FINAL SUBMITTAL

BRUNSWICK EXAM
50-325 & 50-324

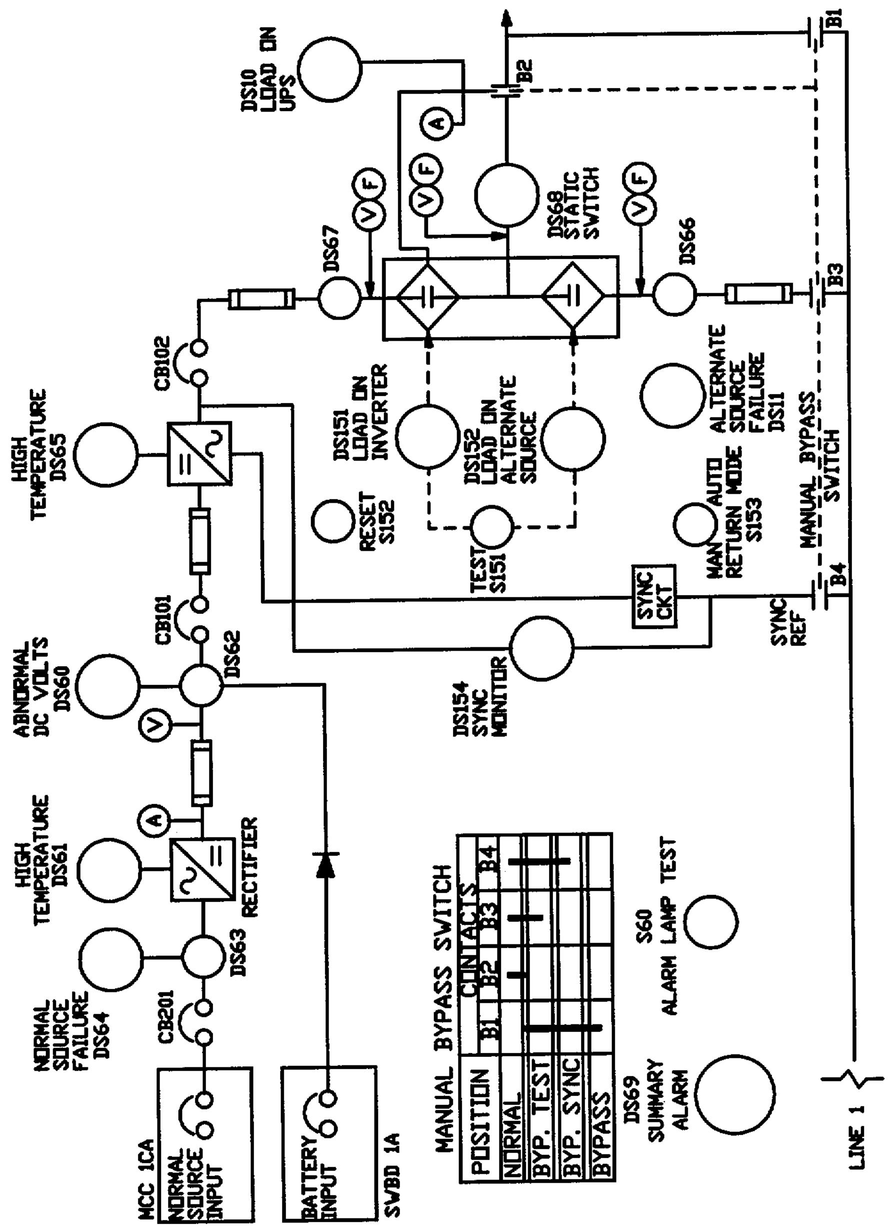
JULY 27 - AUGUST 3, 2001

RO/SRO

FINAL SRO LICENSE EXAM

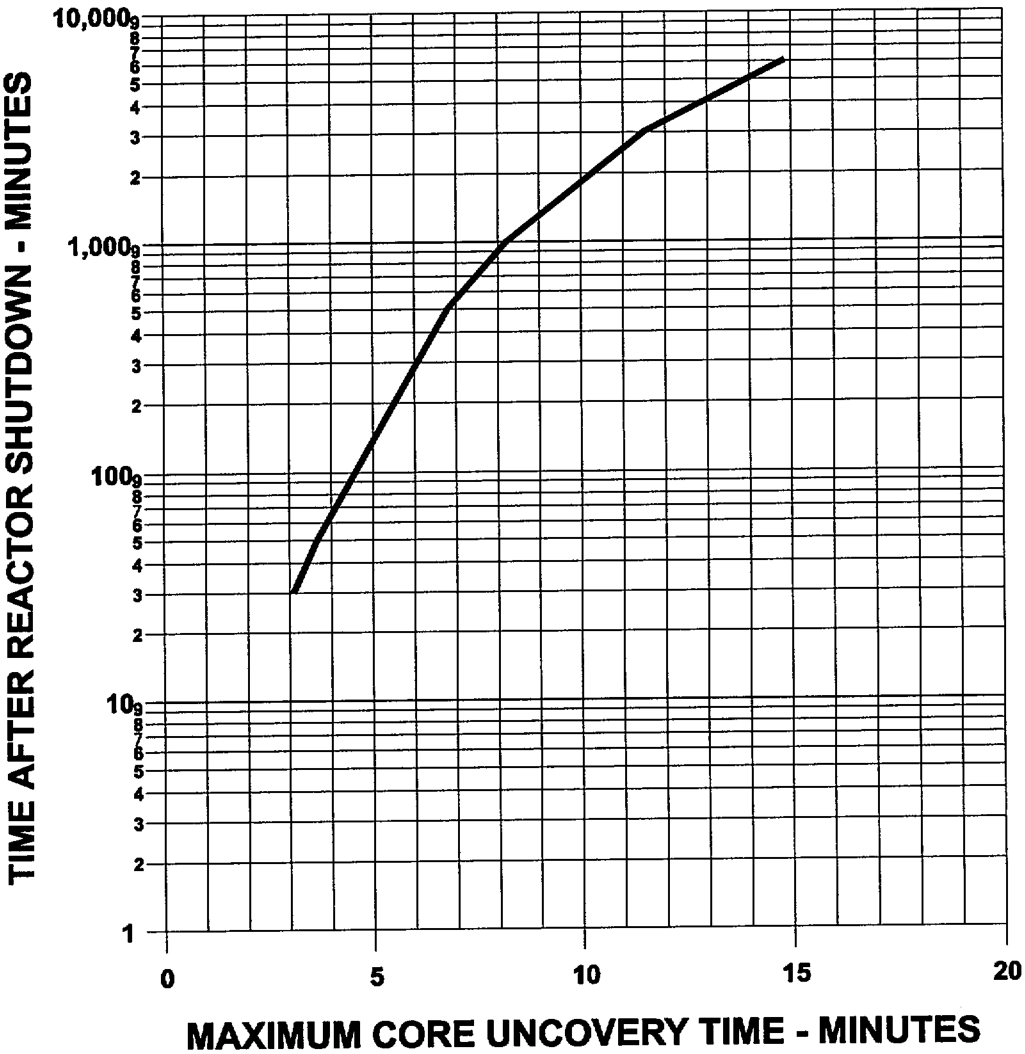
REFERENCE MATERIAL

FIGURE 11
Primary UPS Unit Mimic Bus



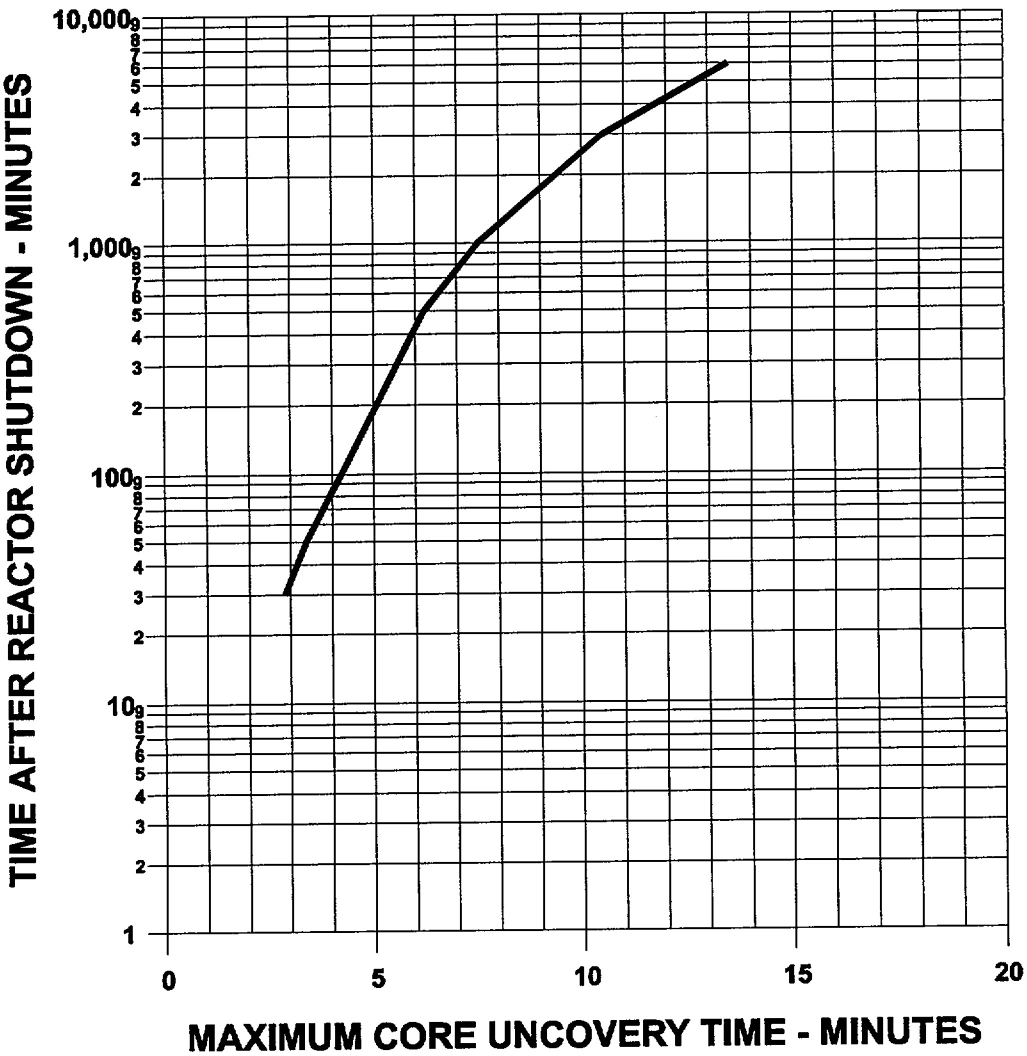
ATTACHMENT 5 (Cont'd)

FIGURE 4
UNIT 1 MAXIMUM CORE UNCOVERY TIME LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 4A
UNIT 2 MAXIMUM CORE UNCOVERY TIME LIMIT



EOP-01-UG
Attachment 6
Reactor Water Level Caution
(Caution 1)

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

If the temperature near any instrument run is in the UNSAFE region of the REACTOR SATURATION LIMIT (Figure 14), the instrument may be unreliable due to boiling in the run.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	Unit 1 only: The indicated level is in the SAFE region of Figure 15. Unit 2 only: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Figure 16. NOTE To determine reactor water level at the Main Steam Line Flood Level (MSL), see Figure 21. NOTE The figure has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, OR B21TA103) AND 2. IF the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 20 inches OR IF the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 10 inches.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. <u>IF</u> the reference leg area drywell temperature is less than 440°F, <u>THEN</u> the indicated level is greater than -150 inches <u>OR</u> <u>IF</u> the reference leg area drywell temperature is greater than or equal to 440°F, <u>THEN</u> the indicated level is greater than -130 inches. <u>AND</u> 2. Reactor Recirculation Pumps are shutdown. <u>NOTE</u> To determine reactor water level at TAF, see <u>Unit 1 only</u>: Figure 17 and <u>Unit 2 only</u>: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see <u>Unit 1 only</u>: Figure 18 and <u>Unit 2 only</u>: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see <u>Unit 1 only</u>: Figure 19 and <u>Unit 2 only</u>: Figure 19A To determine reactor water level at 90 inches, see Figure 20. Continued on next page.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	NOTE Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. NOTE These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 13
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

1. For all Level Instruments EXCEPT B21-LI-R605 A, B, (N027 A, B); the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1B Point 1258-1 ______
CAC-TR-4426-1B Point 1258-3 ______
CAC-TR-4426-2B Point 1258-2 ______
CAC-TR-4426-2B Point 1258-4 ______

OR

Microprocessor

CAC-TY-4426-1 Point 5801 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

ATTACHMENT 6 (Cont'd)

FIGURE 13 (Cont'd)
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

2. For Level Instruments B21-LI-R605A, B (N027A, B), the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1A Point 1258-22 _____
CAC-TR-4426-1B Point 1258-3 _____
CAC-TR-4426-2A Point 1258-23 _____
CAC-TR-4426-2A Point 1258-24 _____
CAC-TR-4426-2B Point 1258-2 _____
CAC-TR-4426-2B Point 1258-4 _____

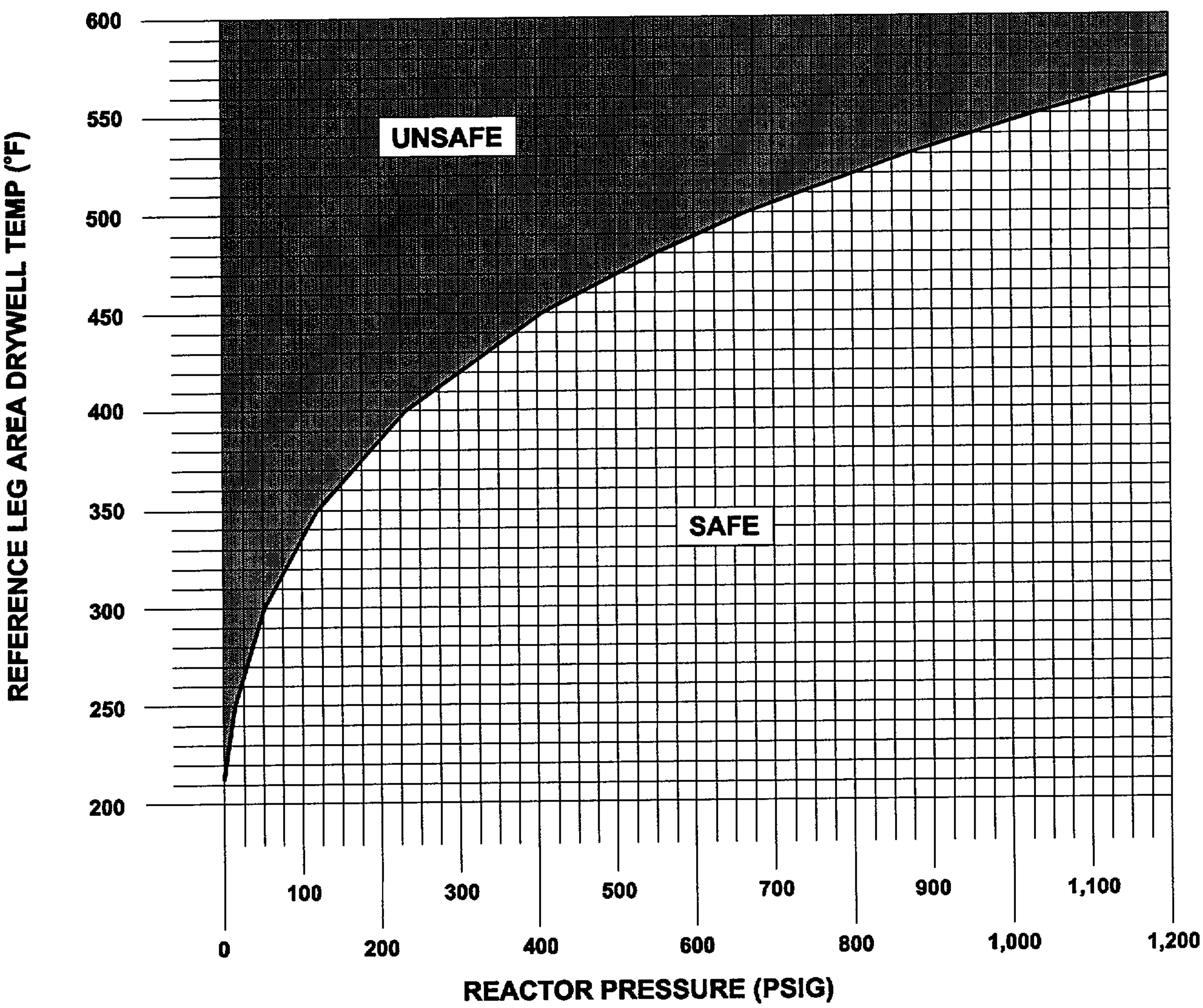
<u>OR</u>

Microprocessor

CAC-TY-4426-1 Point 5822 _____
CAC-TY-4426-1 Point 5803 _____
CAC-TY-4426-2 Point 5823 _____
CAC-TY-4426-2 Point 5824 _____
CAC-TY-4426-2 Point 5802 _____
CAC-TY-4426-2 Point 5804 _____

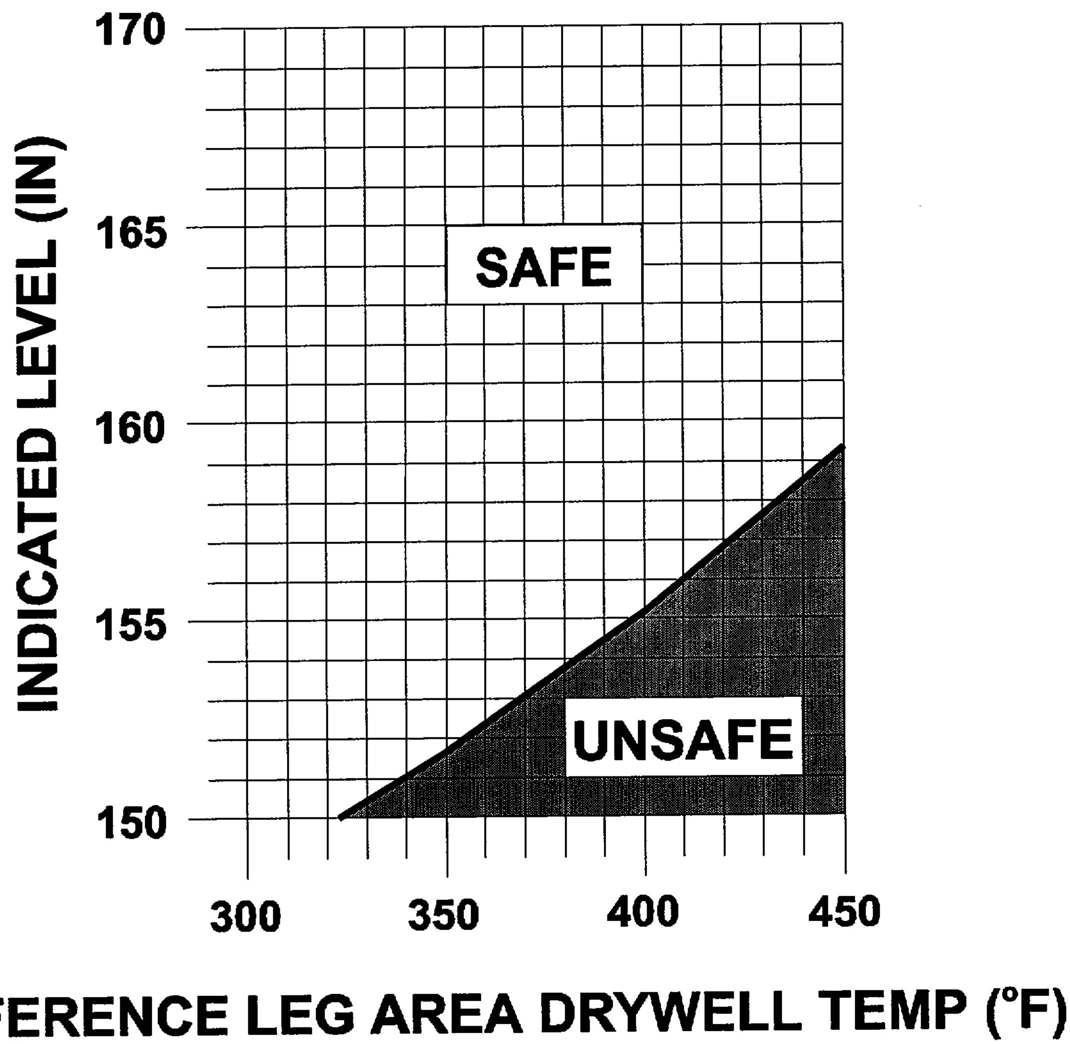
ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



