will first annunciate a main control board alarm whenever the actual measured pressure approaches, within a predetermined amount, the reference pressure. On a further increase in measured pressure, the error signal will generate an ~~[illegible]~~ actuation signal.

~~The actuation signal available from the process control cabinets will control PORV "[illegible]" whenever a temperature-dependent permissive signal from process control group 4 is present. The temperature-dependent permissive to the PORVs actuation device effectively disarms (blocks) the actuation~~

INSERT (A) (to page 7.6-5)

The wide range RCS temperatures in each protection set are auctioneered in an auctioneering device in each protection set to select the lowest reading.
An alarm is actuated when the auctioneered low temperature from the RCS wide range temperature channels falls within the range of cold overpressure applicability, thereby alerting the operator to arm the RCS cold overpressure mitigation system which automatically opens the block valve, when the block valve control switch is in the automatic position.

INSERT (X) (to page 7.6-5)

a. Pressure and Temperature Inputs to PCV455A

   (1) Four wide range RCS temperature signals derived from channels in a Train A related protection set.
   (2) One wide range RCS pressure signal derived from a channel in a Train A related protection set.

b. Pressure and Temperature Inputs to PCV456A

   (1) Four wide range RCS temperature signals derived from channels in a Train B related protection set.
   (2) One wide range RCS pressure signal derived from a channel in a Train B related protection set.

→ INSERT (B) 9F

~~signal at temperatures greater than the range of concern. This will prevent unnecessary system actuation when at normal RCS operating conditions, as a result of a failure in the process sensors.~~

The monitored generating redundant station variables that generate the actuation signal for the [illegible] PORV are processed in a similar manner. ~~In the case of PORV "B," the reference temperature is generated [illegible] from the lowest auctioneered wide range cold leg temperature, the auctioneering device deriving its input from the RCS wide range temperature in protection set II, and the actual measured pressure signal is available from protection set IV. Therefore, the plant variables used for PORV "B" are derived from a protection set that is independent of the sets from which plant variables used for PORV "A" are derived. The error signal derivation used for [illegible] available from the control group.~~

Upon receipt of the actuation signal, the actuation device will automatically cause the PORV to open. Upon sufficient RCS inventory letdown, the operating RCS pressure will decrease, clearing the actuation signal. Removal of this signal causes the PORV to close.

7.6.6.1 Analysis of Interlocks

Many criteria presented in IEEE Standards 279-1971 and 338-1971 do not apply to the interlocks for RCS pressure control during low temperature operation, because the interlocks do not perform a protective function but, rather, provide automatic pressure control at low temperatures as a back-up to the operator. However, although IEEE Standard 279-1971 criteria do not apply, some advantages of the dependability and benefits of an IEEE Standard 279-1971 design have accrued by including selected elements, as noted above, in the protection sets and by organizing the control of the two PORVs (either of which can accomplish the RCS pressure control function) into dual channels. [illegible]

The design of the low temperature interlocks for RCS pressure control is such that pertinent features include:

a. No credible failure at the output of the protection set racks, after the output leaves the racks to interface with the interlocks, will prevent the associated protection system channel from performing its protective function because [illegible] of the separation of Train B interlocks from Train A (see Figure 7.6-4).

b. Testing capability for elements of the interlocks within (not external to) the protection system is consistent with the testing principles and methods

INSERT (B) (to page 7.6-6)

Logic is also provided to close the block valve automatically if the relief valve fails or sticks in the open position following some plant transient, and the RCS pressure drops below the reset pressure for the relief valve.

when the RCS temperature is above the cold overpressurization setpoint,

discussed in Section 7.2.2.2.3, item J. It should be noted that there is an annunciator which provides an alarm when the block valve is armed ~~[illegible]~~ coincident with a closed position of the motor-operated (MOV) pressurizer ~~relief~~ block valve. This MOV is in the same fluid path as the PORV, with a separate MOV and alarm used with the second PORV.

c. A loss of offsite power will not defeat the provisions for an electrical power source for the interlocks because these provisions are through onsite power, which is described in Section 8.3.

7.6.7 ISOLATION OF ESSENTIAL SERVICE WATER (ESW) TO THE AIR COMPRESSORS

7.6.7.1 Description

As stated in Section 9.2.1.2.2.1, ESW flow to the nonsafety-related air compressors and associated aftercoolers is maintained following a DBA. Instrumentation and controls are provided to automatically isolate each train of the ESW to the air compressors on high flow. ESW to the air compressors can also be isolated by remote manual means.

Each control system (one per train of the ESW) utilizes a differential pressure transmitter and bistable which senses flow through the associated isolation valve. On high flow (indicative of gross leakage in the nonseismic portion of the system), the control system automatically closes the isolation valve.

The isolation valve will remain in the closed position until the valve is manually reset by the operator in the control room.

A means of remote manual isolation is provided in the control room. The status of each isolation valve is indicated by open and closed indicating lights in the control room.

The isolation valves are air operated and are designed to fail closed on the loss of air and electrical power.

a. Initiating circuits

Each isolation valve is automatically actuated by flow monitoring instrumentation. The isolation valves can also be closed via control switches in the control room.

b. Logic

The logic diagram for the isolation of the ESW to the air compressors is provided in Section 1.7.

c. Conformance to other criteria and standards

Conformance to other criteria and standards is indicated in Table 7.1-2.

7.6.9 FIRE PROTECTION AND DETECTION

Fire protection and detection is discussed in Section 9.5.1.

INSERT C

7.6.10 INTERLOCKS FOR PRESSURIZER PRESSURE RELIEF SYSTEM

7.6.10.1 Description of Pressurizer Pressure Relief System

The pressurizer pressure relief (PPR) system provides the following:

a. Capability for RCS overpressure mitigation during cold shutdown, heatup, and cooldown operations to minimize the potential for impairing reactor vessel integrity when operating at or near the vessel ductility limits.

b. Capability for RCS depressurization following Condition II, III, and IV events.

c. Interlock that, with the pressurizer PORVs and PORV block valves in auto control, closes the PORV block valves and prevents signals from the pressurizer pressure control system from opening the PORVs when pressurizer pressure is low.

7.6.10.2 Description of Pressurizer Pressure Relief System Interlocks

Interlocks for the PPR system control the opening and closing of the pressurizer PORVs and the PORV block valves. These

interlocks provide the following functions:

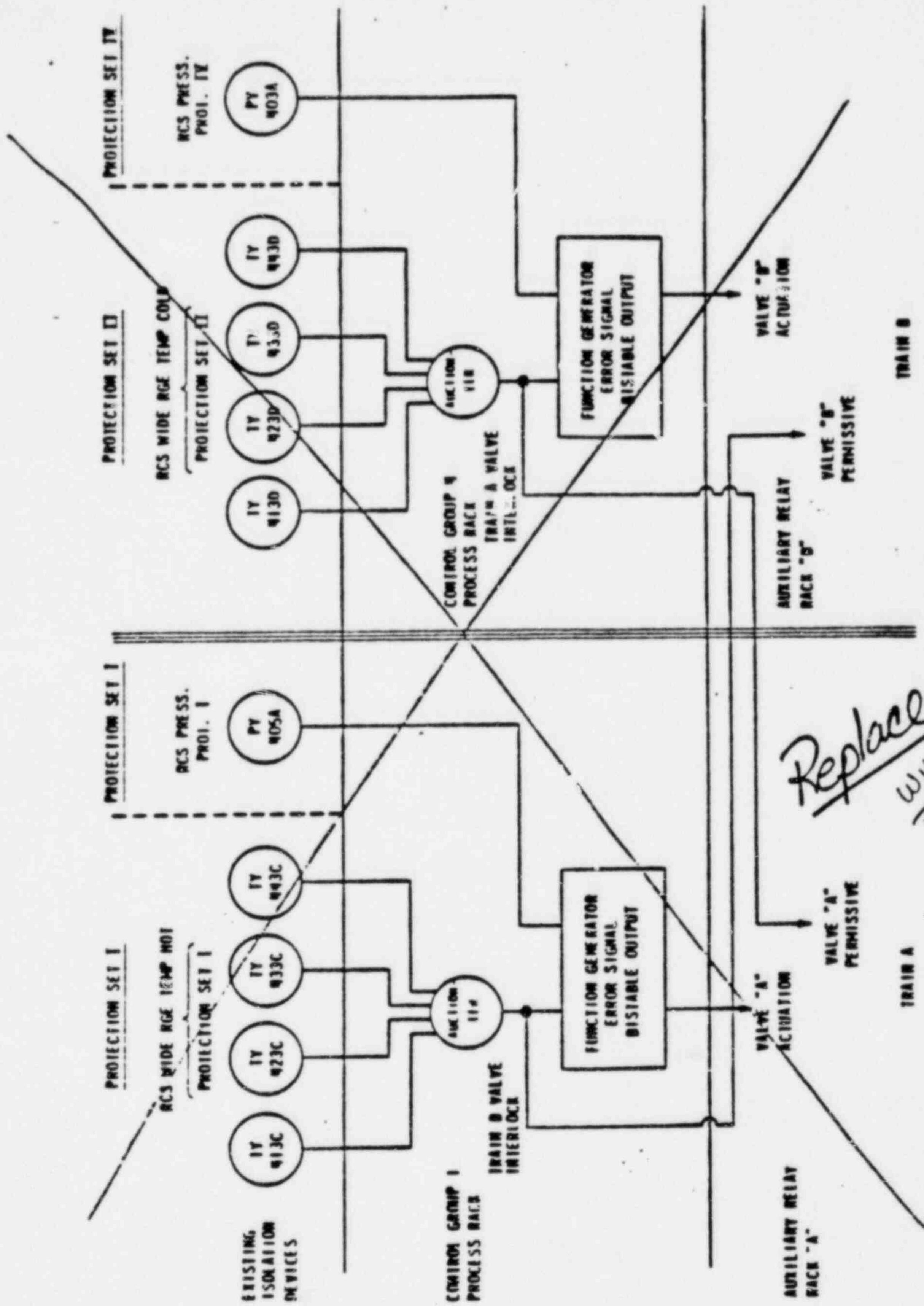
a. Pressurizer pressure control (refer to Section 7.7.1.5 for a description).

b. RCS pressure control during low temperature operation (refer to Sections 5.2.2 and 7.6.6 for a description).

c. RCS pressure control to achieve and maintain a cold shutdown and to heatup using equipment that is required for safety (refer to Appendix 5.4A for a description).

The interlock functions that provide pressurizer pressure control are derived from process parameters as shown on Figure 7.2-1, sheet 11 and the interlock logic functions as well as process parameter inputs required for low temperature operation as shown on Figure 7.6-4. The functions shown on Figure 7.6-4 include those needed for the PORV block valves as well as the pressurizer PORVs to meet both interlock logic and manual operation requirements where manual operation is ~~given~~ at the main control board. ~~or at the shutdown panel outside the main control room.~~

13,597-39



Replace with the following sheets.

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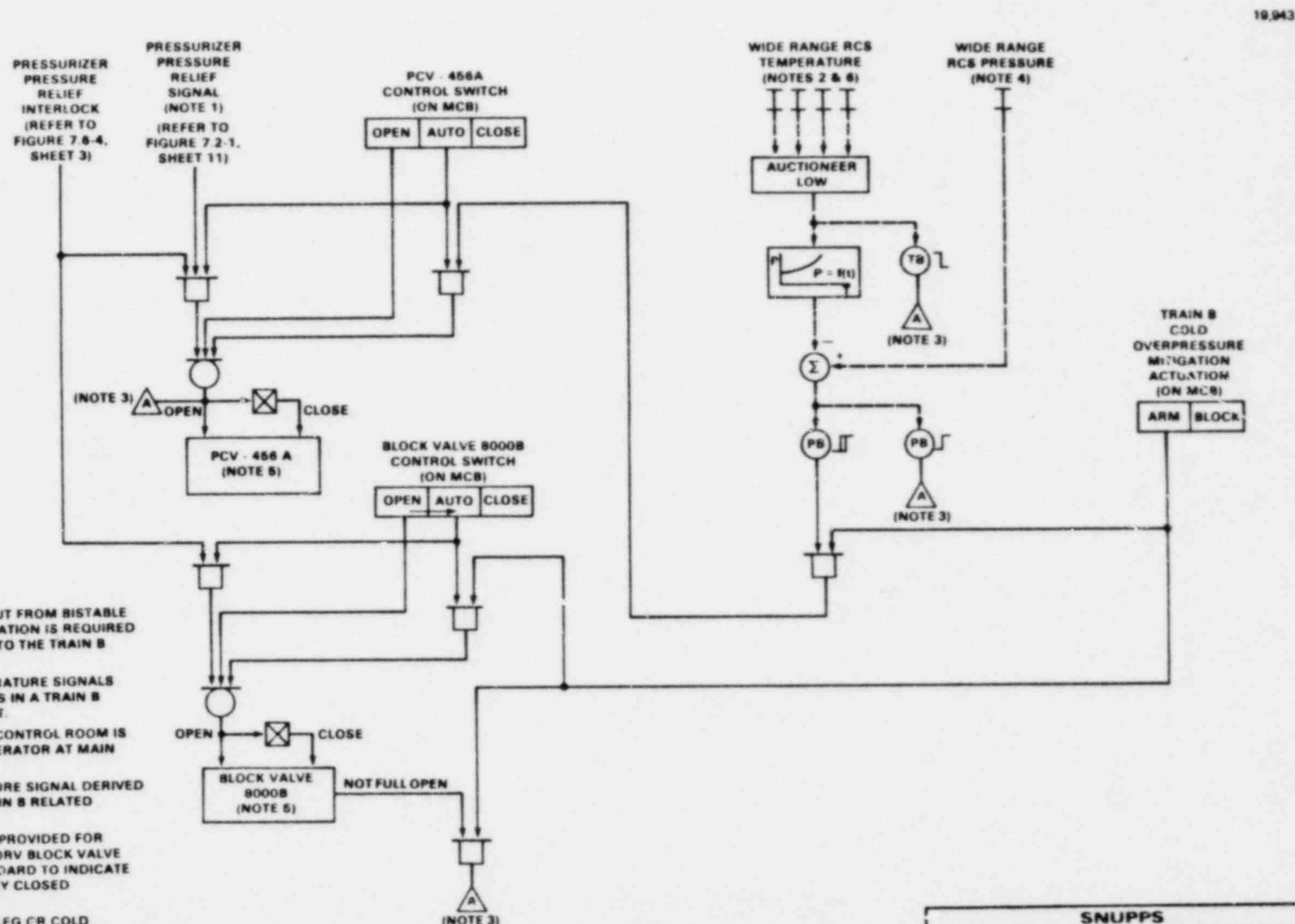
FIGURE 7.6-4

DIAGRAM SHOWING GENERATING PLANT VARIABLE PROCESSING FOR LOW TEMPERATURE INTERLOCKS FOR RCS PRESSURE CONTROL



NOTES:

1. THIS SIGNAL IS THE OUTPUT FROM BISTABLE PB-456E ELECTRICAL ISOLATION IS REQUIRED TO BRING THIS SIGNAL INTO THE TRAIN B PROTECTION CABINET.
2. WIDE RANGE RCS TEMPERATURE SIGNALS DERIVED FROM CHANNELS IN A TRAIN B RELATED PROTECTION SET.
3. ANNUNCIATION IN MAIN CONTROL ROOM IS REQUIRED VISIBLE TO OPERATOR AT MAIN CONTROL BOARD.
4. WIDE RANGE RCS PRESSURE SIGNAL DERIVED FROM CHANNEL IN A TRAIN B RELATED PROTECTION SET.
5. STATUS LIGHTS MUST BE PROVIDED FOR EACH PORV AND EACH PORV BLOCK VALVE AT THE MAIN CONTROL BOARD TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
6. THE RCS LOOP AND HOT LEG OR COLD LEG ASSIGNMENTS FOR THE WIDE RANGE RCS TEMPERATURE SIGNALS MUST BE CONSISTENT WITH THE REQUIREMENTS FOR RVLIS AND PAMS.