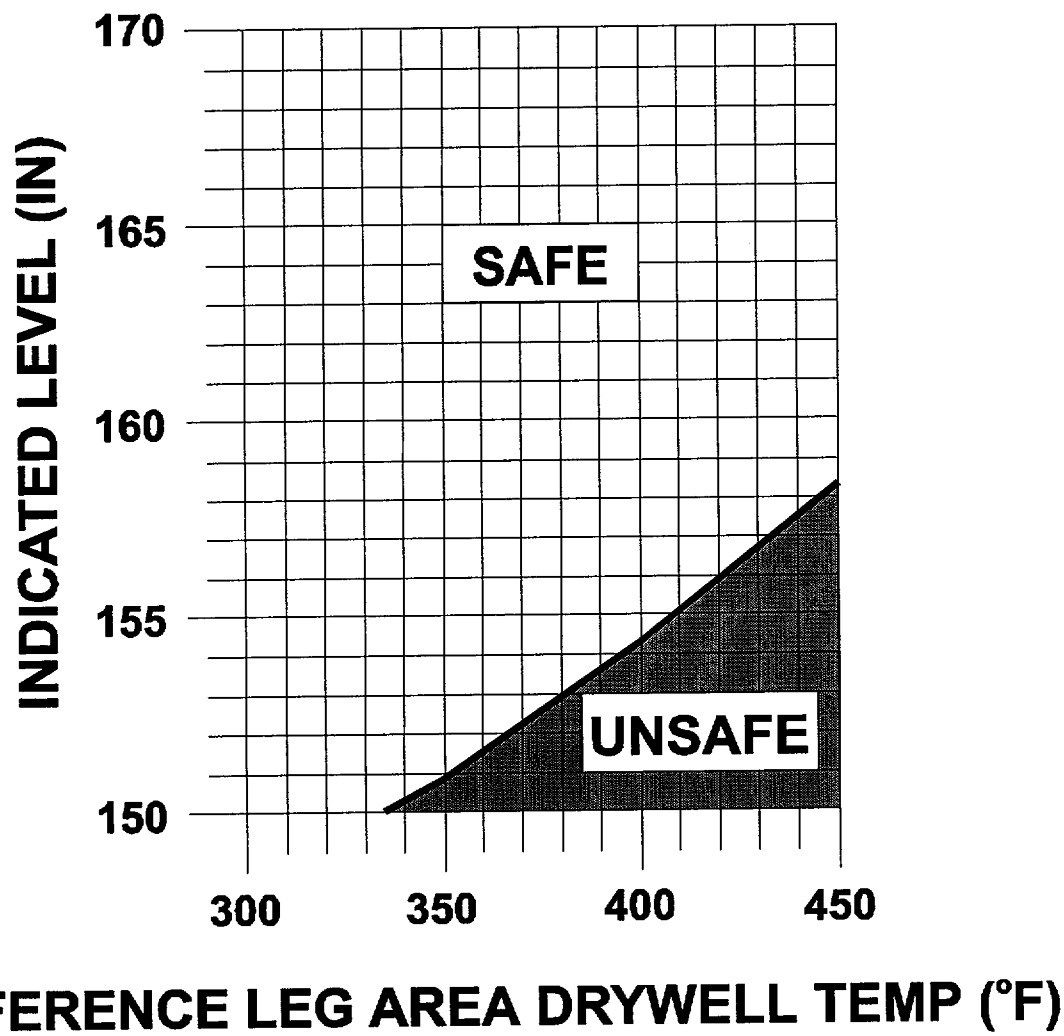
ATTACHMENT 6 (Cont'd)

FIGURE 15
UNIT 1 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



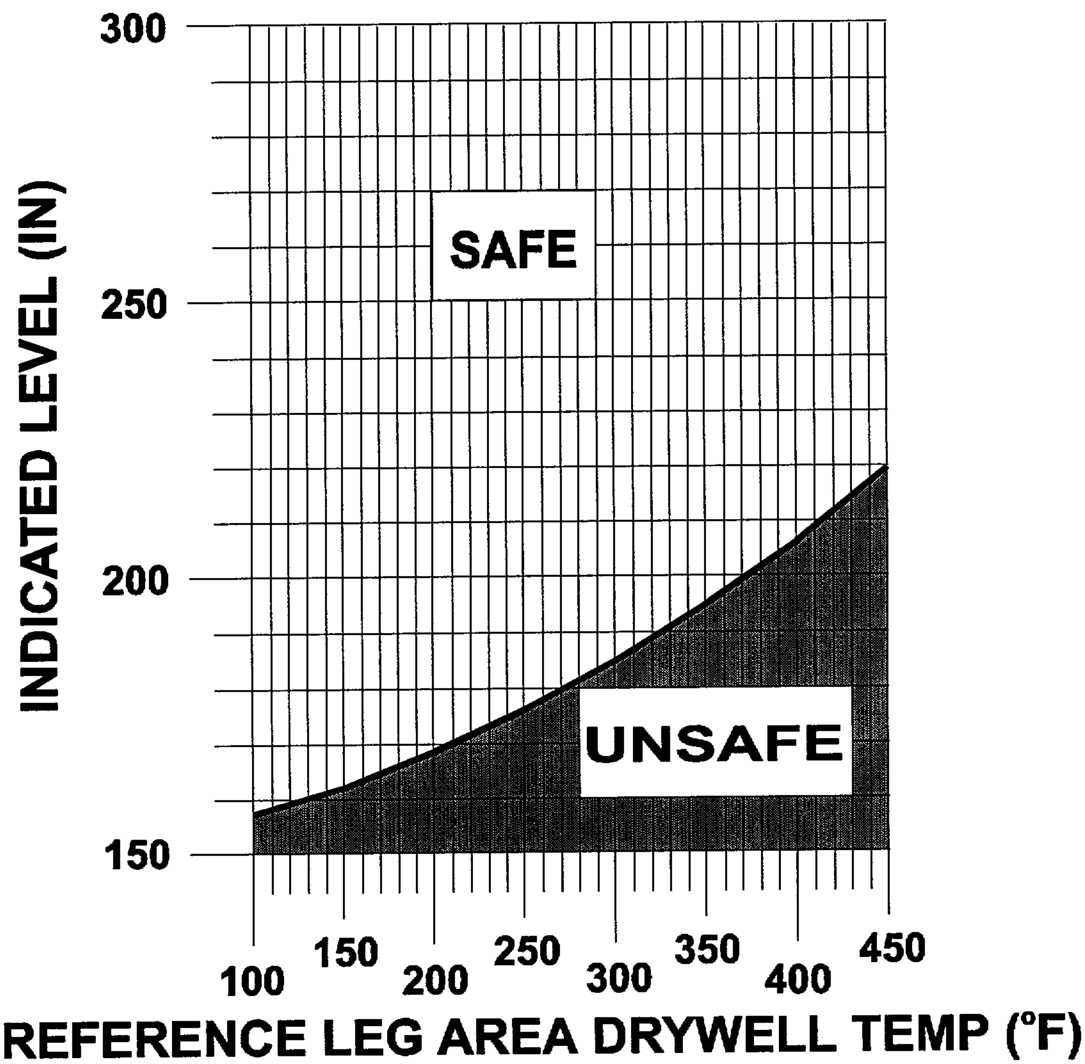
ATTACHMENT 6 (Cont'd)

FIGURE 15A
UNIT 2 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



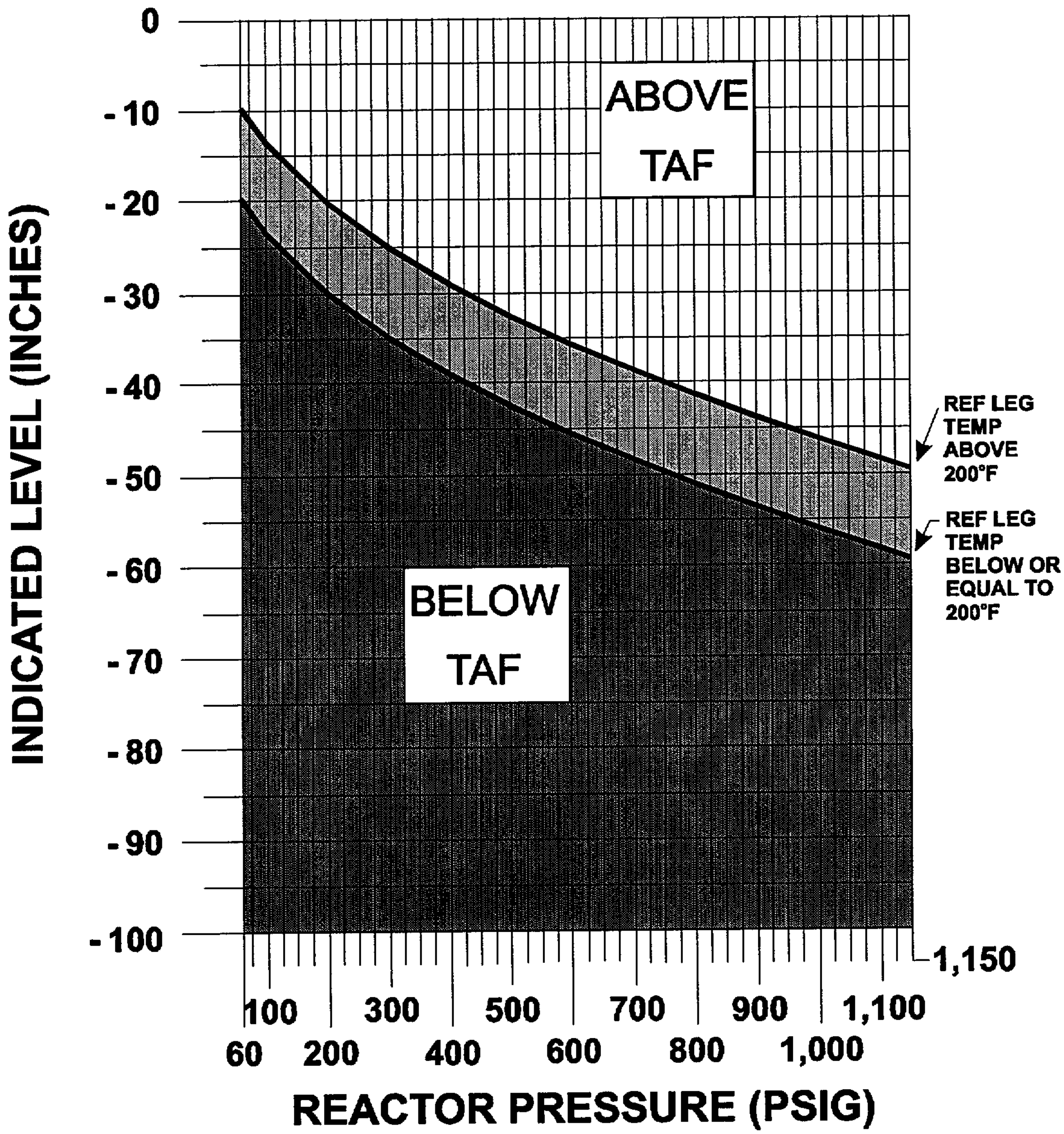
ATTACHMENT 6 (Cont'd)

FIGURE 16
SHUTDOWN RANGE LEVEL
INSTRUMENT (N027A, B) CAUTION



ATTACHMENT 6 (Cont'd)

FIGURE 17
UNIT 1 REACTOR WATER LEVEL AT TAF

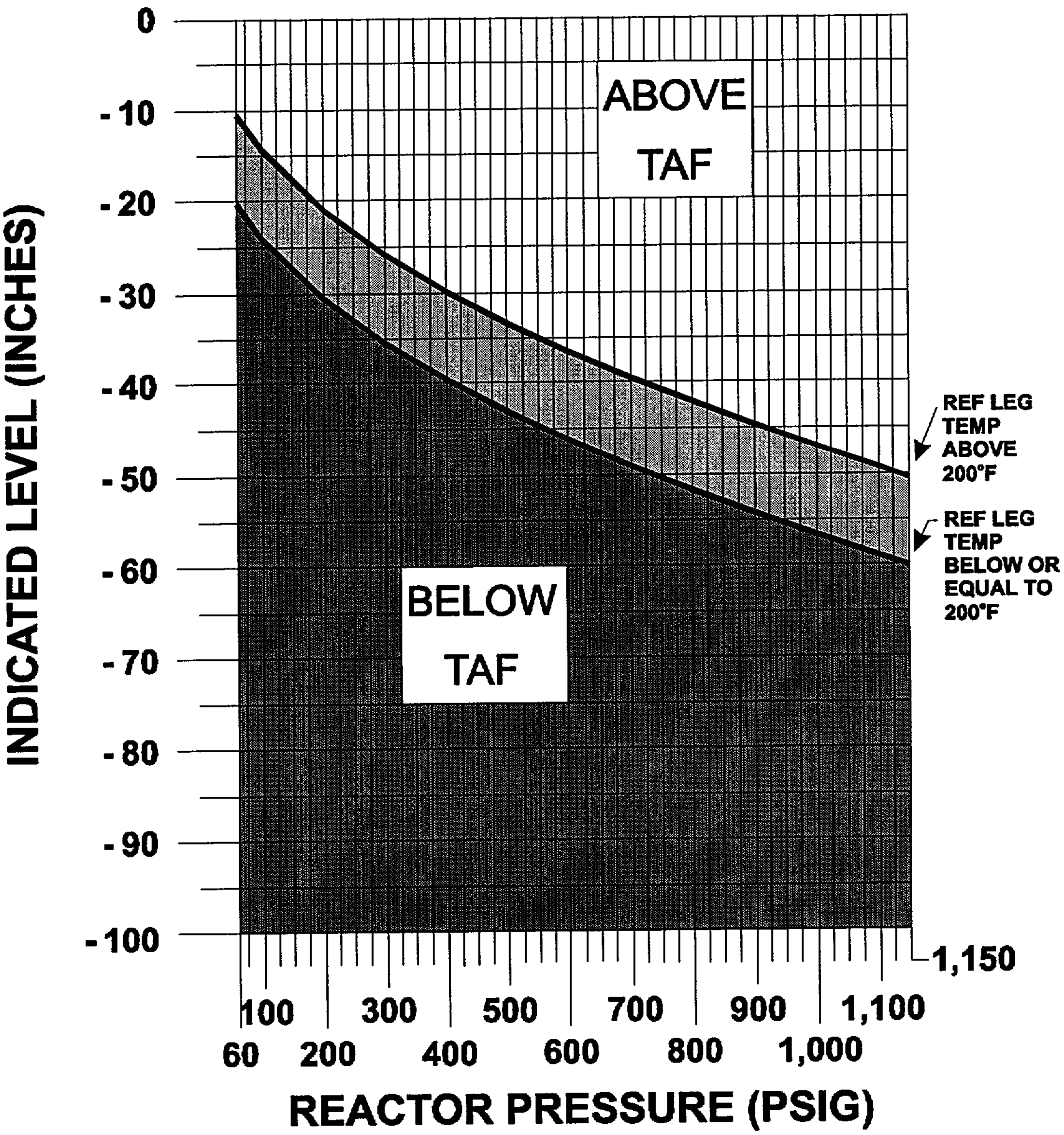


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 17A
UNIT 2 REACTOR WATER LEVEL AT TAF

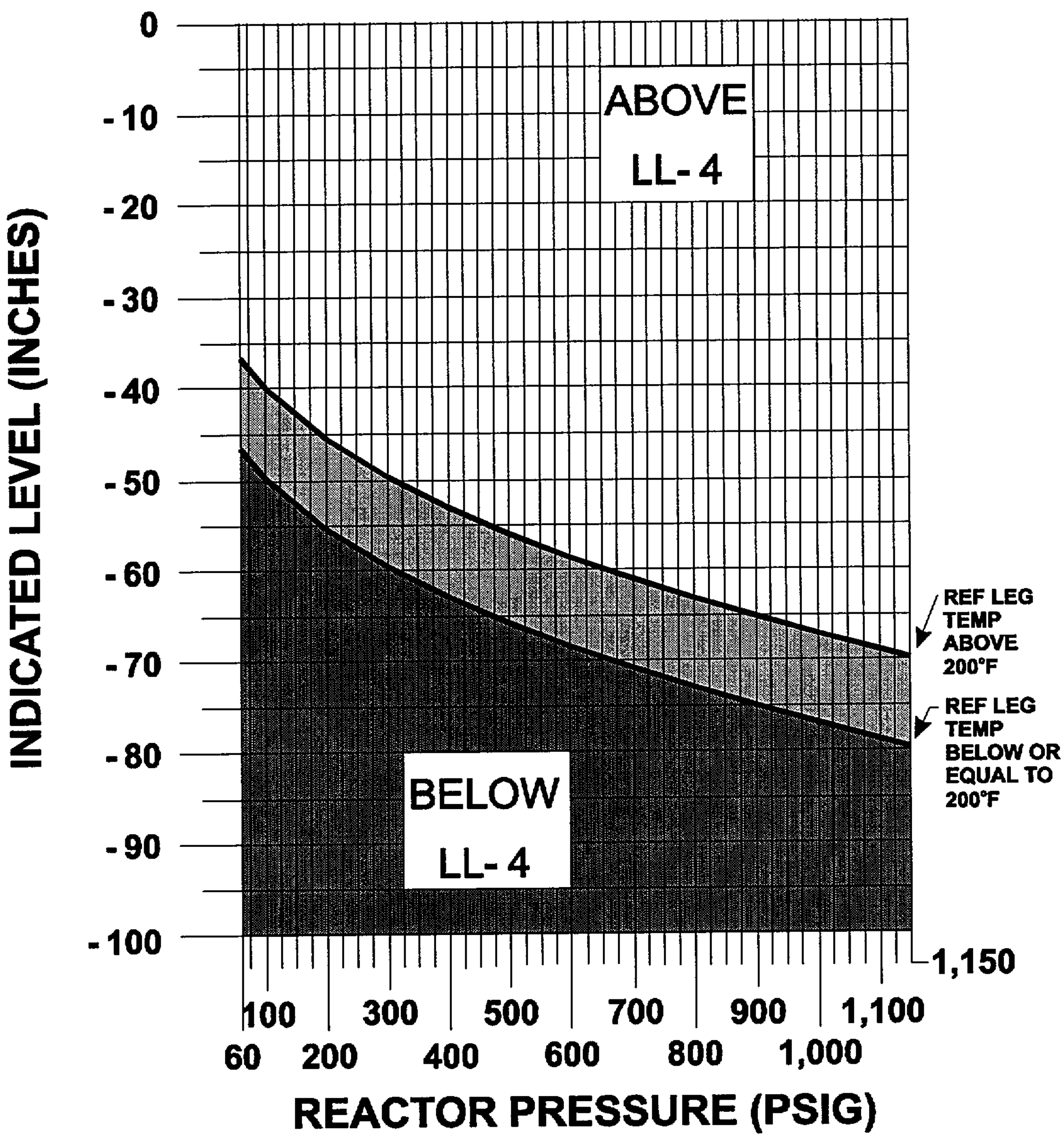


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18
UNIT 1 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

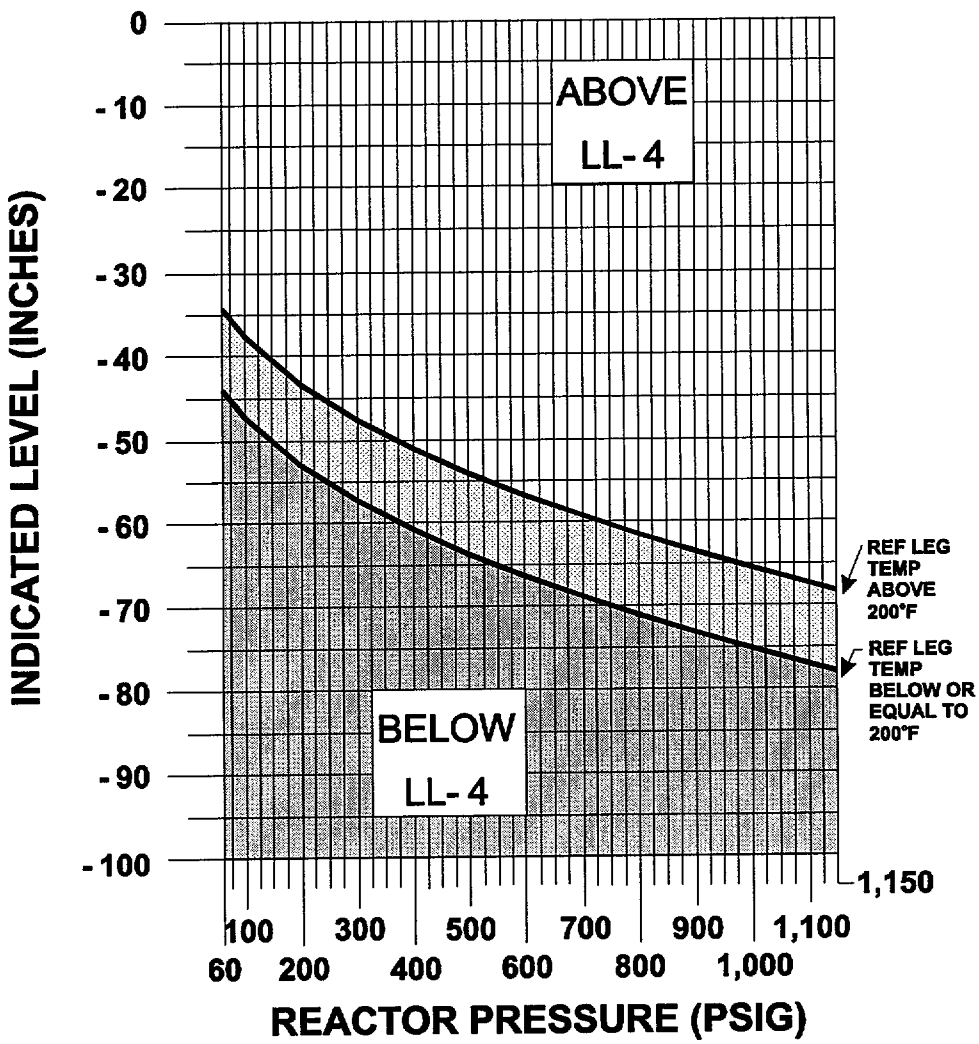


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -35 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18A
UNIT 2 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

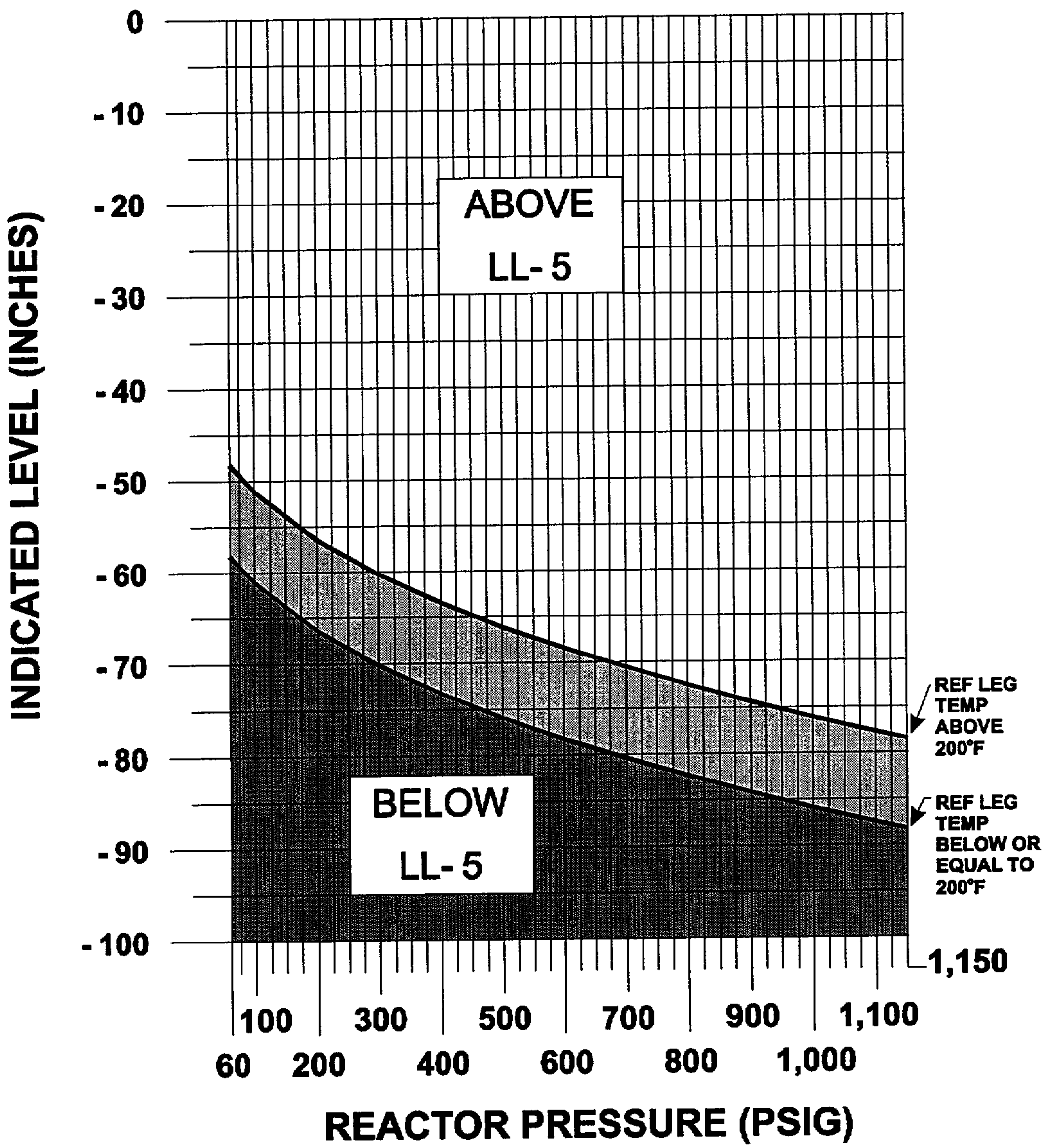


NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL-4 IS -32.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19
UNIT 1 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

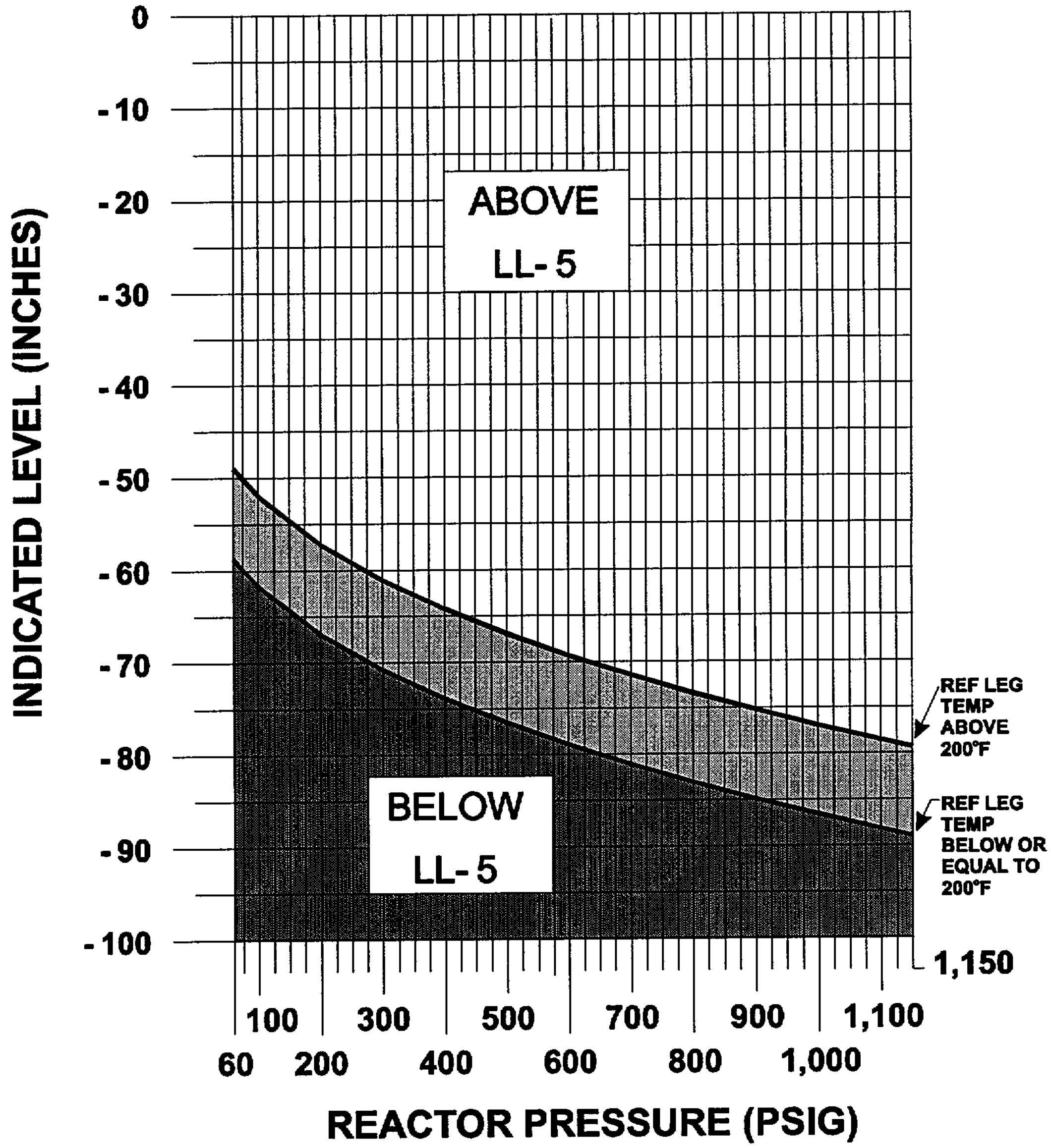


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-5 IS -47.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19A
UNIT 2 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

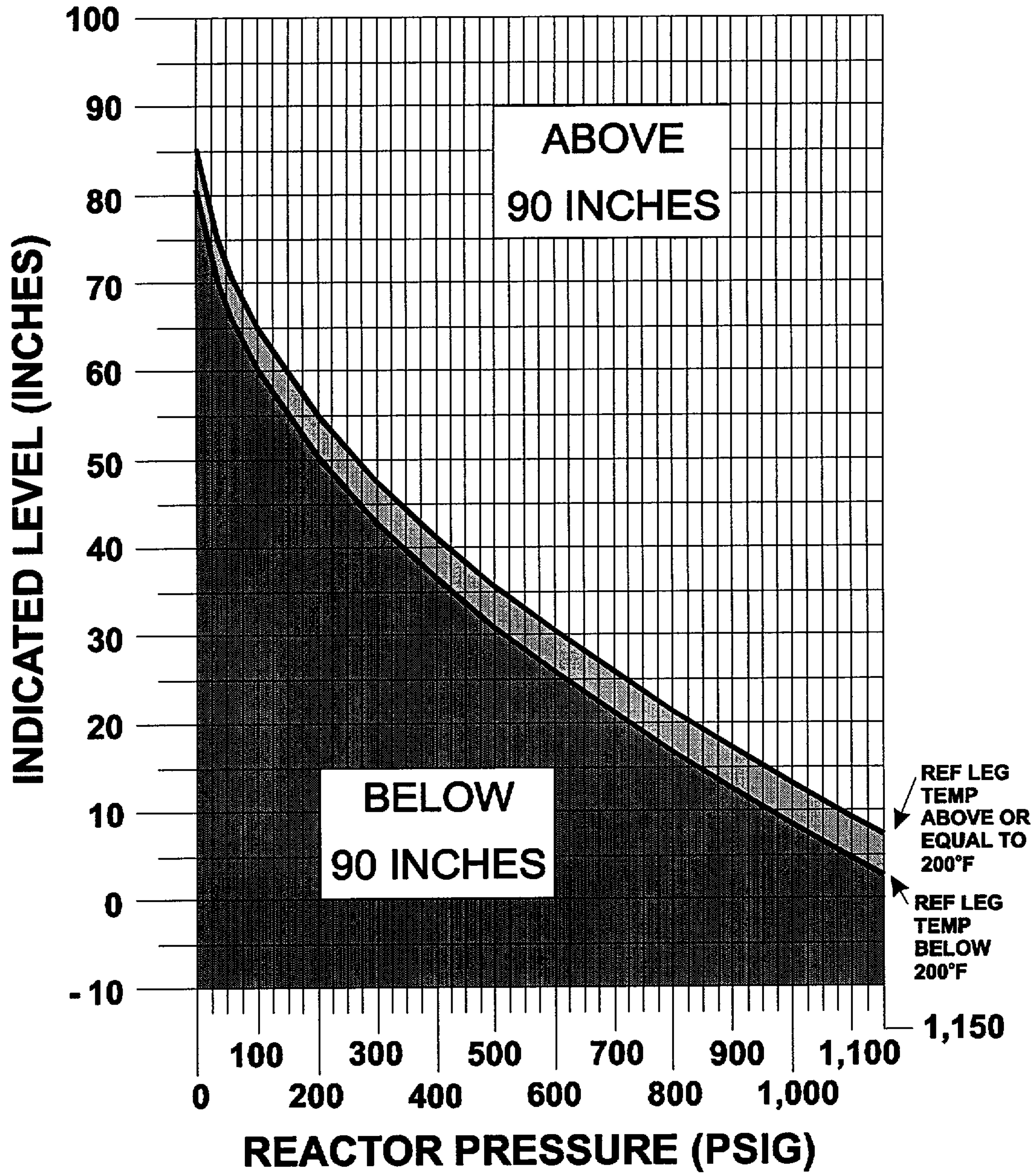


NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL-5 IS -47.5 INCHES.

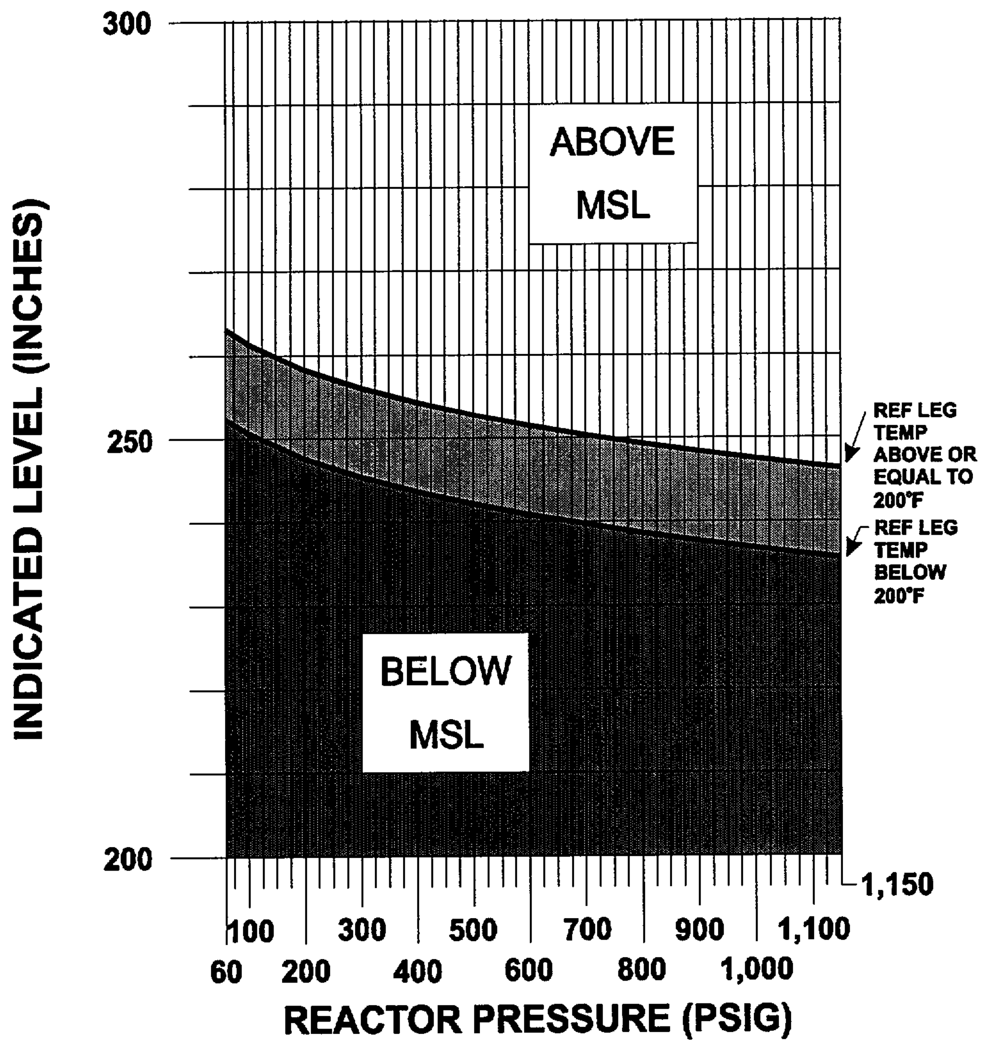
ATTACHMENT 6 (Cont'd)

FIGURE 20
REACTOR WATER LEVEL AT 90 INCHES



ATTACHMENT 6 (Cont'd)

FIGURE 21
REACTOR WATER LEVEL AT MSL
(MAIN STEAM LINE FLOOD LEVEL)



NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
MSL IS +250 INCHES.

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date: ____________

TABLE 1

TIME	APPROX. RX PWR	BYPASS VALVE % PWR	STEAM FLOW % PWR	LPRM % PWR	HEAT BALANCE % PWR	APRM GAFs ≤ 1.0	INIT.
	10%						
	20%						
N/A	TURBINE ON LINE	N/A	N/A	N/A	N/A	N/A	N/A

1. **USE** this attachment to validate the heat balance between 10% reactor power and placing the turbine on line, by performing the following:

   a. **OBTAIN** valid Heat Balance (Display 820, C067, C003, or 0PT-01.8D) **AND RECORD** heat balance % power in Table 1.

   b. **OBTAIN** LPRM % PWR (Display 815) **AND RECORD** in Table 1.

   c. **OBTAIN** Total Steam Flow (Mlb/hr).

Steam Line	(A)	(B)	(C)	(D)
(ERFIS)	C32FA014	C32FA015	C32FA016	C32FA017
(P603)	C32-R603A	C32-R603B	C32-R603C	C32-R603D
	______	______	______	______

Total Steam Flow = (A) + (B) + (C) + (D) = ________

OR

USE Computer Point B041

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:__________

d. **OBTAIN** total bypass valve position (% open):

ERFIS computer point EHCXA002: ________%

OR

RTGB Panel XU1: (record the position (% open) of each)

Unit 1:	MS-BPV-1	MS-BPV-2	MS-BPV-3	MS-BPV-4
	_______%	_______%	_______%	_______%

Total BPV % open = $\frac{\text{(The sum of adding BPV-1 through BPV-4)}}{4}$ = ________%

Unit 2:	MS-BPV-1	MS-BPV-2	MS-BPV-3	MS-BPV-4	MS-BPV-5
	_______%	_______%	_______%	_______%	_______%
	MS-BPV-6	MS-BPV-7	MS-BPV-8	MS-BPV-9	MS-BPV-10
	_______%	_______%	_______%	_______%	_______%

Total BPV % open = $\frac{\text{(The sum of adding BPV-1 through BPV-10)}}{10}$ = ________%

> **NOTE:** Typing MAN runs an interactive program called MAN_ALTDSP that performs alternate power calculations based upon user supplied plant inputs. Remember to enter decimal points for all values. Use the equivalent % power output from this program for the comparison in the next step.

e. **LOG** on to the ERFIS terminal on the SRO's desk:

1) **TYPE:** SET HOST EC01B (EC02B)

OR

SET HOST EC01A (EC02A)

2) **TYPE:** GEPACUSER, at USERNAME prompt

3) **TYPE:** GEPAC, at PASSWORD prompt

4) **TYPE:** MAN (for manual input)

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:____________

> **NOTE:** The equivalent % power output from this program is to be used for the comparison in the next step.

f. **RECORD** BYPASS valve, and STEAM FLOW % power alternate indications on Table 1 of this attachment using the values obtained from MAN_ALTDSP in Step e.

g. **COMPARE** the Heat Balance (%) with the other alternate indications (%) recorded on Table 1.