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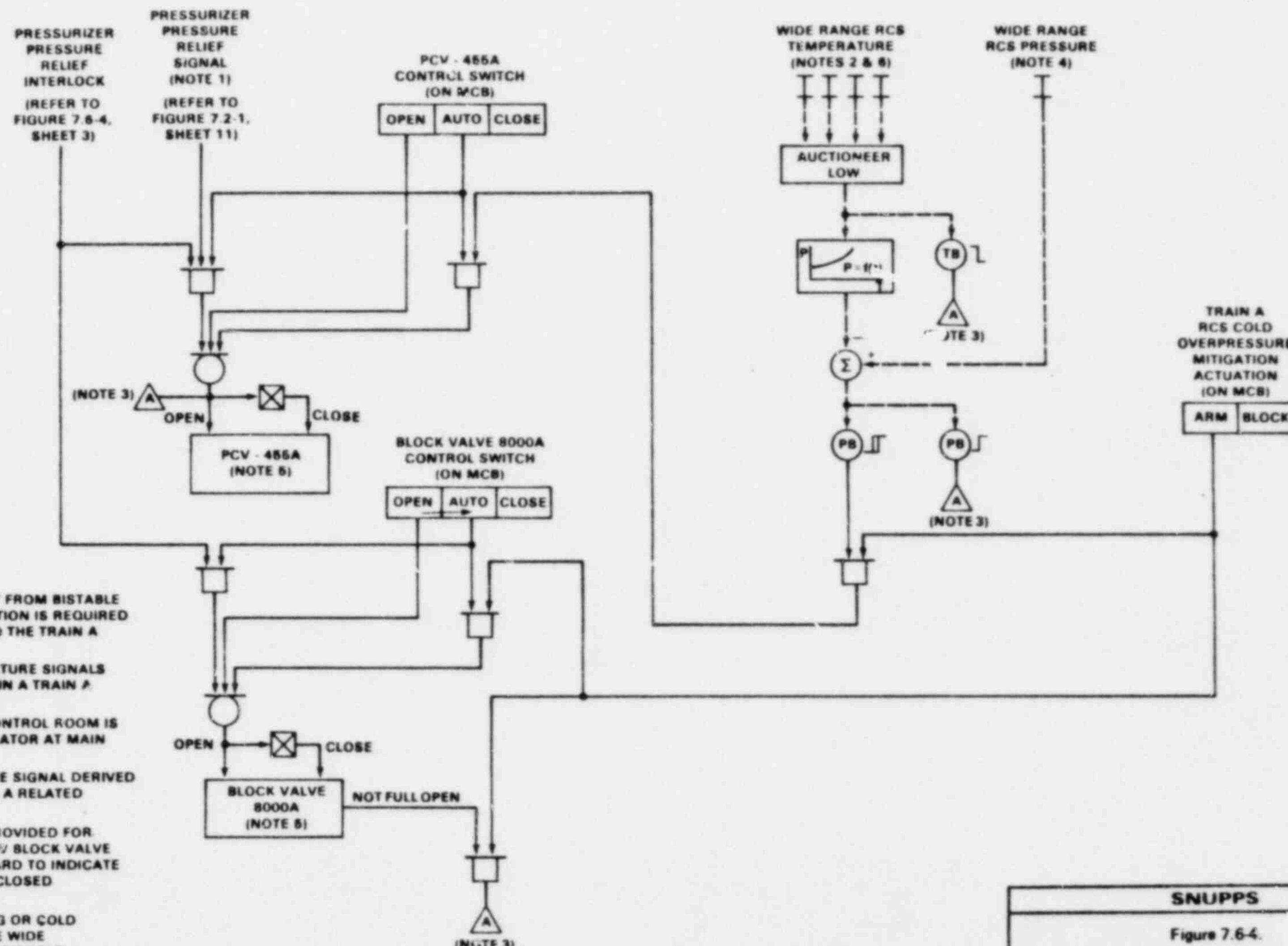
Figure 7.6-4.

Train B Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System
Sheet 1

19,943.1



NOTES:

1. THIS SIGNAL IS THE OUTPUT FROM BISTABLE PB-455E ELECTRICAL ISOLATION IS REQUIRED TO BRING THIS SIGNAL INTO THE TRAIN A PROTECTION CABINET.
2. WIDE RANGE RCS TEMPERATURE SIGNALS DERIVED FROM CHANNELS IN A TRAIN A RELATED PROTECTION SET.
3. ANNUNCIATION IN MAIN CONTROL ROOM IS REQUIRED VISIBLE TO OPERATOR AT MAIN CONTROL BOARD.
4. WIDE RANGE RCS PRESSURE SIGNAL DERIVED FROM CHANNEL IN A TRAIN A RELATED PROTECTION SET.
5. STATUS LIGHTS MUST BE PROVIDED FOR EACH PORV AND EACH PORV BLOCK VALVE AT THE MAIN CONTROL BOARD TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
6. THE RCS LOOP AND HOT LEG OR COLD LEG ASSIGNMENTS FOR THE WIDE RANGE RCS TEMPERATURE SIGNALS MUST BE CONSISTENT WITH THE REQUIREMENTS FOR RVLIS AND PAMS.