Power Ascension may continue **IF**:

1) The heat balance is greater than all alternate indications (conservative as is).

OR

2) One or more alternate indications are within ± 5% of the heat balance (normal acceptance).

OR

3) There are no alternate indications within ± 5% of the heat balance provided the APRMs are adjusted greater than or equal to the next highest alternate indication above the heat balance (conservative action) in accordance with 1(2)OP-09.

OTHERWISE, STOP power ascension **AND CONTACT** Reactor Engineering to account for the differences in agreement.

h. **REPEAT** the above steps at 10% increments until the turbine is placed on line.

2. Definitions for Table 1:

HEAT BALANCE % POWER - A calculation of core thermal power obtained by solving an energy balance on the reactor vessel. Valid heat balance calculations may be obtained from Display 820 edits, Process Computer Points C003 and C067, or manually by performing 0PT-01.8D. Caution must be taken to ensure any failed sensors have valid substituted values.

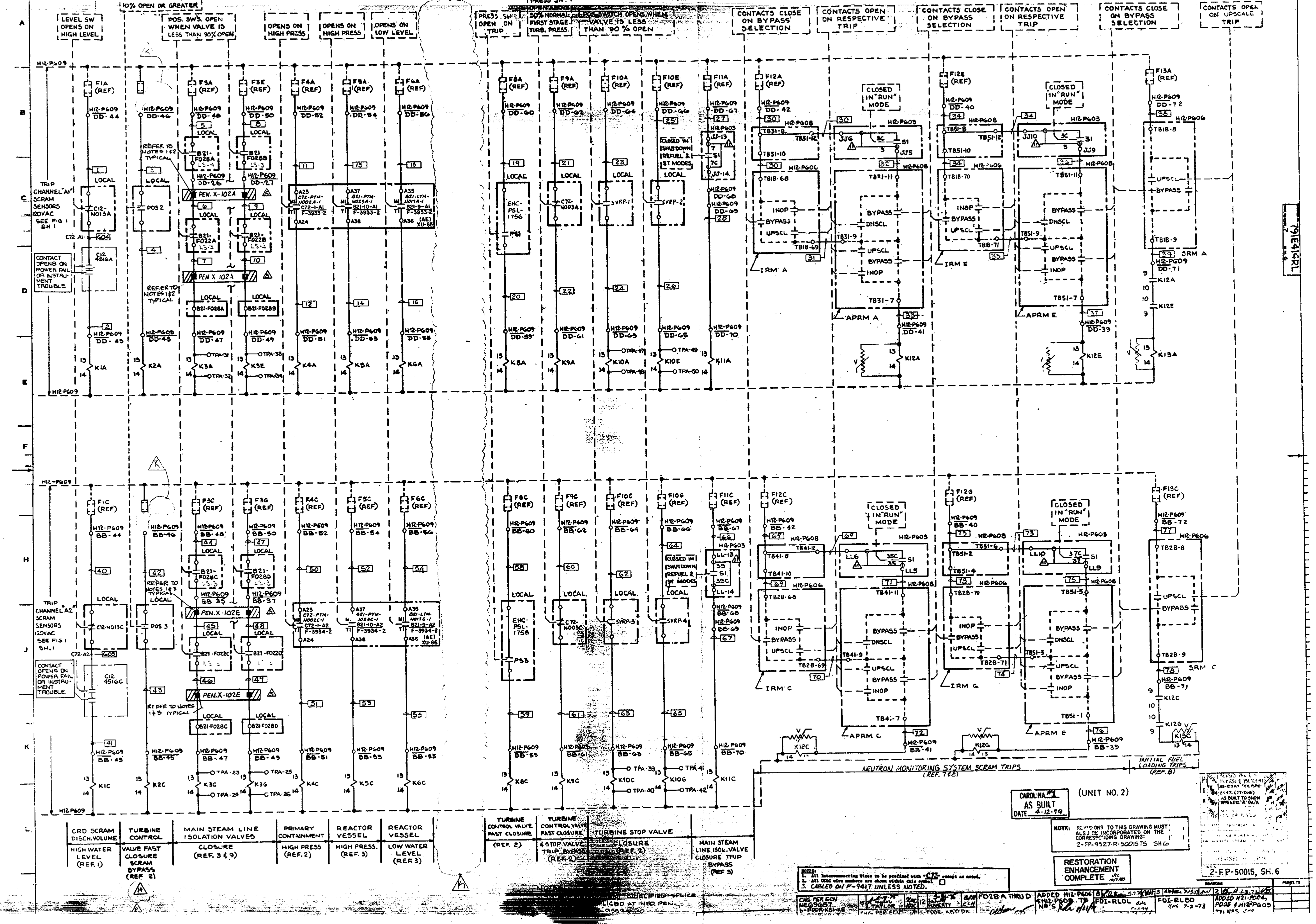
Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:____________

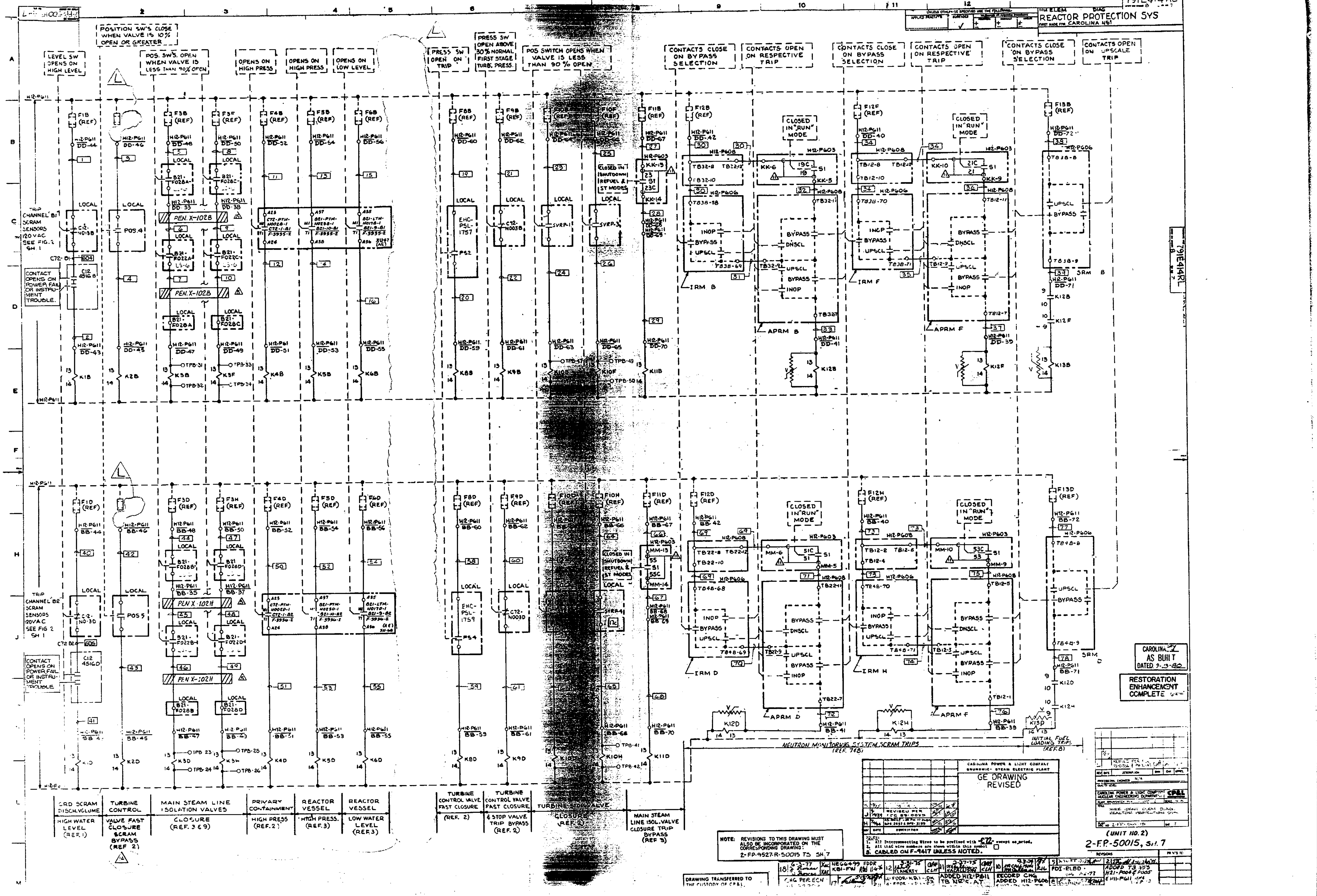
LPRM % POWER - An alternate indication of reactor power calculated only on the process computer which is obtained by averaging calibrated LPRM readings. This uses points U1NSSALTP and U2NSSALTP.

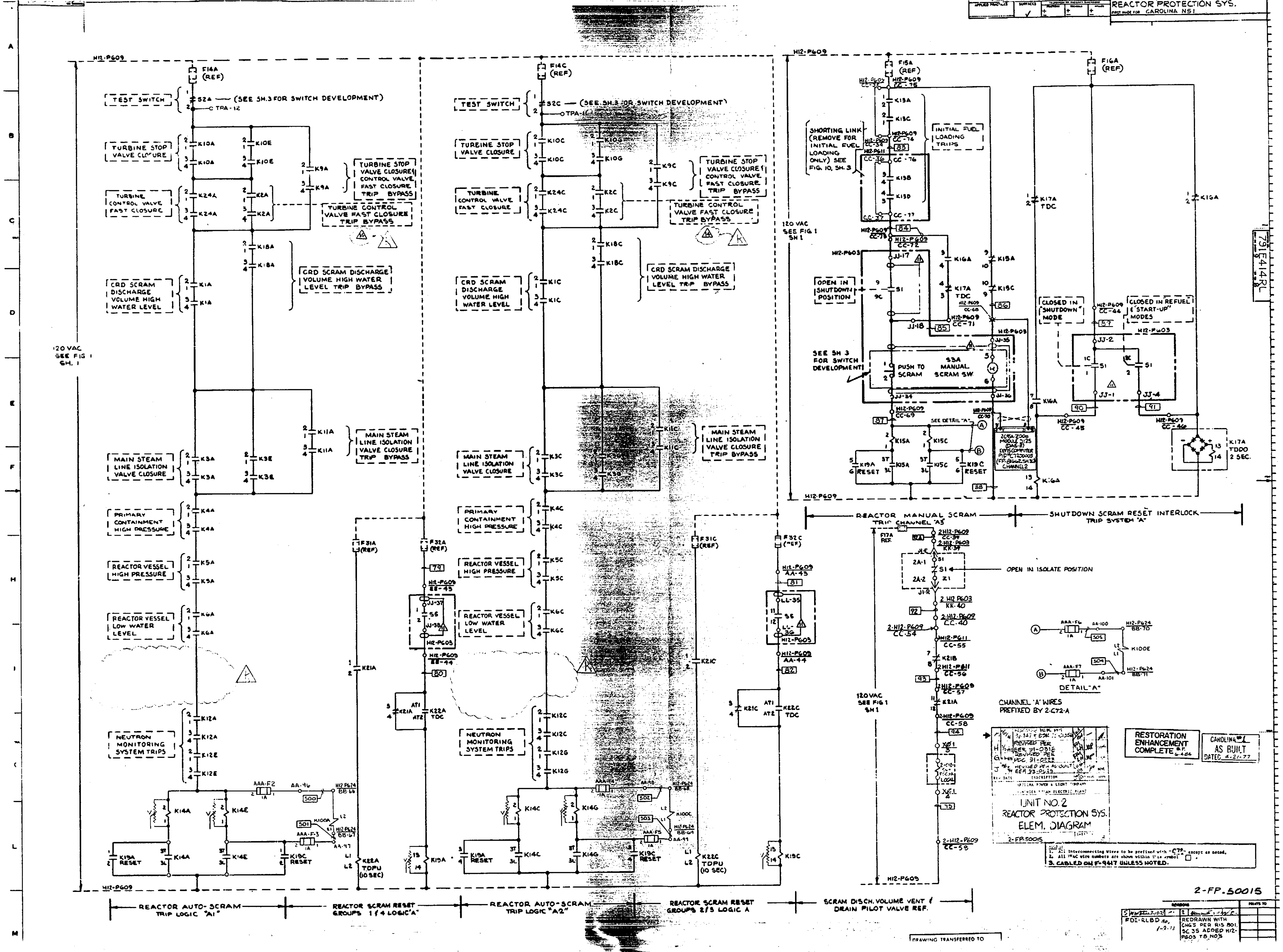
STEAM FLOW % POWER - An alternate indication of reactor power obtained by correlating the total steam flow to a valid heat balance. Total steam flow can be obtained from Plant Process Computer Point B041, the summation of ERFIS points C32FA014, C32FA015, C32FA016, C32FA017 (main steam lines A, B, C, D), or the summation of RTGB indications C32-R603A, B, C, D, on P603.

BYPASS VALVE % POWER - An alternate indication of reactor power obtained by correlating the total bypass valve position (percent open) to a valid heat balance. Total bypass valve position can be obtained from ERFIS point EHCXA002, or RTGB Panel XU1.

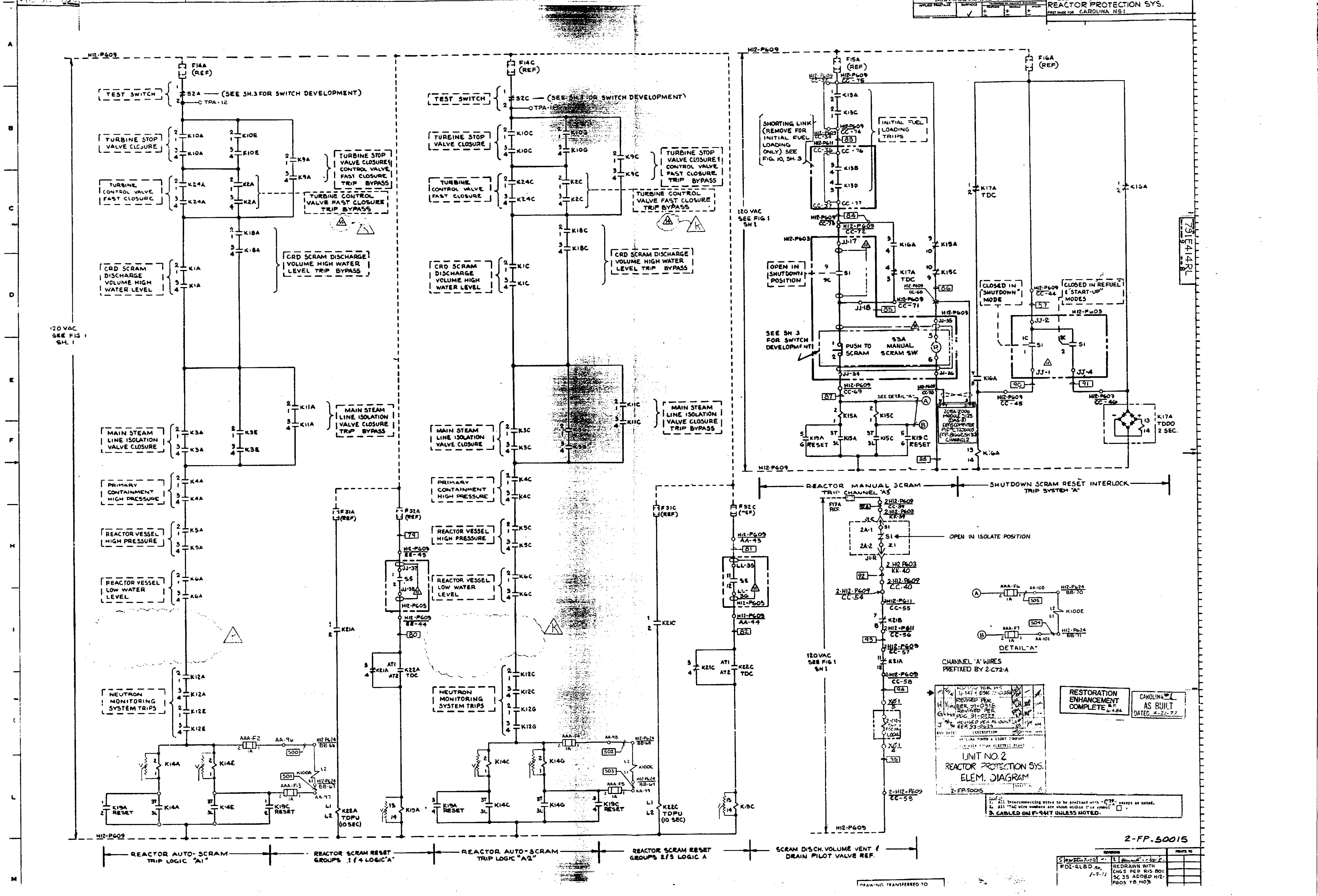


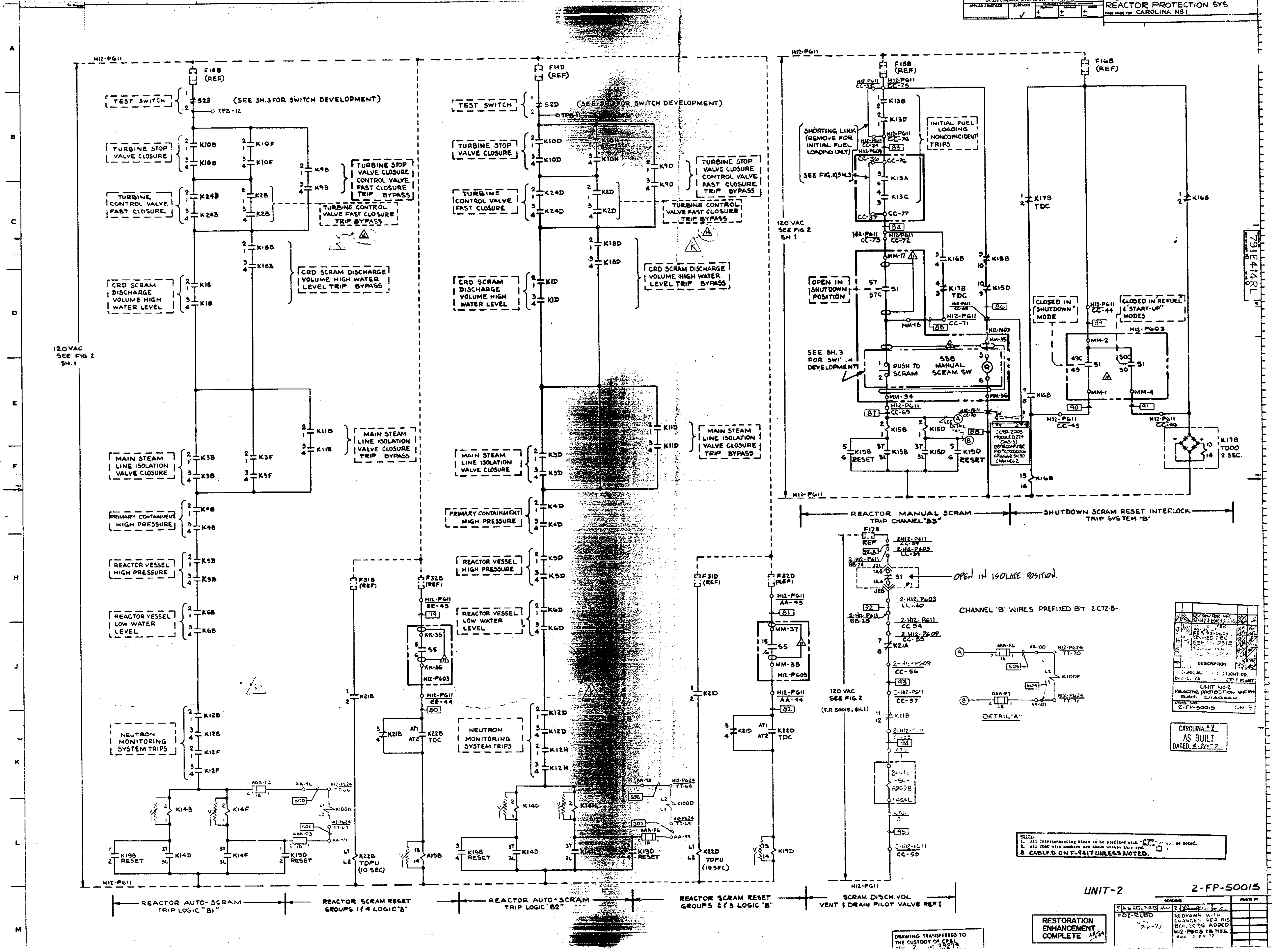
LEVEL SW OPENS ON HIGH LEVEL
10% OPEN OR GREATER
POS. SWS. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
50% NORMAL FIRST STAGE TURB. PRESS.
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
TRIP CHANNEL "A1" SCRAM SENSORS 120VAC SEE FIG I SH I
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
REFER TO NOTES 1&2 TYPICAL
CLOSED IN "RUN" MODE
CLOSED IN SHUTDOWN REFUEL & ST MODES
IRM A
APRM A
IRM E
APRM E
SRM A
TRIP CHANNEL "A2" SCRAM SENSORS 120VAC SEE FIG I SH. I
IRM C
APRM C
IRM G
APRM E
SRM C
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 7&8)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME
HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL
VALVE FAST CLOSURE SCRAM BYPASS (REF 2)
MAIN STEAM LINE ISOLATION VALVES
CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT
HIGH PRESS (REF. 2)
REACTOR VESSEL
HIGH PRESS. (REF. 3)
REACTOR VESSEL
LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE
(REF. 2)
TURBINE CONTROL VALVE FAST CLOSURE & STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE
CLOSURE (REF. 2)
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF 3)
CAROLINA P&L AS BUILT DATE 4-12-79
(UNIT NO. 2)
NOTE: REVISIONS TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING DRAWING: 2-FP-9527-R-50015TS SH6
RESTORATION ENHANCEMENT COMPLETE
2-FP-50015, SH. 6
3. CABLED ON F-9417 UNLESS NOTED.
79IE414RL