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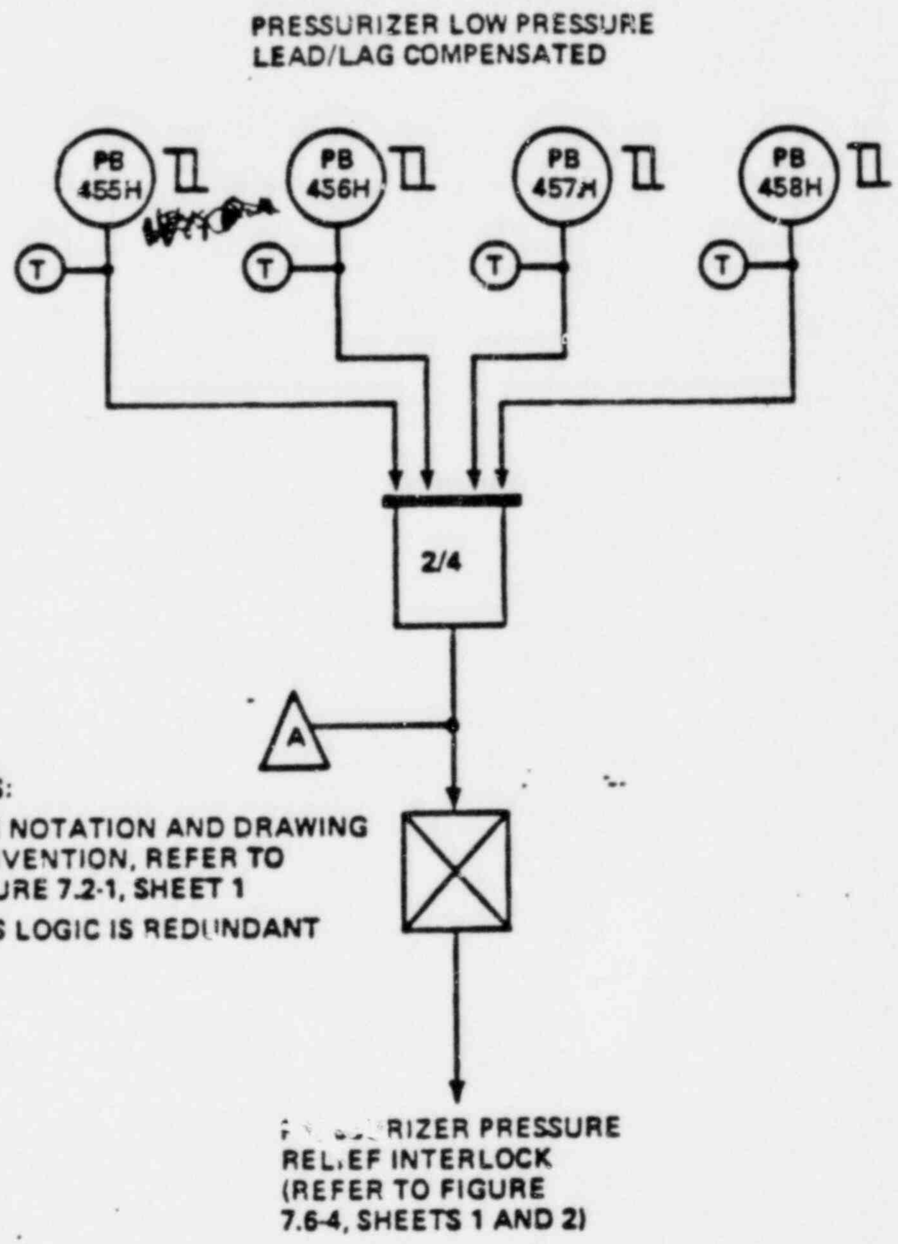
Figure 7.6-4.

Train A Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System Sheet 2

19156-1



NOTES:

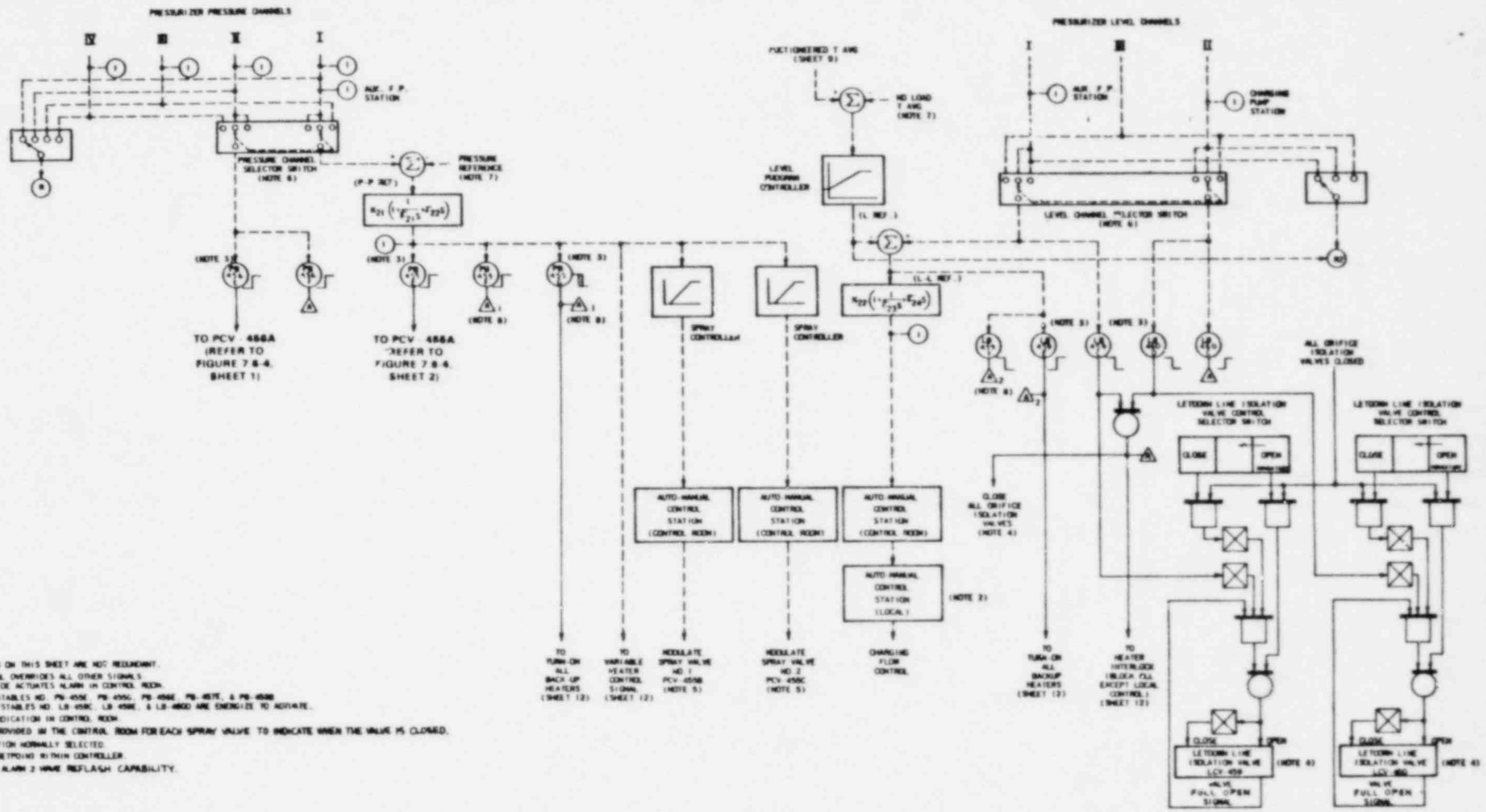
1. FOR NOTATION AND DRAWING CONVENTION, REFER TO FIGURE 7.2-1, SHEET 1
2. THIS LOGIC IS REDUNDANT

SNUPPS

Figure 7.6-4

Functional Diagram of Logic Requirement for Pressurizer Pressure Relief System Interlock

Sheet 3



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
3. PRESSURE BISTABLES NO. PB-455E, PB-455G, PB-456E, PB-457E, & PB-458B AND LEVEL BISTABLES NO. LB-459C, LB-459E, & LB-460D ARE ENERGIZE TO ACTUATE.
4. OPEN/SHUT INDICATION IN CONTROL ROOM.
5. A LIGHT IS PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN THE VALVE IS CLOSED.
6. CENTER POSITION NORMALLY SELECTED.
7. ADJUSTABLE SETPOINT WITHIN CONTROLLER.
8. ALARM 1 AND ALARM 2 HAVE REFLASH CAPABILITY.

SNUPPS

FIGURE 7.2-1
FUNCTIONAL DIAGRAMS
(PRESSURIZER PRESSURE AND LEVEL CONTROL)
(SHEET 11)

RESPONSE TO ICSB AGENDA ITEM #13
(AND NUREG-0737, ITEM II.K.3.10)

The NRC has raised the question of whether the pressurizer power operated relief valves would be actuated for a turbine trip without reactor trip below a power level of 50% (P-9 setpoint). An analysis has been performed using realistic yet conservative values for the core physics parameters (primarily reactivity feedback coefficients and control rod worths), and a conservatively hi h initial power, average reactor temperature (T_{AVG}), and pressurizer press re level to account for instrument inaccuracies.

The transient was initiated from the setpoint for the P-9 interlock, namely 50% of the reactor full power level plus 2% for power measurement uncertainty. This is a conservative starting point, and would bracket all transients initiated from a lower power level. The core physics parameters used were the ones that would result in the most positive reactivity feedbacks (i.e. highest power levels). The steam dump valves were assumed to be actuated by the load rejection controller.

Based upon the results from the analysis, the peak pressure reached in the pressurizer would be 2302 psia. The setpoint for the actuation of the pressurizer power operated relief valves is 2350 psia. Even including the ± 20 psi pressure measurement uncertainty, there is still a margin of 28 psi between the peak pressure reached and the minimum activation pressure for the pressurizer power operated relief valves.

INSERT A

pg 18.2-75

INSERT A

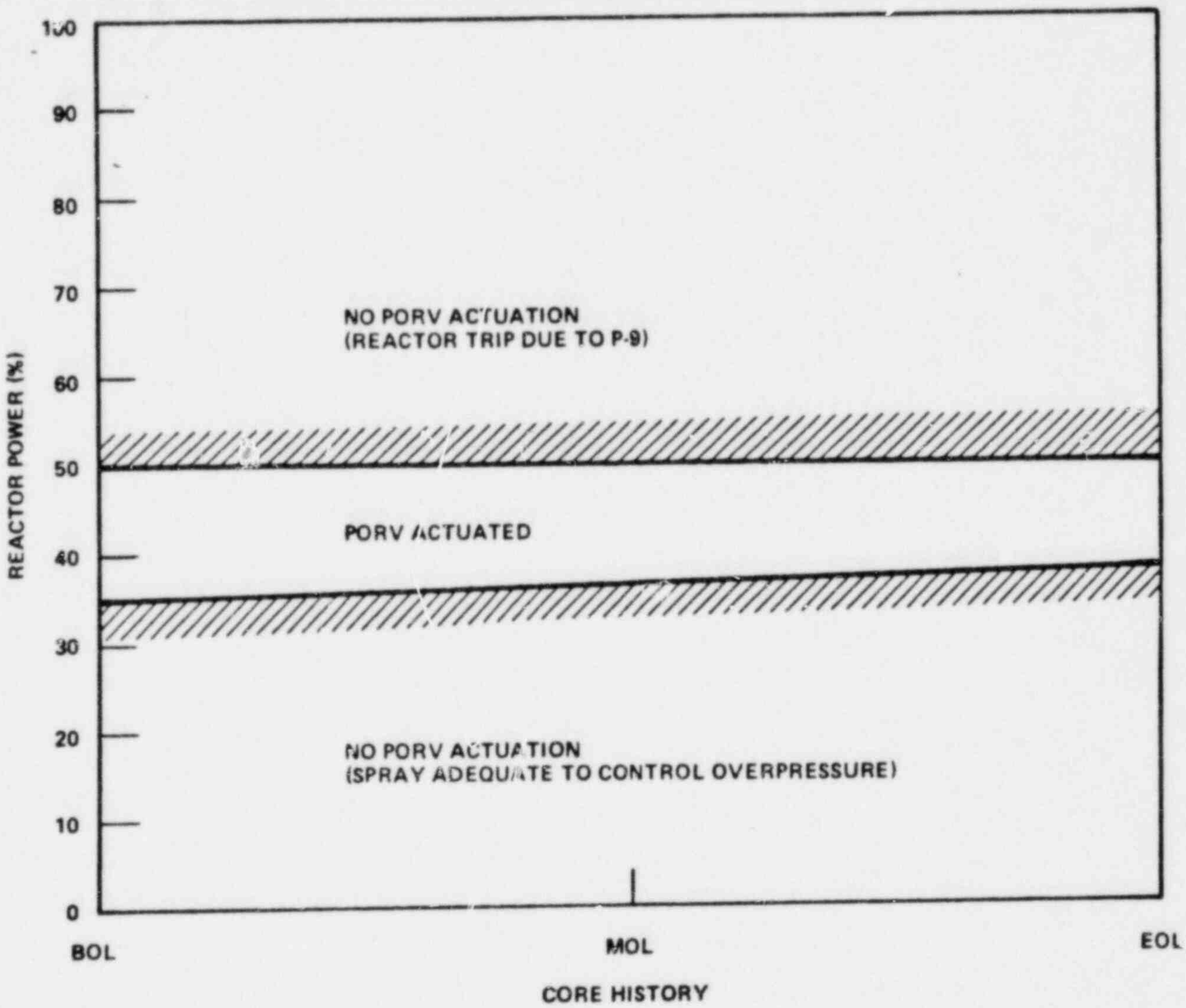
An additional analysis has been performed to determine the consequences (specifically the likelihood of the pressurizer power-operated relief valves opening) of having a turbine trip due to a loss of condenser vacuum.

The major difference between this [redacted] analysis and the one presented above [redacted] is that now the normal steam dump system is unavailable, and the steam relief must be done through the atmospheric relief valves. Since there is a longer delay time before the atmospheric reliefs reach their setpoint (in comparison to the normal steam dump system) and their capacity is about one-half of the steam dump system, there is an increased likelihood that the pressurizer PORVs will open.

Figure 1 shows the plant operating ranges for which the pressurizer PORVs will open for a turbine trip due to a loss of condenser signal. Above 50% power, a turbine trip will cause a reactor trip (due to P-9 setpoint), and the pressurizer PORV setpoint will not be reached. Below a power level of 35%-40% (depending on fuel burnup), the pressurizer spray rate is adequate to maintain the pressurizer pressure below the setpoint. Therefore, only in the narrow band between about 35% and 50% power will the pressurizer PORVs open for a loss of condenser.

Based upon the operating history of current plants, the chances of getting a condenser unavailable signal (and hence a turbine trip) is about 156 out of 10^7 operating hours. [redacted] Assuming 98% plant availability and a 40-year plant lifetime, this works out to about 4 condenser unavailable turbine trips occurring during the normal life of a plant. Assuming an equal chance of having the plant operate anywhere between 0% and 100% power (an unrealistic value, since they usually operate either at a full or no load level), the chances of having a condenser unavailable signal generate a transient which would result in the opening of the pressurizer PORVs is less than one per plant lifetime.

19980-1



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PORV Opening Band – Turbine Trip With Condenser Unavailable

Figure 1.

ENCLOSURE C TO SLNRC 81-82

LEVEL MEASUREMENT ERRORS RESULTING FROM POST-ACCIDENT ENVIRONMENTAL EFFECTS ON LEVEL INSTRUMENT REFERENCE LEGS

1. System Descriptions

The following liquid level measuring systems inside containment are used to initiate safety actions.

a. Steam Generator Narrow Range Water Level

The steam generator narrow range water level detection system consists of four differential pressure measurement channels per steam generator, each with an open column reference leg, a condensing pot to ensure that the reference leg maintains a constant level, and a pressure transmitter. The upper taps are located at a reference elevation of 581 in. (approximate distance above tube sheet) and the lower taps at a reference elevation of 453.25 in. The vertical distance between taps is 127.75 in. The top of the tube bundle is approximately 90 in. below the lower taps. The feedring is approximately 22 in. above the lower taps. The differential pressure transmitters and the level indicators are Class IE and are qualified both to the post-accident environment and an SSE.