REACTOR PROTECTION SYS
CAROLINA Nº1
POSITION SW'S CLOSE WHEN VALVE IS 10% OPEN OR GREATER
LEVEL SW OPENS ON HIGH LEVEL
POS. SW'S. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
PRESS SW OPEN ABOVE 30% NORMAL FIRST STAGE TURB. PRESS
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
TRIP CHANNEL "B1" SCRAM SENSORS 120VAC SEE FIG. 2 SH 1
TRIP CHANNEL "B2" SCRAM SENSORS 120VAC SEE FIG 2 SH 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE.
CLOSED IN "RUN" MODE
CLOSED IN SHUTDOWN, REFUEL & ST MODES
IRM B
IRM F
IRM D
IRM H
APRM B
APRM F
APRM D
SRM B
SRM D
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 748)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL VALVE FAST CLOSURE SCRAM BYPASS (REF 2)
MAIN STEAM LINE ISOLATION VALVES CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT HIGH PRESS (REF. 2)
REACTOR VESSEL HIGH PRESS. (REF. 3)
REACTOR VESSEL LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE (REF. 2)
TURBINE CONTROL FAST CLOSURE & STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE CLOSURE (REF. 2)
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF 3)
CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT
GE DRAWING REVISED
NOTE: REVISIONS TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING DRAWING: 2-FP-9527-R-50015 T5 SH 7
CABLED ON F-9417 UNLESS NOTED.
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
CAROLINA 2 AS BUILT DATED 9-13-82
RESTORATION ENHANCEMENT COMPLETE
(UNIT NO. 2)
2-FP-50015, SH. 7
791E414RL

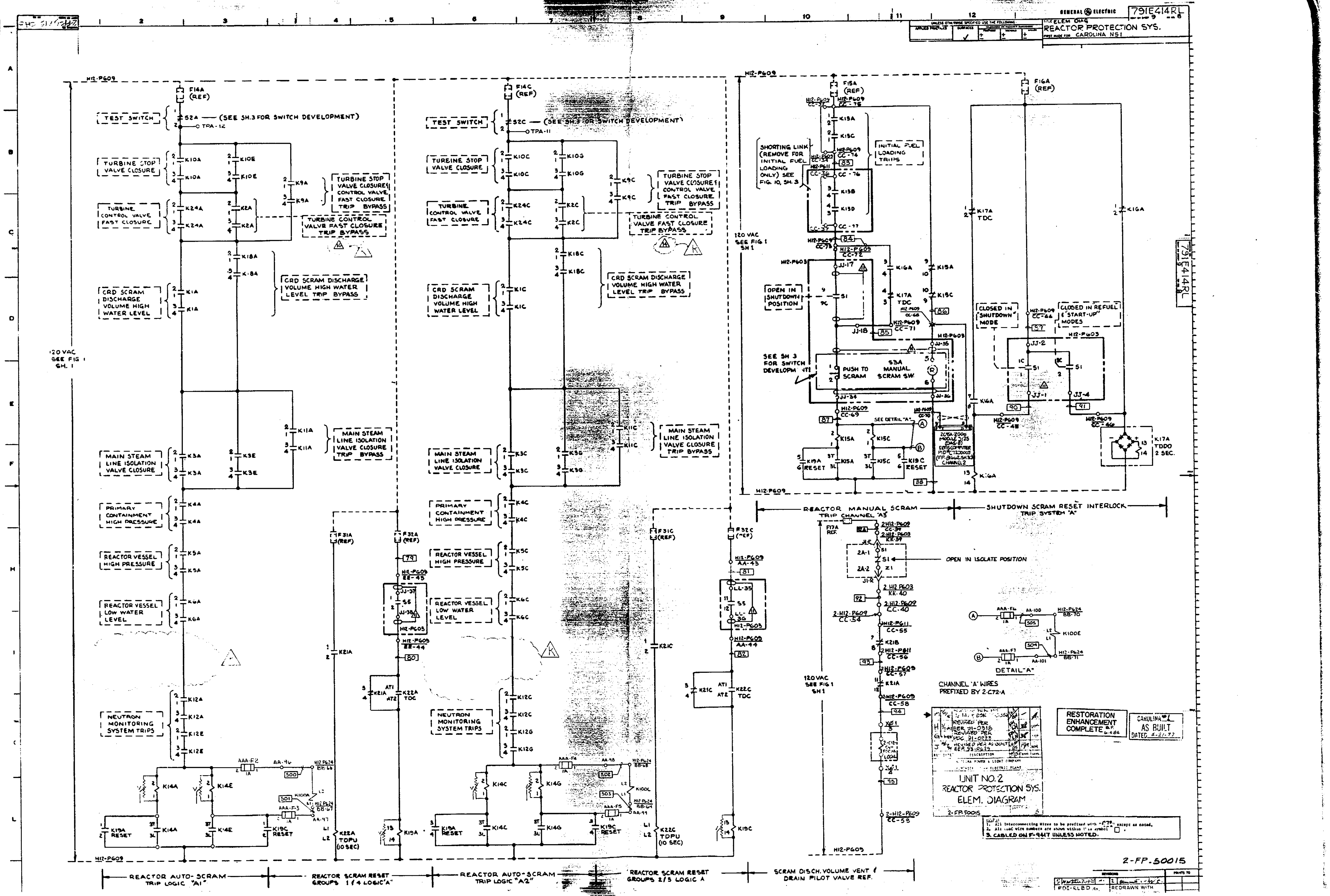


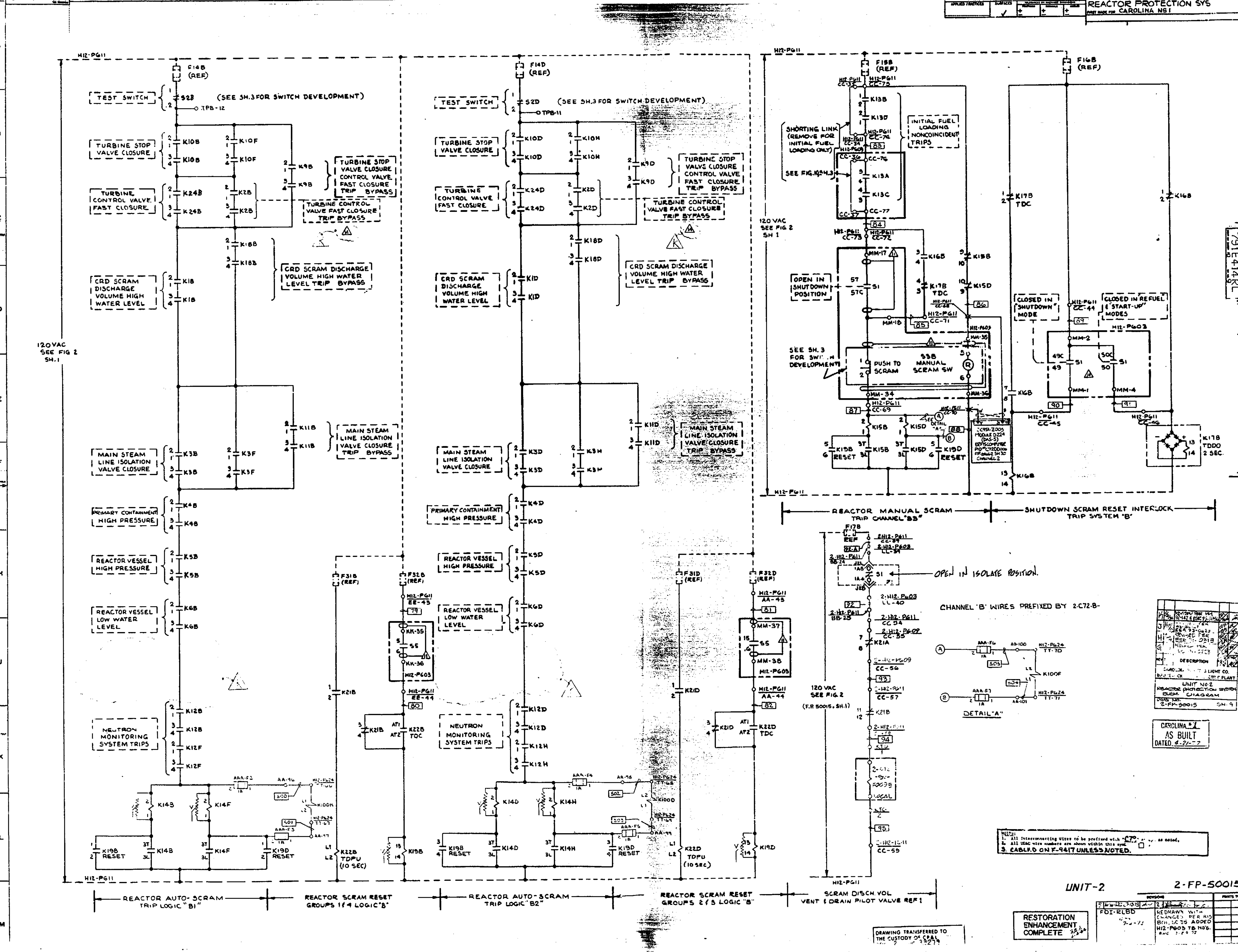
REACTOR PROTECTION SYS.
FIRST MADE FOR CAROLINA NS1
TEST SWITCH
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
120 VAC SEE FIG 1 SH. 1
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY) SEE FIG. 10, SH. 3
INITIAL FUEL LOADING TRIPS
OPEN IN SHUTDOWN POSITION
CLOSED IN "SHUTDOWN" MODE
CLOSED IN REFUEL & "START-UP" MODES
SEE SH 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3A MANUAL SCRAM SW
SEE DETAIL "A"
K17A TDDO 2 SEC.
REACTOR MANUAL SCRAM TRIP CHANNEL "A"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "A"
OPEN IN ISOLATE POSITION
DETAIL "A"
CHANNEL 'A' WIRES PREFIXED BY 2-C72-A
RESTORATION ENHANCEMENT COMPLETE
CAROLINA AS BUILT DATED 4-21-77
UNIT NO. 2
REACTOR PROTECTION SYS.
ELEM. DIAGRAM
2-FP-50015
1. All interconnecting wires to be prefixed with "C72" except as noted.
2. All P&C wire numbers are shown within the symbol □.
3. CABLED ON F-9447 UNLESS NOTED.
REACTOR AUTO-SCRAM TRIP LOGIC "A1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC 'A'
REACTOR AUTO-SCRAM TRIP LOGIC "A2"
REACTOR SCRAM RESET GROUPS 2 & 3 LOGIC A
SCRAM DISCH. VOLUME VENT & DRAIN PILOT VALVE REF.
REDRAWN WITH CHG'S PER RIS 801. SC 35 ADDED H12-P603 TB NO'S
DRAWING TRANSFERRED TO
79 E414 RL


REACTOR PROTECTION SYS.
CAROLINA NS1
H12-P609
F14A (REF)
TEST SWITCH
S2A — (SEE SH.3 FOR SWITCH DEVELOPMENT)
TPA-12
TURBINE STOP VALVE CLOSURE
K10A
K10E
K9A
TURBINE STOP VALVE CLOSURE | CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE
K24A
K2A
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
K18A
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
K1A
120 VAC SEE FIG 1 SH. 1
K11A
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
MAIN STEAM LINE ISOLATION VALVE CLOSURE
K3A
K3E
PRIMARY CONTAINMENT HIGH PRESSURE
K4A
REACTOR VESSEL HIGH PRESSURE
K5A
REACTOR VESSEL LOW WATER LEVEL
K6A
NEUTRON MONITORING SYSTEM TRIPS
K12A
K12E
AAA-F2
AA-96
H12-P624 BB-66
K14A
K14E
K100A
AAA-F-3
H12-P624 BB-67
AA-97
K19A RESET
K19C RESET
K22A TDPU (10 SEC)
F31A (REF)
K21A
F32A (REF)
H12-P609 EE-45
JJ-37
JJ-38
H12-P603
H12-P609 EE-44
ATI
AT2
K22A TDC
F14C (REF)
S2C — (SEE SH.3 FOR SWITCH DEVELOPMENT)
TPA-11
K10C
K10G
K9C
K24C
K2C
K18C
K1C
K11C
K3C
K4C
K5C
K6C
K12C
K12G
AAA-F4
K14C
K14G
K100C
AAA-F5
H12-P624 BB-68
F31C (REF)
K21C
K22C TDPU (10 SEC)
F32C (REF)
H12-P609 AA-43
LL-35
LL-36
H12-P609 AA-44
K22C TDC
K19C
120 VAC SEE FIG 1 SH 1
H12-P609
F15A (REF)
F16A (REF)
K13A
K13C
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY) SEE FIG. 10, SH. 3
INITIAL FUEL LOADING TRIPS
CC-36
CC-76
K13B
K13D
CC-37
CC-77
K17A TDC
K16A
H12-P609 CC-72
H12-P603
JJ-17
K16A
K15A
OPEN IN SHUTDOWN POSITION
K17A TDC
K15C
CLOSED IN "SHUTDOWN" MODE
H12-P609 CC-44
CLOSED IN REFUEL & "START-UP" MODES
H12-P609 CC-71
JJ-18
JJ-35
JJ-2
SEE SH 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3A MANUAL SCRAM SW.
JJ-34
JJ-36
JJ-1
JJ-4
H12-P609 CC-69
SEE DETAIL "A"
H12-P609 CC-45
H12-P603 CC-46
ZC15A-Z006 MODULE 2/25 (DAS-8) EXT'S COMPUTER P/P: C72D005 (FP-B661-SH3) CHANNEL 2
K17A TDDO 2 SEC.
K19A RESET
K15A
K15C
K19C RESET
K16A
REACTOR MANUAL SCRAM TRIP CHANNEL "A3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "A"
F17A REF.
2-H12-P609 CC-39
2-H12-P609 KK-39
2A-1
2A-2
S1
Z1
OPEN IN ISOLATE POSITION
2 H12-P603 KK-40
2-H12-P609 CC-40
2-H12-P609 CC-54
H12-P611 CC-55
K21B
H12-P611 CC-56
H12-P609 CC-57
K21A
2-H12-P609 CC-58
120VAC SEE FIG 1 SH 1
AAA-F6
AA-100
H12-P624 BB-70
K100E
AAA-F7
AA-101
H12-P624 BB-71
DETAIL "A"
CHANNEL 'A' WIRES PREFIXED BY 2-C72-A
RESTORATION ENHANCEMENT COMPLETE
CAROLINA AS BUILT DATED 4-21-77
UNIT NO. 2
REACTOR PROTECTION SYS.
ELEM. DIAGRAM
2-FP-50015
3. CABLED ON F-9417 UNLESS NOTED.
2-H12-P609 CC-58
H12-P609
2-FP-50015
REACTOR AUTO-SCRAM TRIP LOGIC "A1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "A"
REACTOR AUTO-SCRAM TRIP LOGIC "A2"
REACTOR SCRAM RESET GROUPS 2/3 LOGIC A
SCRAM DISCH. VOLUME VENT & DRAIN PILOT VALVE REF.
FDI-RLBD
REDRAWN WITH CHGS PER RIS 801 SC 35 ADDED H12-P603 TB NOS
DRAWING TRANSFERRED TO
79 E414 RL

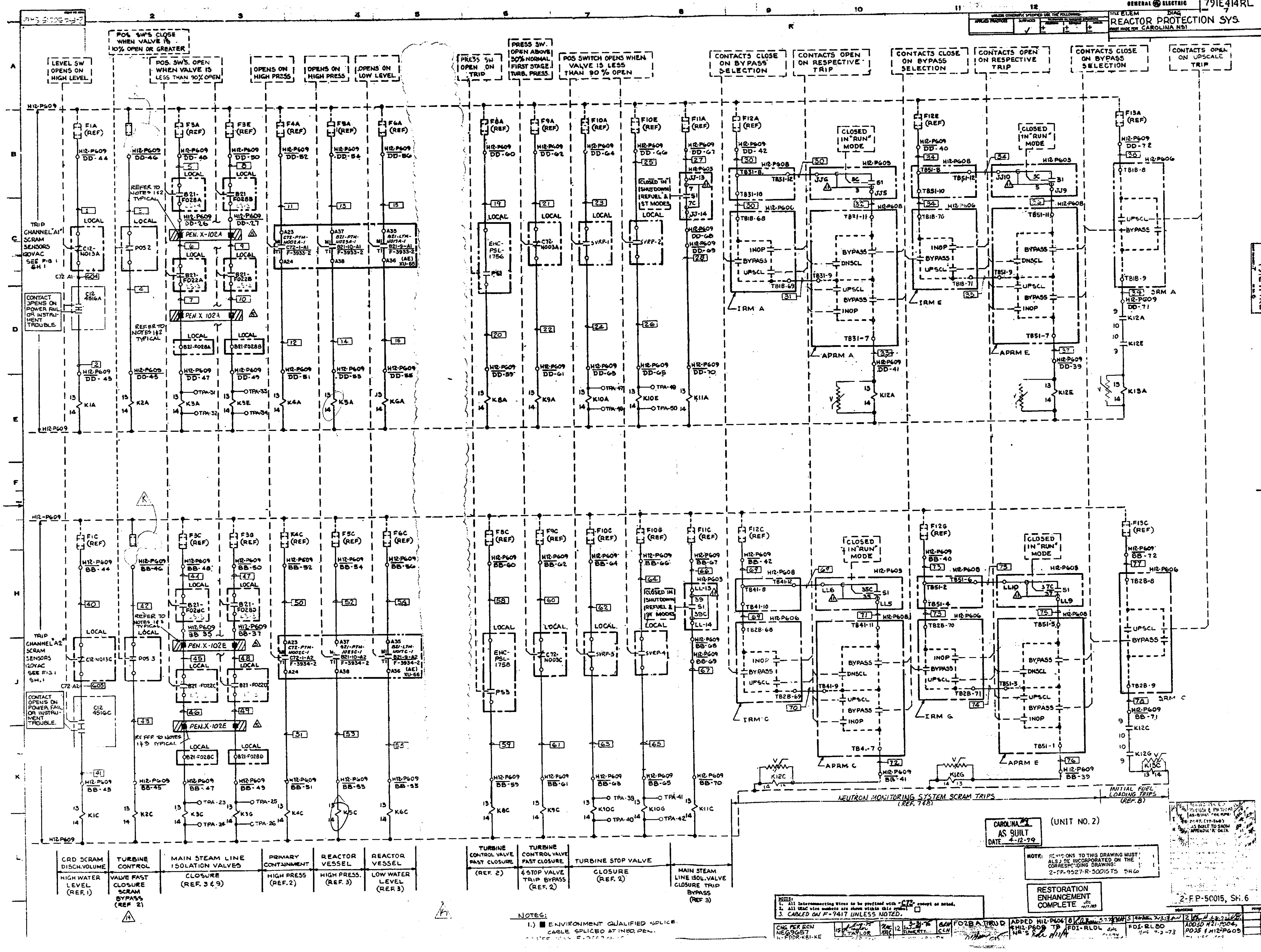


REACTOR PROTECTION SYS
CAROLINA N&L
TEST SWITCH
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
120 VAC SEE FIG 2 SH.1
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
REACTOR AUTO-SCRAM TRIP LOGIC "B1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "B"
REACTOR AUTO-SCRAM TRIP LOGIC "B2"
REACTOR SCRAM RESET GROUPS 2 & 3 LOGIC "B"
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY)
INITIAL FUEL LOADING NONCOINCIDENT TRIPS
OPEN IN SHUTDOWN POSITION
PUSH TO SCRAM
S3B MANUAL SCRAM SW
CLOSED IN SHUTDOWN MODE
CLOSED IN REFUEL & START-UP MODES
REACTOR MANUAL SCRAM TRIP CHANNEL "B3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "B"
OPEN IN ISOLATE POSITION
CHANNEL 'B' WIRES PREFIXED BY 2-C72-B-
DETAIL "A"
SCRAM DISCH. VOL. VENT & DRAIN PILOT VALVE REF 1
791E414RL
CAROLINA #1 AS BUILT
UNIT NO 2 REACTOR PROTECTION SYSTEM ELEM. DIAGRAM
3. CABLED ON F-9417 UNLESS NOTED.
UNIT-2
2-FP-50015
RESTORATION ENHANCEMENT COMPLETE
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L