The SG narrow range water level is used for the following safety functions:

- Turbine trip and feedwater isolation on high-high SG water level.
- Reactor trip on low-low SG water level.
- Auxiliary feedwater initiation on low-low SG water level.
- Post-accident monitoring. In Emergency Operating Procedures narrow range water level is the basis for manual control of auxiliary feedwater flow to intact steam generators for pipe breaks inside containment. It is also an alternate basis (to auxiliary feedwater flow) for termination of safety injection.

b. Steam Generator Wide Range Water Level

Each steam generator has one wide-range differential pressure measurement channel consisting of an open column reference leg with a

condensing pot, to ensure that the leg maintains a constant level, and a pressure transmitter. The upper tap is located at a reference elevation of 58? in. and the lower tap at a reference elevation of 22 in. The vertical distance between taps is 559 in. The differential pressure transmitters and the level indicators are Class IE and are qualified for both the post-accident environment and an SSE.

The SG wide range water level is used for the following safety functions:

- Post-accident monitoring. SG level indication for pipe breaks outside containment. For pipe breaks inside containment, it has no specific function in Emergency Operating Procedures.

c. Pressurizer Water Level

The pressurizer has three channels of differential pressure measurement, each with a differential pressure transmitter and an open column reference leg with a condensing pot. The span is from the bottom of the pressurizer to the top of the straight shell. The differential pressure transmitters and the level-indicators are Class IE and are qualified for the post-accident environment or an SSE.

The pressurizer water level is used for the following safety functions:

- Reactor trip on high water level.
- Post-accident monitoring. Termination or throttling of safety injection flow.

2. Safety Function Setpoints

a. SG Narrow Range Water Level

The programmed operating water level in the steam generators is at 50% of the narrow range span at all power levels. This water level is maintained by an automatic control system that functions from 0 to 100% power, whenever at least one of the feedwater pumps (either

the small motor er pump or one or two large turbine-driven pumps) is operating. The control system is capable of maintaining level within 5% of the programmed level, that is, within 45 to 55% of narrow range span.

Neglecting, for the moment, the effect of post-accident conditions on the SG reference legs, the high-high setpoint would be at 78.1% of narrow range and the low-low setpoint at 17.2% of narrow range. The potential errors in these setpoints are $\pm$13.2%, which consists of a $\pm$10% error allowance for post-accident effects (radiation and temperature) on the transmitter and $\pm$3.2% statistical combination of other errors. Thus the possible lower limit of actual low-low trips would be 4.0% of narrow range span. Similarly the upper limit of high-high trips would be 91.3% of narrow range span. This would result in the following margins:

Low-low level trip $\geq$ 4% above the bottom of the narrow range span.
Low-low level trip $\geq$ 24.6% below extreme expected swings in operating level.
High-high level trip $\geq$ 19.9% above extreme expected swings in operating level.

b. Pressurizer Water Level

The programmed operating water level in the pressurizer varies from 25 to 60% of span as the power level varies from 0 to 100%. This water level is maintained by an automatic control system. The setpoint for the high water level trip is at 92% of span.

3. Effect of Post-Accident Conditions on Reference Leg Level Indication Systems

a. Reference Leg Heatup

High energy line breaks inside containment can result in heatup of level measurement reference legs. Increased reference leg water column temperature results in a decrease of the water column density

with a consequent apparent increase in the indicated water level (i.e., apparent level exceeding actual level).

The following formula can be used to calculate the magnitude of this bias:

$$E = \frac{H_L}{H} \quad \frac{(\rho_L,\mathrm{cal} - \rho_L)}{(\rho_f,\mathrm{cal} - \rho_g,\mathrm{cal})}$$

Where

E	=	level error due to reference leg heatup, as a fraction of level span
H	=	level span = vertical distance between pressure taps
H_L	=	height of reference leg (maximum vertical distance from lower tap to water level in condensing pot on upper tap)
ρ_L,cal	=	water density at containment temperature and steam generator or pressurizer pressure for which the level indication system was calibrated
ρ_L	=	water density in reference leg at the time of interest
(ρ_f,cal-ρ_g,cal)	=	difference between saturated water density and dry saturated steam density a+ e

steam generator or pressurizer pressure for which the level indication system was calibrated.

This procedure is based on the assumption that the tubing from the upper and lower taps, below the elevation of the lower tap, have the same temperature at all times.

Figure 1 shows the level bias as a function of reference leg temperature, assuming $H_{L/H}$ = 1.1 and the calibration conditions of: containment temperature = 90^{0}F, steam generator pressure = 1000 psia. These are conservative values applicable to the SNUPPS plants.

b. Reference Leg Boiling

In addition to reference leg density change under subcooled conditions, boiling could conceivably occur in the reference leg following depressurization of any steam generator with high-containment temperature. This combination of conditions could occur only after a steamline or feedline rupture inside containment. If such boiling were to occur, it could cause a major bias in the indicated level for a short time period, in the extreme case indicating 100 percent level when the vessel is actually empty.

Containment analyses performed by Westinghouse indicate that such boiling would not occur.

c. Coolant Density Changes

A bias in indicated water level may also be introduced by changes in pressurizer or steam generator pressure, due to changes in the density of the saturated water and steam within those vessels. While prediction of the effects of rapid depressurization requires complex calculations for each specific case, the bias which would exist at low power under quiescent conditions can be calculated directly, using the following formula.

$$E = \frac{H_L}{H}\frac{(\rho_L,cal - \rho L - \rho_g,cal + \rho_g)}{(\rho_f\ ,\ cal - \rho_g,cal)} +$$

$$\frac{L}{H}\frac{(\rho_f - \rho_g)}{(\rho_f,cal - \rho_g,cal)} - \frac{L}{H}$$

Where

E	=	level error due to density changes in both the vessel and the reference leg, as a fraction of level span
L	=	true water level in the vessel, above the lower level tap
ρ_f	=	saturated water density at the pressure of interest
ρ_g	=	dry saturated steam density at the pressure of interest, and other symbols have the same meaning as in Section 2A.

For an example, Figure 2 shows the true water level as a function of steam generator pressure and indicated level, assuming the following calibration conditions: Containment temperature = 90^oF, steam generator pressure = 1000 psia, and the reference leg is at 90^oF. Figure 3 is similarly calculated for a pressurizer, with the assumptions noted on the Figure.

4. Description and Evaluation of Planned Modifications to Water Level Measurement Systems

a. Steam Generator Narrow Range Water Level Trips

The low-low setpoint will be raised 11% of narrow range span to compensate for the effects of reference leg heatup. The bases for this change are analyses specific to the SNUPPS plants.

Low-low steam generator water level is the primary means of tripping the reactor for a feedwater line break. This trip is backed-up by the high-1 containment pressure trip, which trips the reactor by initiating a safety injection signal. The primary means of tripping the reactor for a steam-line break is low steam-line pressure. Diverse (back-up) trips for a steam-line break are high steam-line negative pressure rate, and the high-1 containment pressure trip. Low-low steam generator water level may under certain circumstances also be a diverse trip for a steam-line break, but an increase in feedwater flow could prevent this trip from occuring in the event of a steam-line break.

Calculations have been performed for a wide spectrum of feedwater line breaks inside containment. For a feedwater line break postulated upstream of the feedline check valve in a feedwater line, feedwater supplied by the feed train, at a maximum temperature of $420^{O}F$, is released to containment. At the time that the actual water level in the affected SG reaches 4% of narrow range span (conservatively assumed to correspond to the low-low setpoint) the containment temperature is less than $150^{O}F$ for any break size or operating power level.

For a feedwater line break postulated between the feedline check valve and the steam generator at full power, initially a 50-50 mixture of $420^{O}F$ feedwater and saturated water at 1000 psi ($545^{O}F$) blows down to containment. When the feedring becomes uncovered, saturated steam from the affected steam generator blows down to containment. At the time the actual water level in the affected

steam generator reaches 4% of narrow range span (assumed to correspond to the trip setpoint, as above) the containment temperature is less than 200°F for any break size.

For a feedwater line break postulated between the feedline check valve and the steam generator at zero power, initially saturated water at 1100 psia (556°F) blows down to containment. When the feedring becomes uncovered, saturated steam from all four steam generators blows down to containment. At the time the actual water level in the affected steam generator reaches 4% of narrow range span the containment temperature is 292°F.

The reference leg heatup error for 292°F containment temperature is 11% of narrow level span from Figure 1.

For a steam-line break inside containment, low-low SG is only a weakly diverse trip as previously discussed. The nominal setpoint for the high-1 containment pressure trip will be 3.1 psig and the potential error is ±1.9 psi. Assuming release of 1000 psia saturated steam to containment, the containment temperature at 5 psig is 200°F.

Raising the steam generator narrow range low-low setpoint by 11% of narrow range span will preserve the margin of actual water level ≥ 4% above bottom of narrow range span at the time of low-low level trip for any feedwater line break and although the margin between the trip setpoint and operating water level will be reduced, the discussion of Section 2.a, above, shows that spurious trips are unlikely to occur.

The high-high SG water level trip is not required for accident situations that could cause significant errors in level indication. The setpoint of this trip will remain unchanged.

b. Pressurizer Water Level Trip

The pressurizer level trip is not required for accident situations that could cause significant errors in level indication. The setpoint of this trip will remain unchanged.

c. Post-Accident Monitoring - SG Level

The standard emergency operating instructions (EOI's) for Westinghouse reactors require use of the narrow range level indication for pipe breaks inside containment or steam generator tube ruptures. Narrow range level is used as the basis for throttling auxiliary feedwater flow, but only after the affected steam generator has been identified. After the affected SG is identified and isolated, the EOI's instruct the operator to maintain SG indicated level within the narrow range. If the water level is below this range, the operator need not know the level accurately because his objective will be to obtain the maximum possible auxiliary feedwater flow.

The presence of steam-generator level within the narrow-range span is also one of the criteria (along with pressurizer pressure, pressurizer level, subcooling and auxiliary feedwater flow) for throttling safety injection flow. The EOI's require either steam-generator level to be within the narrow-range or auxiliary feedwater flow rate to exceed a specified rate. The purpose is to keep the tube bundle covered or to limit the duration of any uncovery of the tube bundle.

Potential errors in narrow-range level may amount to approximately 40% of span in the post-accident monitoring mode. That is, the actual level could be 40% below the indicated level. Since the SG tube bundle is approximately 70% of narrow range span below the bottom narrow range tap, if the indicated water level is maintained within the narrow range, the tube bundle will always be covered. This is a conservative basis for ensuring that heat removal via the steam generators and natural circulation capability within the reactor coolant loops exist.

When the indicated water level is below the narrow range span or for steam/feed-line breaks outside containment, the wide range level indication will be used by the operators. There is no need for the operators to know the precise level. They will know from other indications if the heat removal function of the steam generators has been lost and will have emergency instructions that will relate the potential error in level indication to containment temperature.

d. Post-Accident Monitoring - Pressurizer Level

The standard emergency operating instructions (EOI's) for Westinghouse reactors require use of the pressurizer level indication to throttle and/or terminate safety injection following a high energy line break, after which the pressurizer level is restored.

Reference leg heatup effects could introduce an error in level indication of 40%, with the actual level being lower than the indicated level. The operators will be provided with operating instructions to cover this phenomenon.