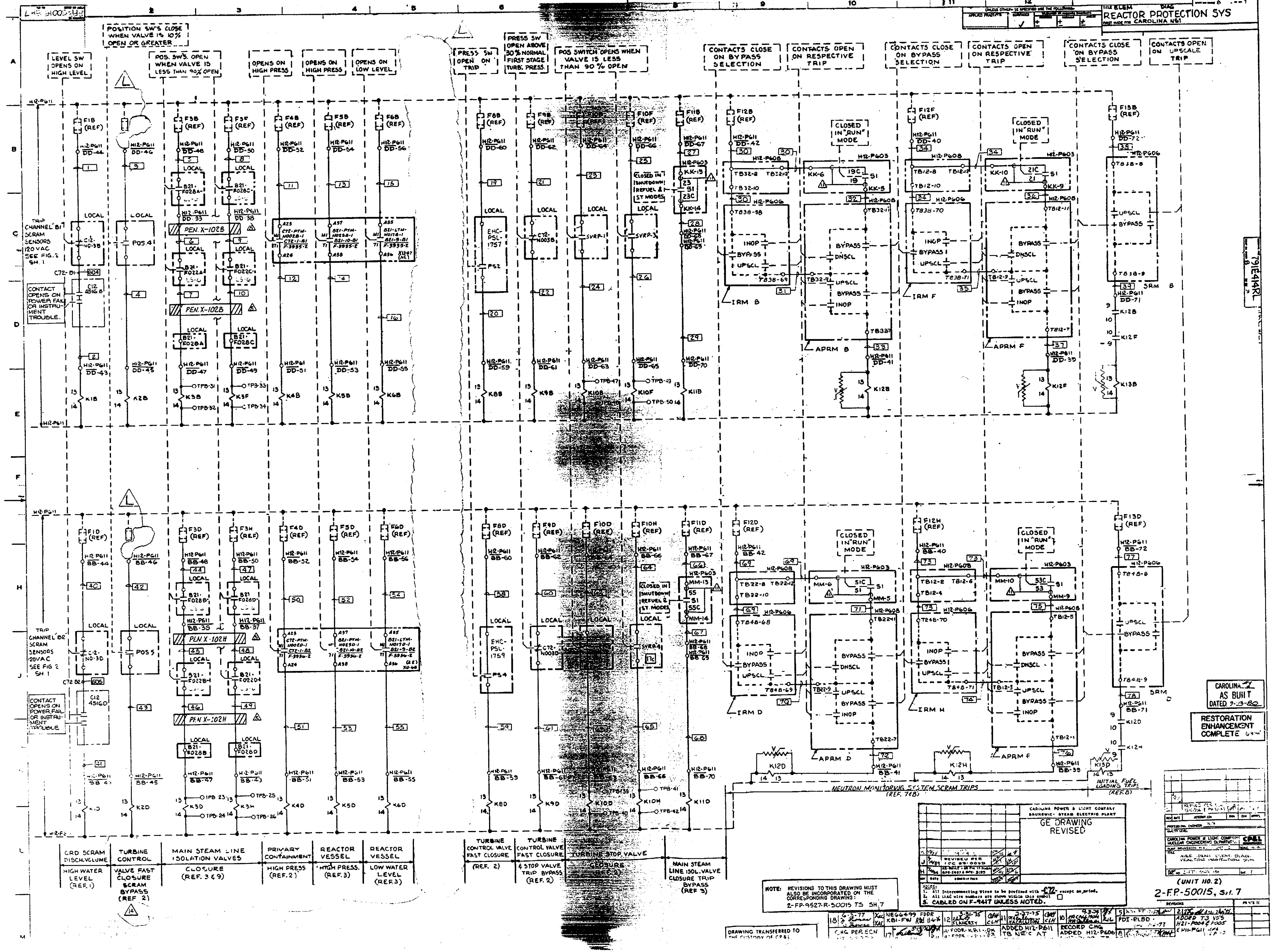

GENERAL ELECTRIC
791E414RL
ELEM DIAG
REACTOR PROTECTION SYS.
CAROLINA NSI
TEST SWITCH
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE/ CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
120 VAC SEE FIG 1 SH. 1
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY) SEE FIG. 10, SH. 3
INITIAL FUEL LOADING TRIPS
OPEN IN SHUTDOWN POSITION
SEE SH 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3A MANUAL SCRAM SW.
CLOSED IN "SHUTDOWN" MODE
CLOSED IN REFUEL & "START-UP" MODES
K19A RESET
K19C RESET
K17A TDDO 2 SEC.
REACTOR MANUAL SCRAM TRIP CHANNEL "A3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "A"
OPEN IN ISOLATE POSITION
DETAIL "A"
CHANNEL "A" WIRES PREFIXED BY 2-C72-A
RESTORATION ENHANCEMENT COMPLETE
CAROLINA AS BUILT
K22A TDPU (10 SEC)
K22C TDPU (10 SEC)
UNIT NO. 2
REACTOR PROTECTION SYS.
ELEM. DIAGRAM
2-FP-50015
NOTES:
3. CABLED ON F-9417 UNLESS NOTED.
REACTOR AUTO-SCRAM TRIP LOGIC "A1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "A"
REACTOR AUTO-SCRAM TRIP LOGIC "A2"
REACTOR SCRAM RESET GROUPS 2 & 3 LOGIC A
SCRAM DISCH. VOLUME VENT & DRAIN PILOT VALVE REF.
REDRAWN WITH

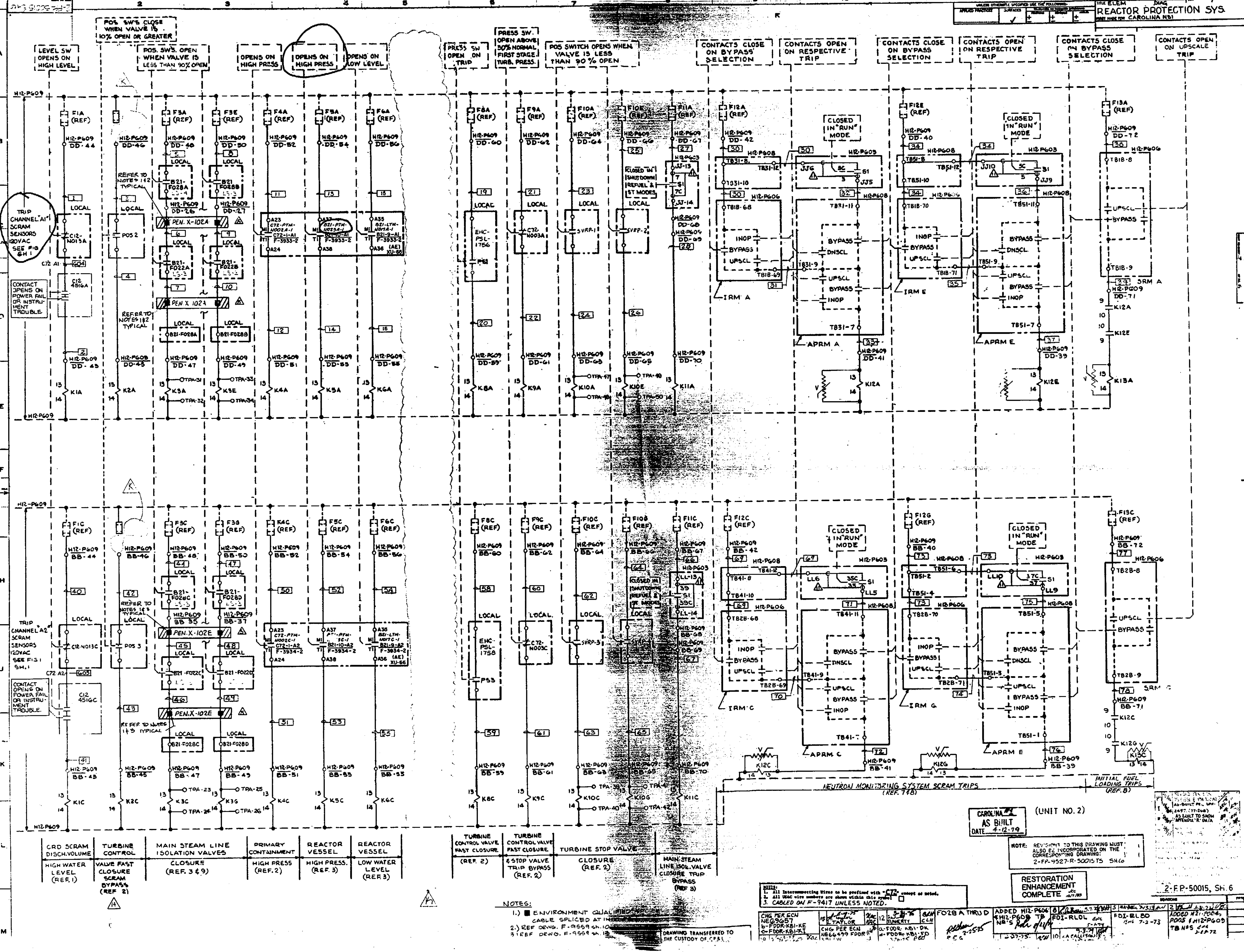


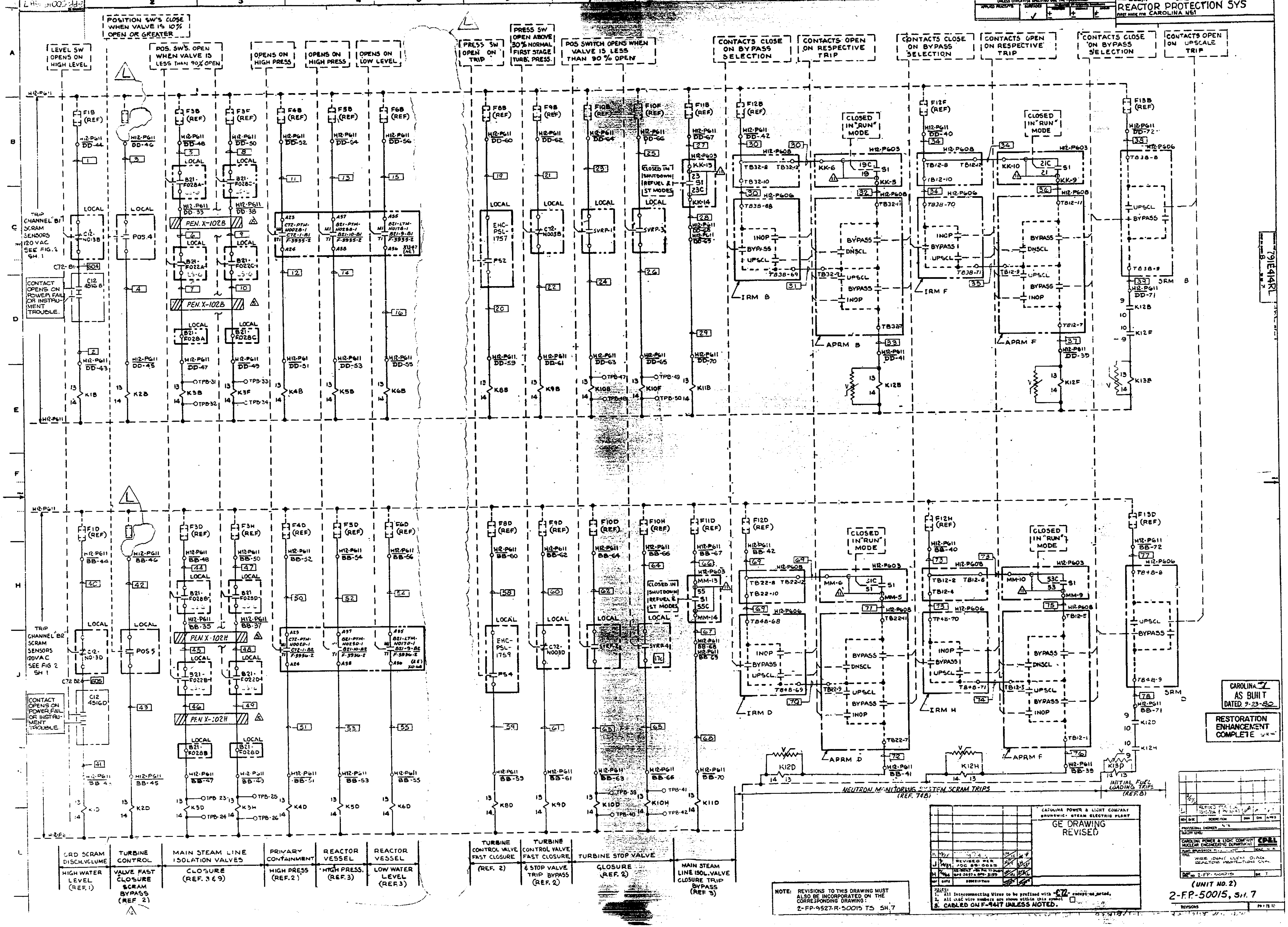
REACTOR PROTECTION SYS
CAROLINA NSI
TEST SWITCH
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
120 VAC SEE FIG 2 SH. 1
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
K19B RESET
K19D RESET
K22B TDPU (10 SEC)
K22D TDPU (10 SEC)
REACTOR AUTO-SCRAM TRIP LOGIC "B1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "B"
REACTOR AUTO-SCRAM TRIP LOGIC "B2"
REACTOR SCRAM RESET GROUPS 2 & 3 LOGIC "B"
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY)
INITIAL FUEL LOADING NONCOINCIDENT TRIPS
SEE FIG. K5 SH.3
OPEN IN SHUTDOWN POSITION
SEE SH. 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3B MANUAL SCRAM SW
CLOSED IN SHUTDOWN MODE
CLOSED IN REFUEL & START-UP MODES
K17B TDC
K17B TDDO 2 SEC.
REACTOR MANUAL SCRAM TRIP CHANNEL "B3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "B"
OPEN IN ISOLATE POSITION.
CHANNEL 'B' WIRES PREFIXED BY 2-C72-B-
DETAIL "A"
120 VAC SEE FIG 2 (F.P. 50015, SH.1)
SCRAM DISCH. VOL VENT & DRAIN PILOT VALVE REF 1
NOTES:
1. All interconnecting wires to be prefixed with "C72" or as noted.
2. All VRAC wire numbers are shown within this symbol.
3. CABLE.D ON F-9417 UNLESS NOTED.
UNIT-2
2-FP-50015
UNIT NO 2 REACTOR PROTECTION SYSTEM ELEM. DIAGRAM
CAROLINA #2 AS BUILT
RESTORATION ENHANCEMENT COMPLETE
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
FDI-RLBD
REDRAWN WITH CHANGES PER R13 BOLTS, CS ADDED H12-P609 TB NOS.
791E414RL



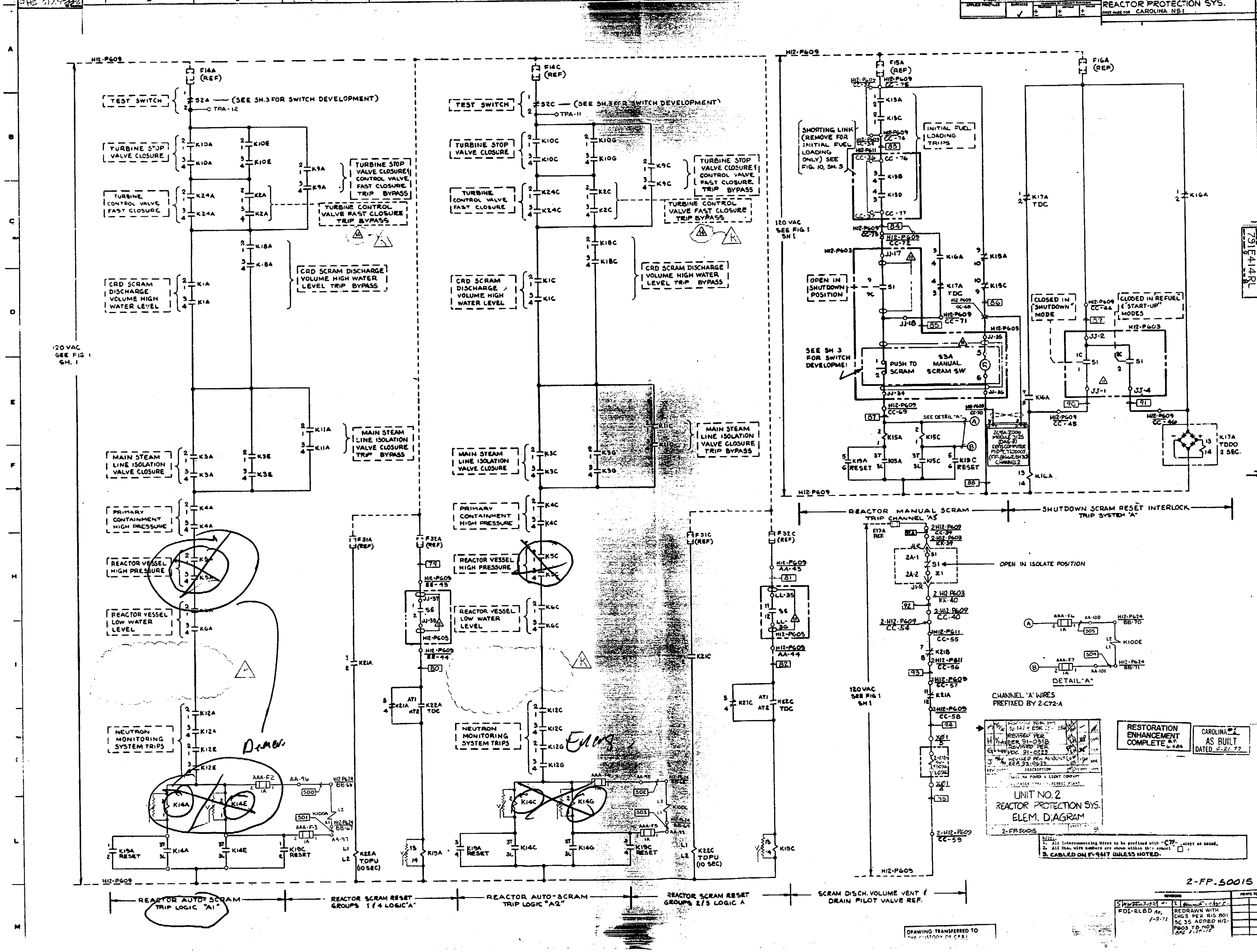
GENERAL ELECTRIC
791E414RL
ELEM DIAG
REACTOR PROTECTION SYS.
CAROLINA NSI
LEVEL SW OPENS ON HIGH LEVEL
POS. SW'S CLOSE WHEN VALVE IS 10% OPEN OR GREATER
POS. SW'S. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
PRESS SW. OPEN ABOVE 30% NORMAL FIRST STAGE TURB. PRESS.
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
TRIP CHANNEL "A1" SCRAM SENSORS 120VAC SEE FIG 1 SH 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
TRIP CHANNEL "A2" SCRAM SENSORS 120VAC SEE FIG 1 SH. 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
CLOSED IN "RUN" MODE
IRM A
APRM A
IRM E
APRM E
IRM C
APRM C
IRM G
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 7&8)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME
HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL
VALVE FAST CLOSURE SCRAM BYPASS (REF 2)
MAIN STEAM LINE ISOLATION VALVES
CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT
HIGH PRESS (REF. 2)
REACTOR VESSEL
HIGH PRESS. (REF. 3)
REACTOR VESSEL
LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE (REF. 2)
TURBINE CONTROL VALVE FAST CLOSURE
& STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE CLOSURE (REF. 2)
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF 3)
CAROLINA #2 AS BUILT DATE 4-12-79
(UNIT NO. 2)
RESTORATION ENHANCEMENT COMPLETE
2-FP-50015, SH. 6
NOTES:
1.) ■ ENVIRONMENT QUALIFIED SPLICE. CABLE SPLICED AT INBD. PEN.


REACTOR PROTECTION SYS
CAROLINA NS1
POSITION SW'S CLOSE WHEN VALVE IS 10% OPEN OR GREATER
LEVEL SW OPENS ON HIGH LEVEL
POS. SW'S. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
PRESS SW OPEN ABOVE 30% NORMAL FIRST STAGE TURB. PRESS.
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
TRIP CHANNEL 'B1' SCRAM SENSORS 120 VAC SEE FIG. 2 SH. 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE.
PEN. X-102B
PEN. X-102B
CLOSED IN "RUN" MODE
CLOSED IN SHUTDOWN, REFUEL & ST. MODES
IRM B
APRM B
IRM F
APRM F
SRM
TRIP CHANNEL 'B2' SCRAM SENSORS 120VAC SEE FIG 2 SH 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
PEN X-102H
PEN X-102H
IRM D
APRM D
IRM H
APRM F
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 7&8)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME
HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL
VALVE FAST CLOSURE SCRAM BYPASS (REF 2)
MAIN STEAM LINE ISOLATION VALVES
CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT
HIGH PRESS (REF. 2)
REACTOR VESSEL
HIGH PRESS. (REF. 3)
REACTOR VESSEL
LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE (REF. 2)
TURBINE CONTROL VALVE FAST CLOSURE & STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE CLOSURE
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF 5)
CAROLINA 2 AS BUILT DATED 9-13-80
RESTORATION ENHANCEMENT COMPLETE
CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT
GE DRAWING REVISED
NOTE: REVISIONS TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING DRAWING: 2-FP-9527-R-50015 TS SH. 7
5. CABLED ON F-9417 UNLESS NOTED.
(UNIT NO. 2)
2-FP-50015, SH. 7
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
ADDED H12-P611 TB NO'S AT
RECORD CHG ADDED H12-P606

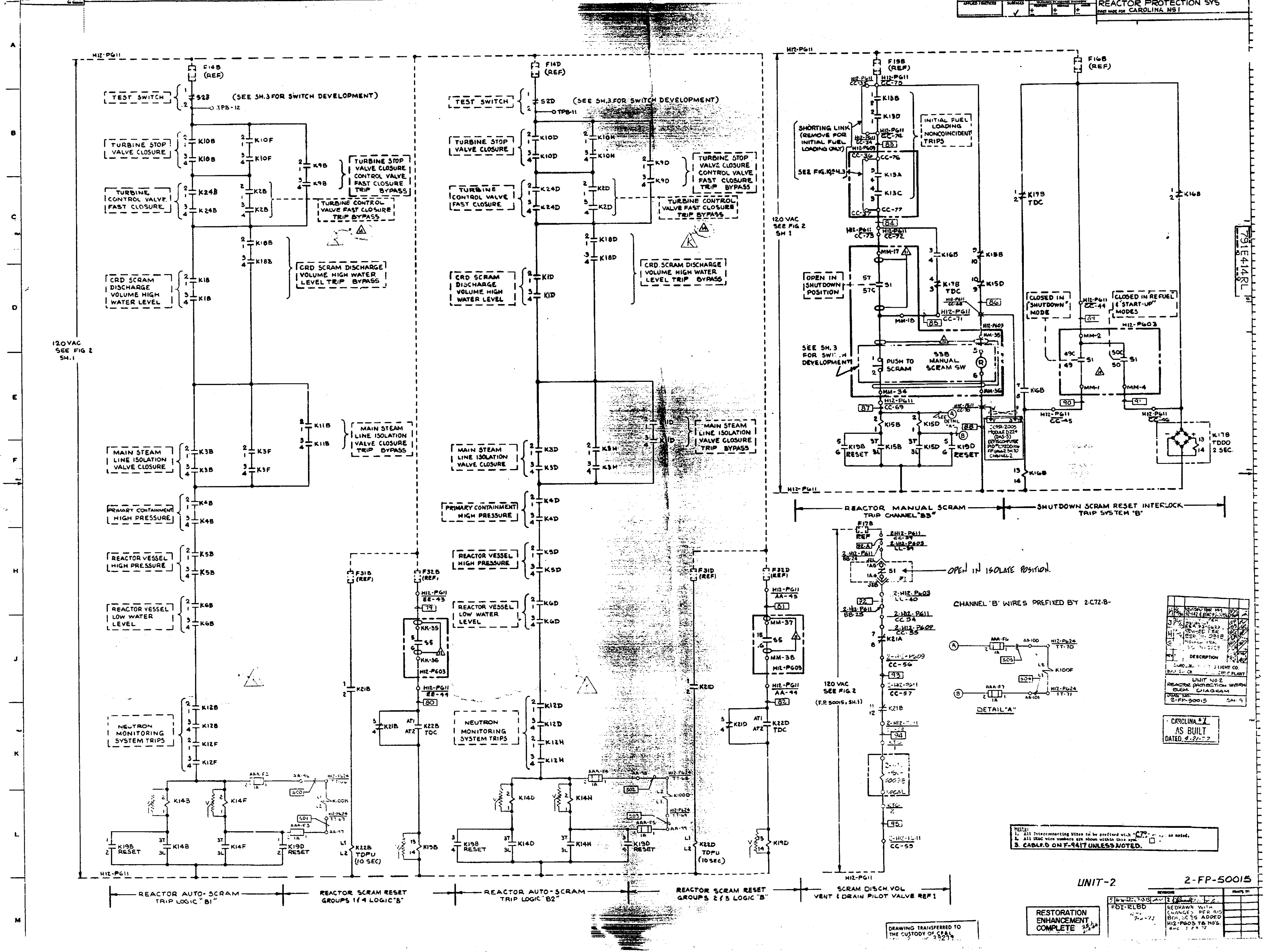

ELEM DIAG
REACTOR PROTECTION SYS.
CAROLINA NSI
LEVEL SW OPENS ON HIGH LEVEL
POS. SW'S CLOSE WHEN VALVE IS 10% OPEN OR GREATER
POS. SW'S. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
PRESS SW. OPEN ABOVE 30% NORMAL FIRST STAGE TURB. PRESS.
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
TRIP CHANNEL "A1" SCRAM SENSORS 120VAC SEE FIG SH.1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
TRIP CHANNEL "A2" SCRAM SENSORS 120VAC SEE FIG SH.1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE
REFER TO NOTES 1&2 TYPICAL
CLOSED IN "RUN" MODE
CLOSED IN SHUTDOWN REFUEL & ST MODES
IRM A
APRM A
IRM E
APRM E
SRM A
IRM C
APRM C
IRM G
APRM E
SRM C
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 7&8)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL VALVE FAST CLOSURE SCRAM BYPASS (REF. 2)
MAIN STEAM LINE ISOLATION VALVES CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT HIGH PRESS (REF. 2)
REACTOR VESSEL HIGH PRESS. (REF. 3)
REACTOR VESSEL LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE (REF. 2)
TURBINE CONTROL VALVE & STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE CLOSURE (REF. 2)
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF. 3)
CAROLINA #2 AS BUILT DATE 4-12-79
(UNIT NO. 2)
NOTE: REVISIONS TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING DRAWING: 2-FP-9527-R-50015TS SH.6
RESTORATION ENHANCEMENT COMPLETE
NOTES:
1.) ■ ENVIRONMENT QUALIFIED SPLICE CABLE SPLICED AT
2.) REF DWG. F-9559 SH.10
3.) REF DWG. F-9559 SH.12
1. All interconnecting wires to be prefixed with "C72" except as noted.
2. All 120AC wire numbers are shown within this symbol
3. CABLED ON F-9417 UNLESS NOTED.
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
2-F.P.-50015, SH. 6
791E414RL



REACTOR PROTECTION SYS
FIRST MADE FOR CAROLINA N51
POSITION SW'S CLOSE WHEN VALVE IS 10% OPEN OR GREATER
LEVEL SW OPENS ON HIGH LEVEL
POS. SW'S. OPEN WHEN VALVE IS LESS THAN 90% OPEN
OPENS ON HIGH PRESS
OPENS ON HIGH PRESS
OPENS ON LOW LEVEL
PRESS SW OPEN ON TRIP
PRESS SW OPEN ABOVE 30% NORMAL FIRST STAGE TURB. PRESS.
POS SWITCH OPENS WHEN VALVE IS LESS THAN 90% OPEN
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON RESPECTIVE TRIP
CONTACTS CLOSE ON BYPASS SELECTION
CONTACTS OPEN ON UPSCALE TRIP
CLOSED IN "RUN" MODE
CLOSED IN SHUTDOWN REFUEL & ST MODES
TRIP CHANNEL "B1" SCRAM SENSORS 120 VAC SEE FIG. 2 SH. 1
CONTACT OPENS ON POWER FAIL OR INSTRUMENT TROUBLE.
TRIP CHANNEL "B2" SCRAM SENSORS 120VAC SEE FIG 2 SH 1
PEN. X-102B
PEN. X-102H
IRM B
APRM B
IRM F
APRM F
SRM
IRM D
APRM D
IRM H
APRM F
NEUTRON MONITORING SYSTEM SCRAM TRIPS (REF. 7&8)
INITIAL FUEL LOADING TRIPS (REF. 8)
CRD SCRAM DISCH. VOLUME
HIGH WATER LEVEL (REF. 1)
TURBINE CONTROL
VALVE FAST CLOSURE SCRAM BYPASS (REF 2)
MAIN STEAM LINE ISOLATION VALVES
CLOSURE (REF. 3 & 9)
PRIMARY CONTAINMENT
HIGH PRESS (REF. 2)
REACTOR VESSEL
HIGH PRESS. (REF. 3)
REACTOR VESSEL
LOW WATER LEVEL (REF. 3)
TURBINE CONTROL VALVE FAST CLOSURE (REF. 2)
TURBINE CONTROL VALVE FAST CLOSURE & STOP VALVE TRIP BYPASS (REF. 2)
TURBINE STOP VALVE CLOSURE (REF. 2)
MAIN STEAM LINE ISOL. VALVE CLOSURE TRIP BYPASS (REF 3)
CAROLINA "7" AS BUILT DATED 9-23-82
RESTORATION ENHANCEMENT COMPLETE
CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT
GE DRAWING REVISED
NOTE: REVISIONS TO THIS DRAWING MUST ALSO BE INCORPORATED ON THE CORRESPONDING DRAWING: 2-FP-9527-R-50015 TS SH. 7
NOTES:
1. All interconnecting wires to be prefixed with "C72" except as noted.
2. All VAC wire numbers are shown within this symbol
3. CABLED ON F-9417 UNLESS NOTED.
CAROLINA POWER & LIGHT COMPANY NUCLEAR ENGINEERING DEPARTMENT CP&L
791E414RL
(UNIT NO. 2)
2-FP-50015, SH. 7

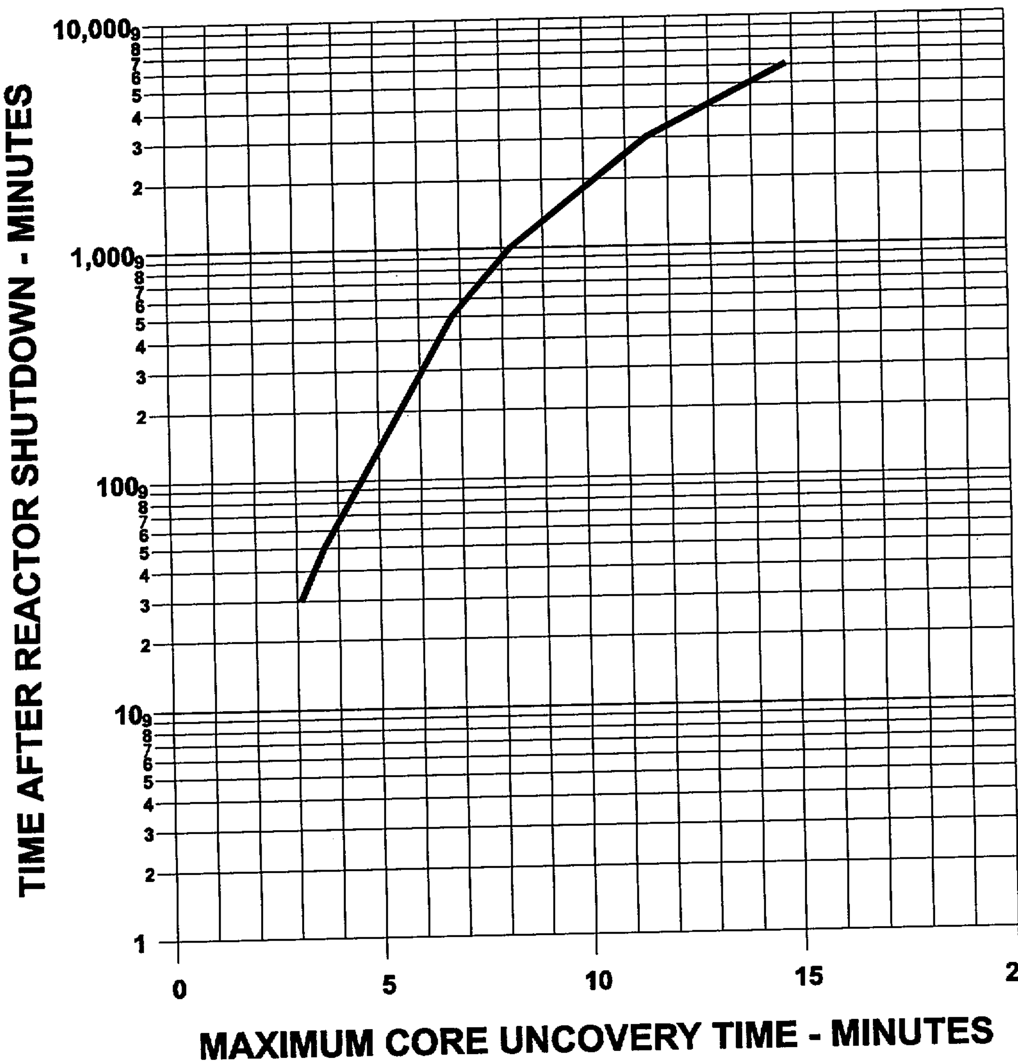


REACTOR PROTECTION SYS.
CAROLINA NSI
TEST SWITCH
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE & CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
120 VAC SEE FIG 1 SH. 1
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
Demer
Energ
K19A RESET
K19C RESET
K22A TDPU (10 SEC)
K22C TDPU (10 SEC)
REACTOR AUTO-SCRAM TRIP LOGIC "A1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "A"
REACTOR AUTO-SCRAM TRIP LOGIC "A2"
REACTOR SCRAM RESET GROUPS 2/3 LOGIC A
SCRAM DISCH. VOLUME VENT & DRAIN PILOT VALVE REF.
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY) SEE FIG. 10, SH. 3
INITIAL FUEL LOADING TRIPS
OPEN IN SHUTDOWN POSITION
SEE SH 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3A MANUAL SCRAM SW
CLOSED IN SHUTDOWN MODE
CLOSED IN REFUEL & "START-UP" MODES
K17A TDDO 2 SEC.
REACTOR MANUAL SCRAM TRIP CHANNEL "A3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "A"
OPEN IN ISOLATE POSITION
DETAIL "A"
CHANNEL "A" WIRES PREFIXED BY 2-C72-A
RESTORATION ENHANCEMENT COMPLETE
CAROLINA AS BUILT
UNIT NO. 2
REACTOR PROTECTION SYS.
ELEM. DIAGRAM
2-FP-50015
3. CABLED ON F-9617 UNLESS NOTED.
2-FP-50015
REDRAWN WITH CHGS PER RIS 801 SC 35 ADDED H12-P603 TB NOS
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
791E414RL


REACTOR PROTECTION SYS
CAROLINA NS I
H12-PG11
F14B (REF)
TEST SWITCH
S2B
(SEE SH.3 FOR SWITCH DEVELOPMENT)
TPB-12
TURBINE STOP VALVE CLOSURE
TURBINE CONTROL VALVE FAST CLOSURE
TURBINE STOP VALVE CLOSURE CONTROL VALVE FAST CLOSURE TRIP BYPASS
TURBINE CONTROL VALVE FAST CLOSURE TRIP BYPASS
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
CRD SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP BYPASS
120 VAC SEE FIG 2 SH. 1
MAIN STEAM LINE ISOLATION VALVE CLOSURE
MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP BYPASS
PRIMARY CONTAINMENT HIGH PRESSURE
REACTOR VESSEL HIGH PRESSURE
REACTOR VESSEL LOW WATER LEVEL
NEUTRON MONITORING SYSTEM TRIPS
F14D (REF)
TEST SWITCH
S2D
TPB-11
F31B (REF)
F32B (REF)
F31D (REF)
F32D (REF)
K21B
K22B TDC
K22B TDPU (10 SEC)
K21D
K22D TDC
K22D TDPU (10 SEC)
K19B RESET
K19D RESET
REACTOR AUTO-SCRAM TRIP LOGIC "B1"
REACTOR SCRAM RESET GROUPS 1 & 4 LOGIC "B"
REACTOR AUTO-SCRAM TRIP LOGIC "B2"
REACTOR SCRAM RESET GROUPS 2 & 3 LOGIC "B"
F15B (REF)
F16B (REF)
SHORTING LINK (REMOVE FOR INITIAL FUEL LOADING ONLY)
INITIAL FUEL LOADING NONCOINCIDENT TRIPS
SEE FIG.10 SH.3
120 VAC SEE FIG 2 SH 1
OPEN IN SHUTDOWN POSITION
SEE SH. 3 FOR SWITCH DEVELOPMENT
PUSH TO SCRAM
S3B MANUAL SCRAM SW
CLOSED IN SHUTDOWN MODE
CLOSED IN REFUEL & "START-UP" MODES
K17B TDDO 2 SEC.
REACTOR MANUAL SCRAM TRIP CHANNEL "B3"
SHUTDOWN SCRAM RESET INTERLOCK TRIP SYSTEM "B"
OPEN IN ISOLATE POSITION.
CHANNEL 'B' WIRES PREFIXED BY 2-C72-B-
120 VAC SEE FIG 2 (F.P. 50015, SH.1)
DETAIL "A"
SCRAM DISCH. VOL VENT & DRAIN PILOT VALVE REF 1
3. CABLED ON F-9417 UNLESS NOTED.
UNIT NO 2
REACTOR PROTECTION SYSTEM ELEM. DIAGRAM
2-FP-50015
CAROLINA #1
AS BUILT
UNIT-2
2-FP-50015
RESTORATION ENHANCEMENT COMPLETE
DRAWING TRANSFERRED TO THE CUSTODY OF CP&L
791E414RL