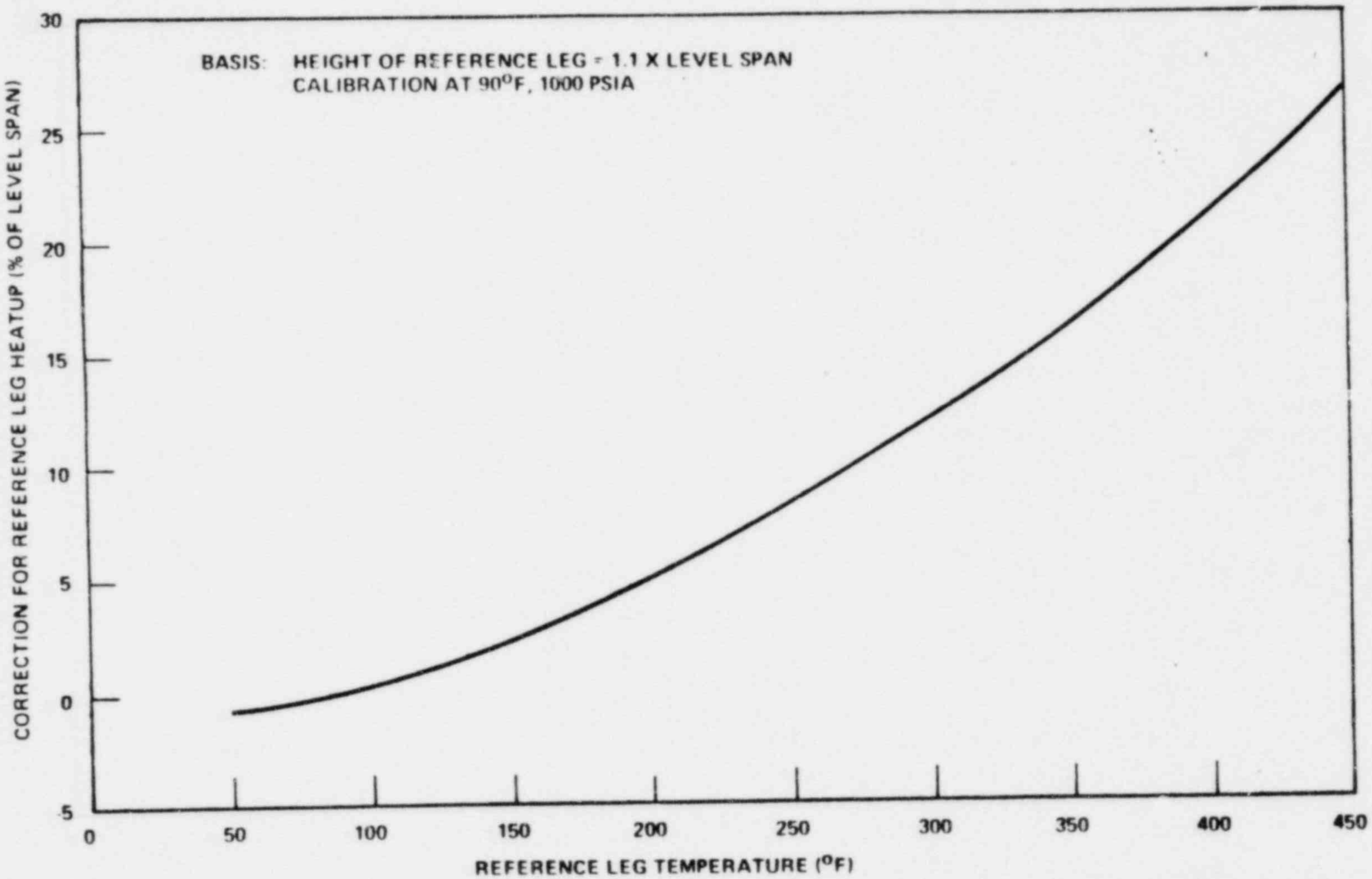

Figure 1

Bias Due to Steam Generator Reference Leg Heatup

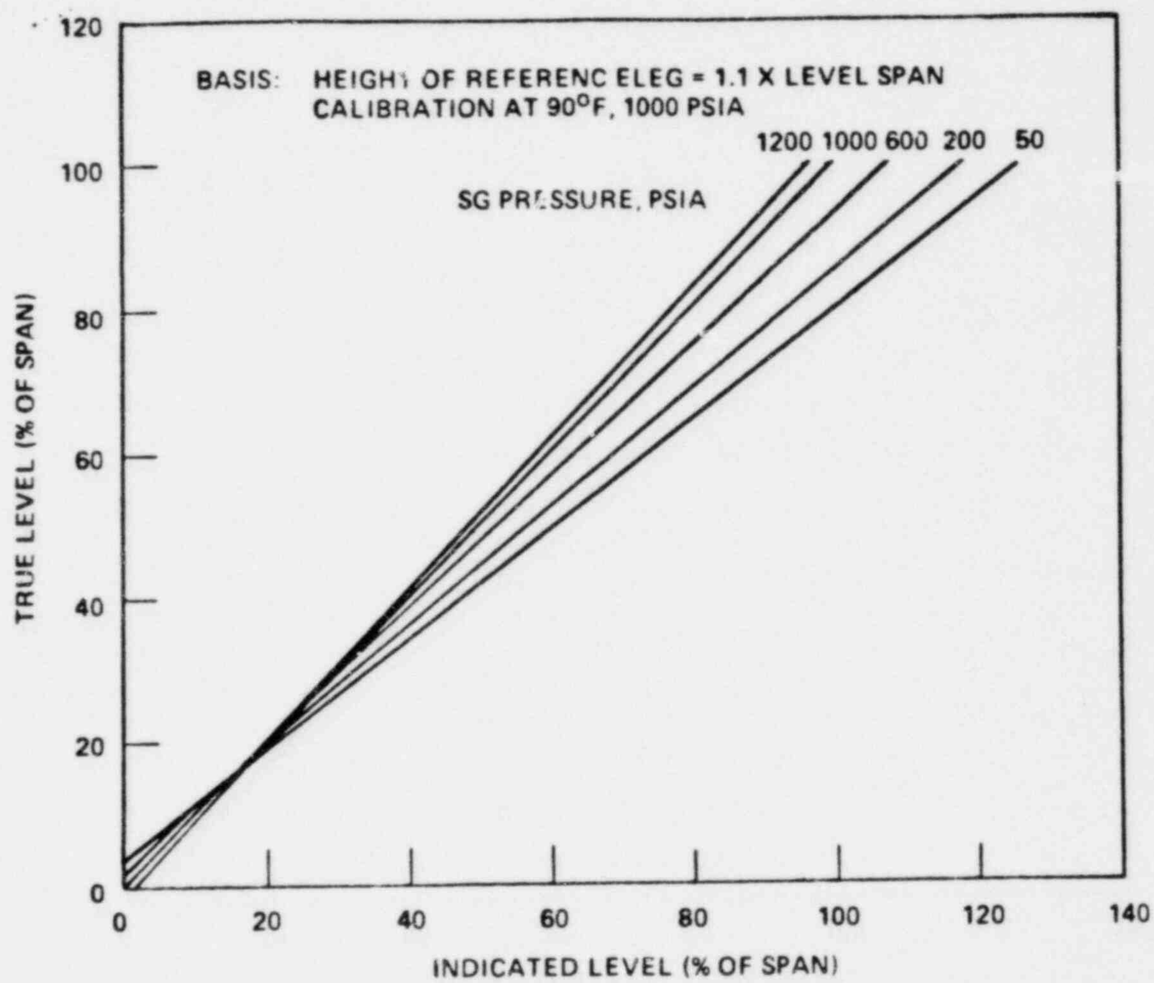

Figure 2

Bias Due to Steam Generator Pressure Change

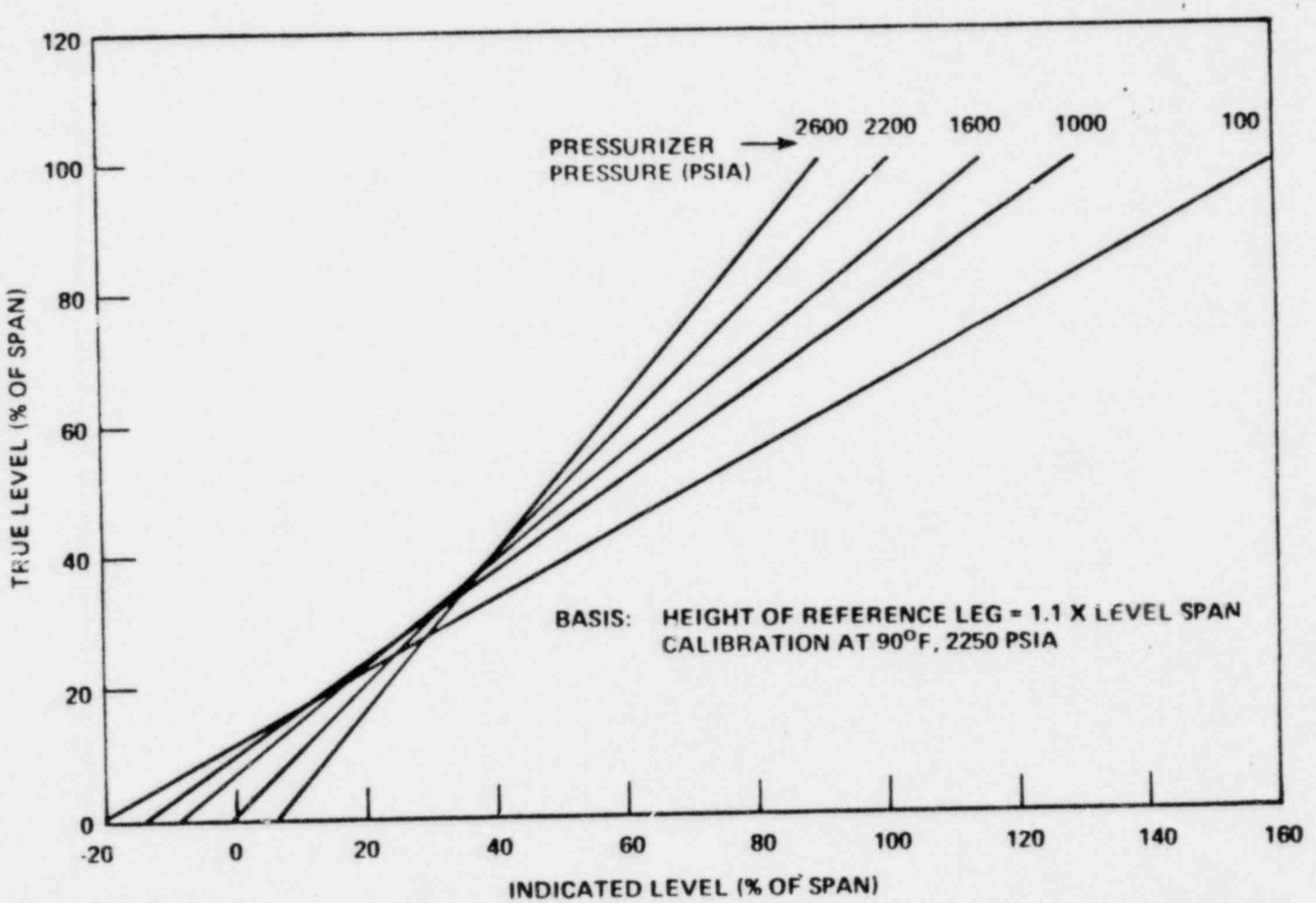

Figure 3

Bias Due to Pressurizer Pressure Change