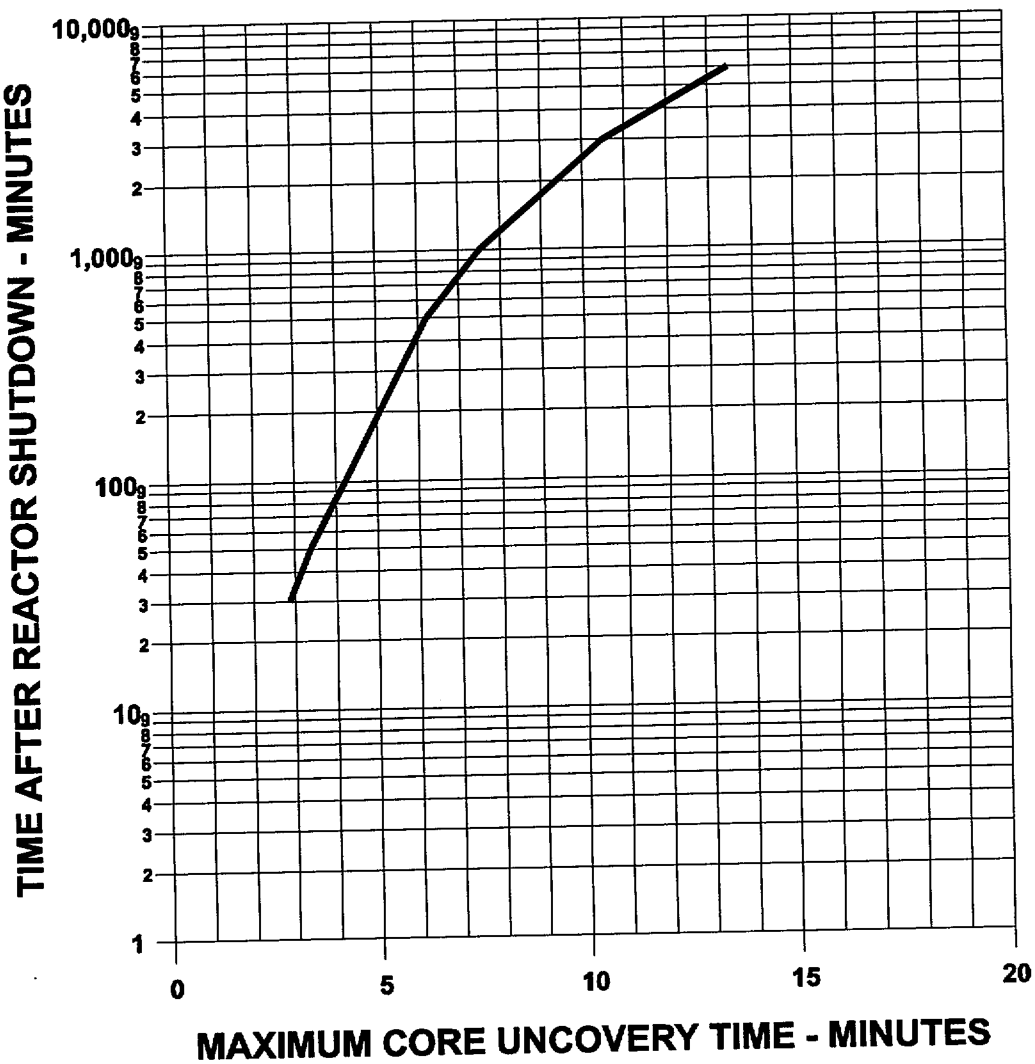
ATTACHMENT 5 (Cont'd)

FIGURE 4
UNIT 1 MAXIMUM CORE UNCOVERY TIME LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 4A
UNIT 2 MAXIMUM CORE UNCOVERY TIME LIMIT



EOP-01-UG
Attachment 6
Reactor Water Level Caution
(Caution 1)

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

If the temperature near any instrument run is in the UNSAFE region of the REACTOR SATURATION LIMIT (Figure 14), the instrument may be unreliable due to boiling in the run.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	Unit 1 only: The indicated level is in the SAFE region of Figure 15. Unit 2 only: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The indicated level is in the SAFE region of Figure 16. NOTE To determine reactor water level at the Main Steam Line Flood Level (MSL), see Figure 21. NOTE The figure has two curves: The upper curve is for reference leg area drywell temperature equal to or greater than 200°F. The lower curve is for reference leg area drywell temperature less than 200°F.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, OR B21TA103) AND 2. IF the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 20 inches OR IF the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), THEN the indicated level is greater than 10 inches.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. IF the reference leg area drywell temperature is less than 440°F, THEN the indicated level is greater than -150 inches OR IF the reference leg area drywell temperature is greater than or equal to 440°F, THEN the indicated level is greater than -130 inches. AND 2. Reactor Recirculation Pumps are shutdown. NOTE To determine reactor water level at TAF, see Unit 1 only: Figure 17 and Unit 2 only: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see Unit 1 only: Figure 18 and Unit 2 only: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see Unit 1 only: Figure 19 and Unit 2 only: Figure 19A To determine reactor water level at 90 inches, see Figure 20. Continued on next page.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	NOTE Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. NOTE These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 13
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

1. For all Level Instruments EXCEPT B21-LI-R605 A, B, (N027 A, B); the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1B Point 1258-1 ______
CAC-TR-4426-1B Point 1258-3 ______
CAC-TR-4426-2B Point 1258-2 ______
CAC-TR-4426-2B Point 1258-4 ______

OR

Microprocessor

CAC-TY-4426-1 Point 5801 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

ATTACHMENT 6 (Cont'd)

FIGURE 13 (Cont'd)
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

2. For Level Instruments B21-LI-R605A, B (N027A, B), the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1A Point 1258-22 ______
CAC-TR-4426-1B Point 1258-3 ______
CAC-TR-4426-2A Point 1258-23 ______
CAC-TR-4426-2A Point 1258-24 ______
CAC-TR-4426-2B Point 1258-2 ______
CAC-TR-4426-2B Point 1258-4 ______

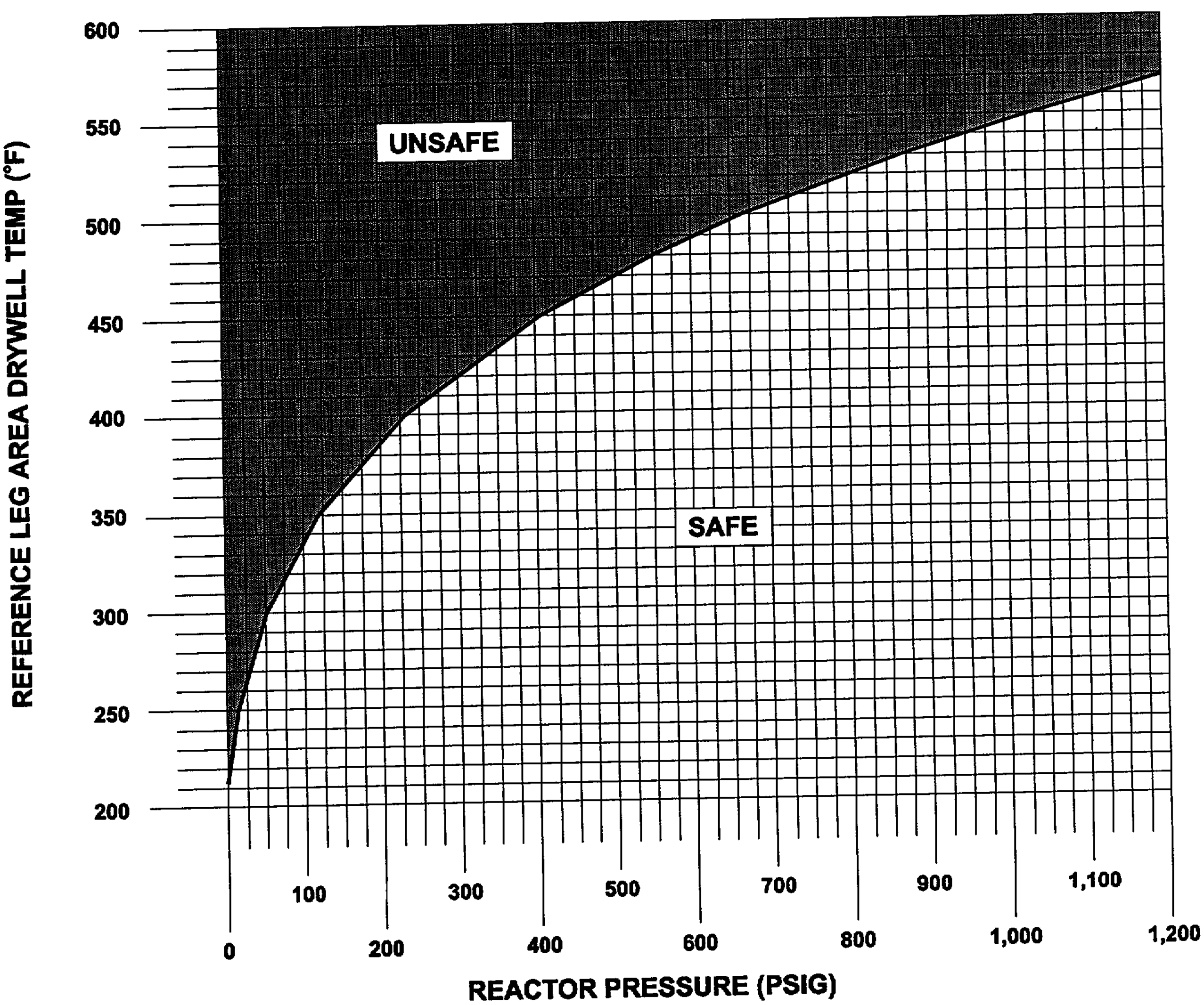
OR

Microprocessor

CAC-TY-4426-1 Point 5822 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5823 ______
CAC-TY-4426-2 Point 5824 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

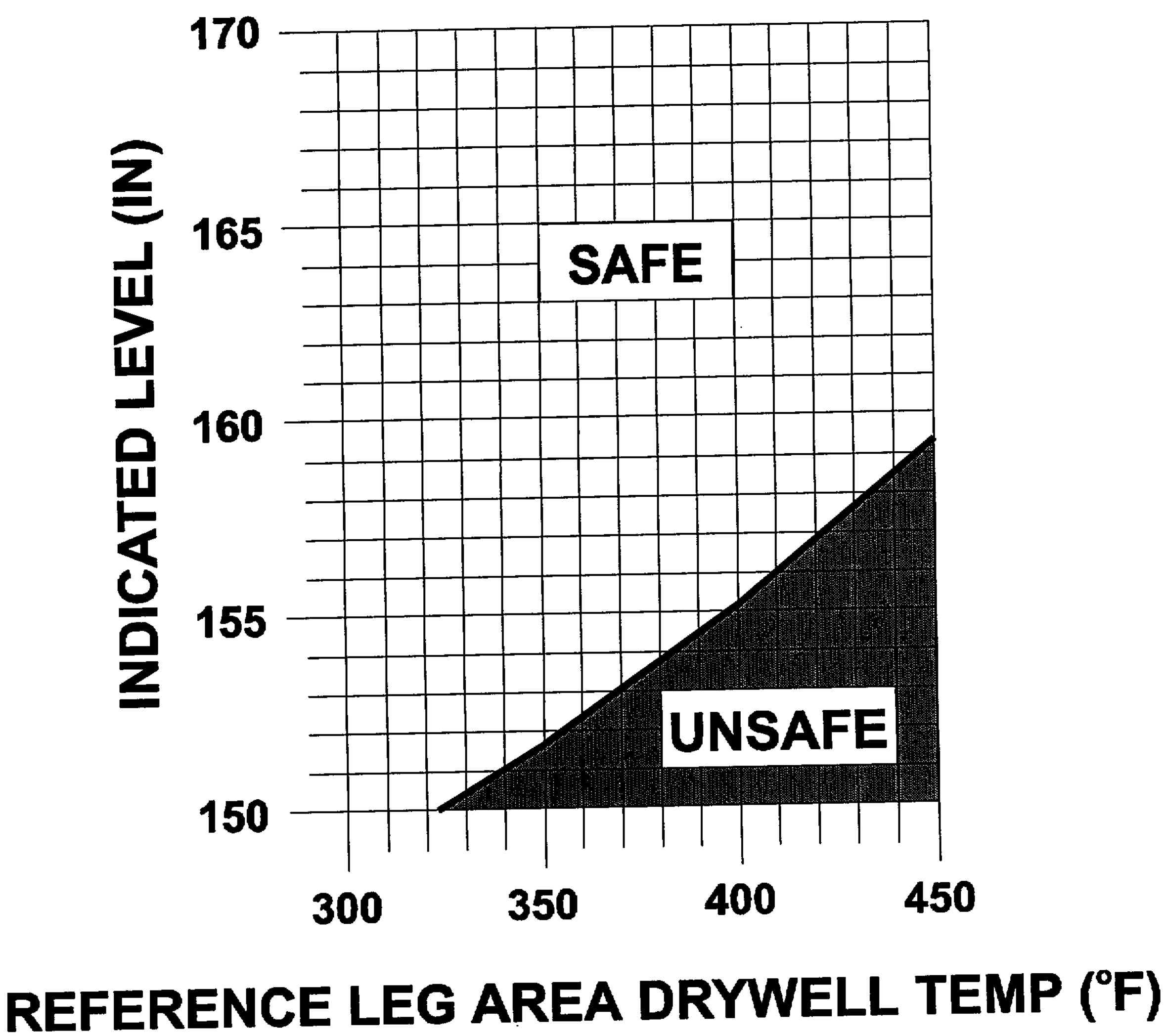
ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



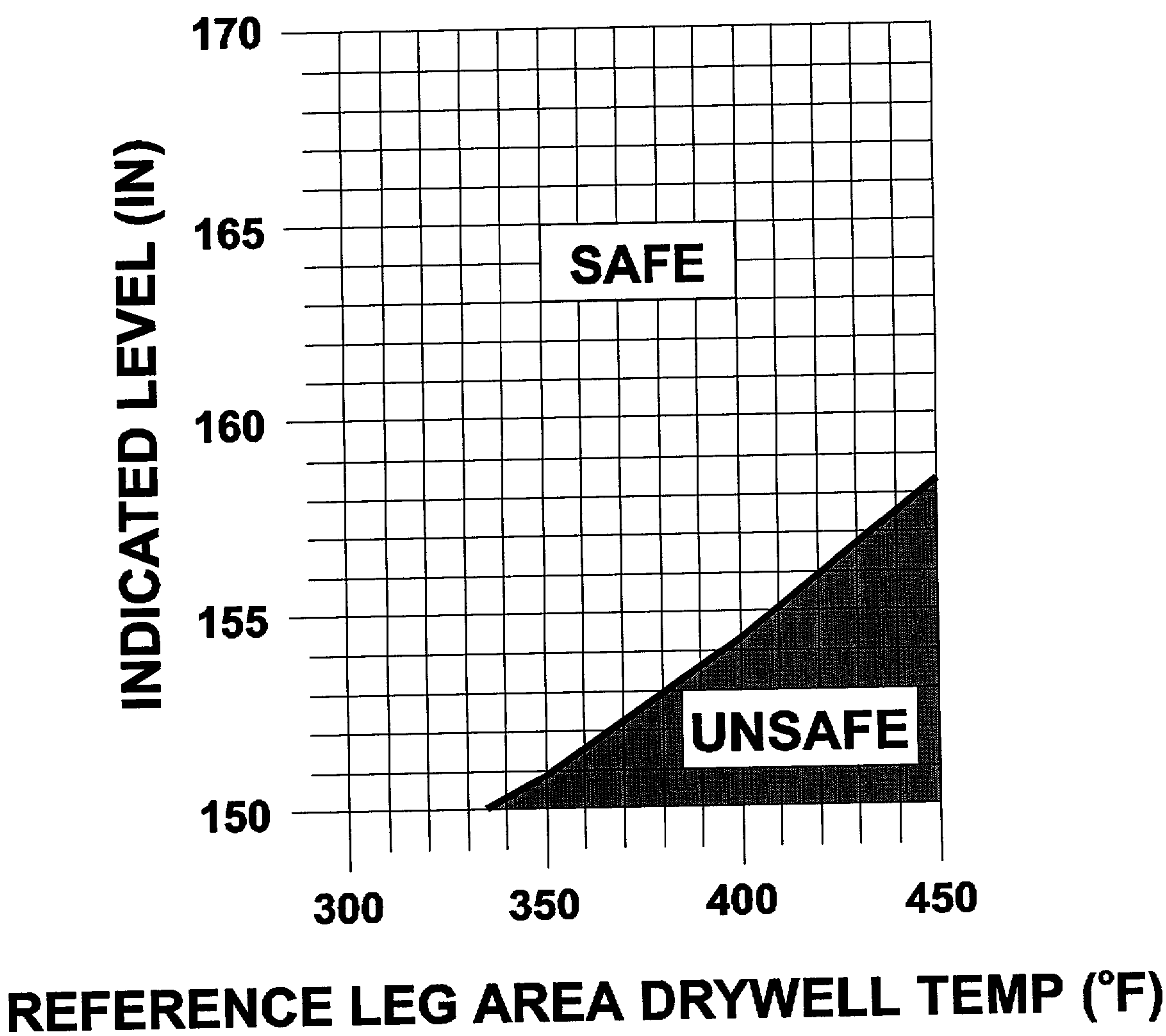
ATTACHMENT 6 (Cont'd)

FIGURE 15
UNIT 1 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



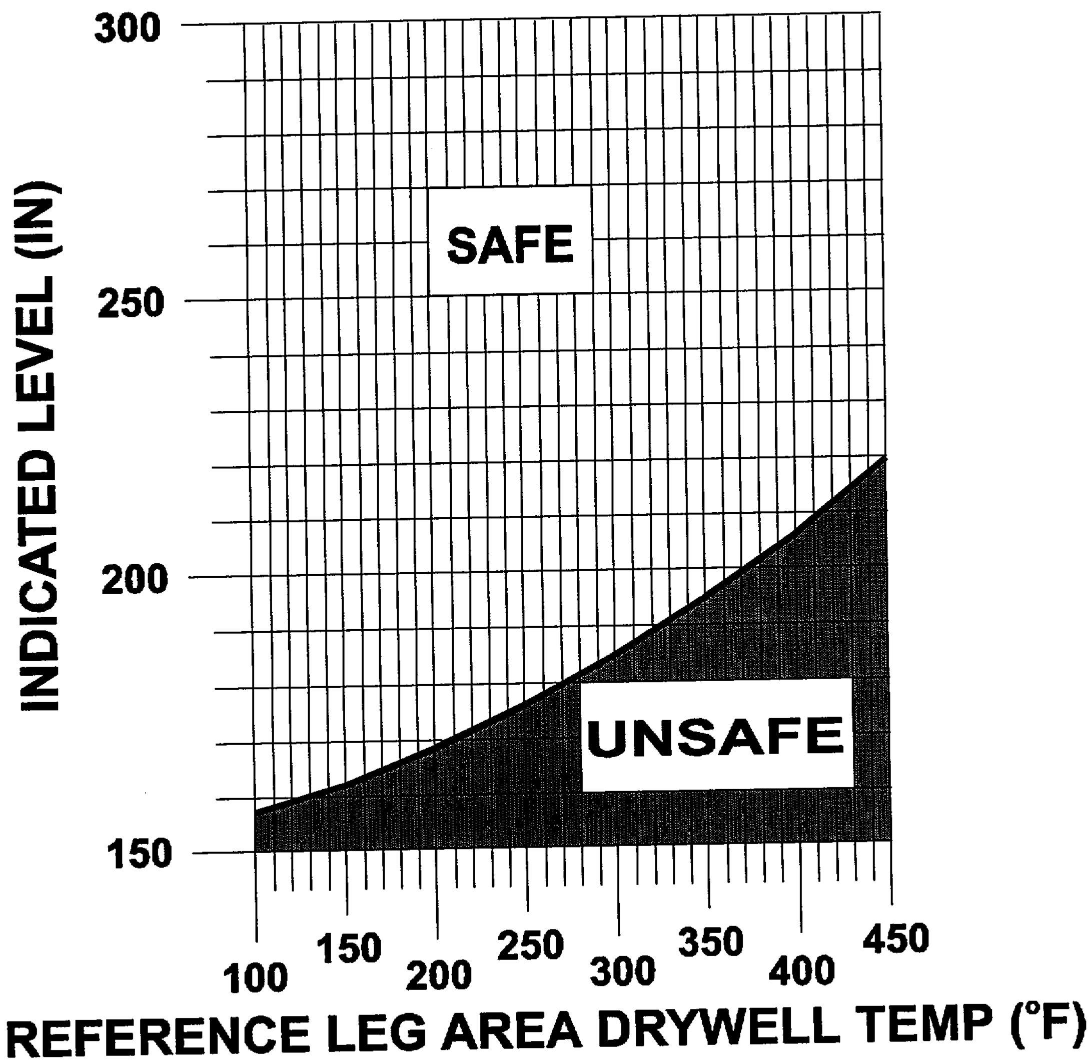
ATTACHMENT 6 (Cont'd)

FIGURE 15A
UNIT 2 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



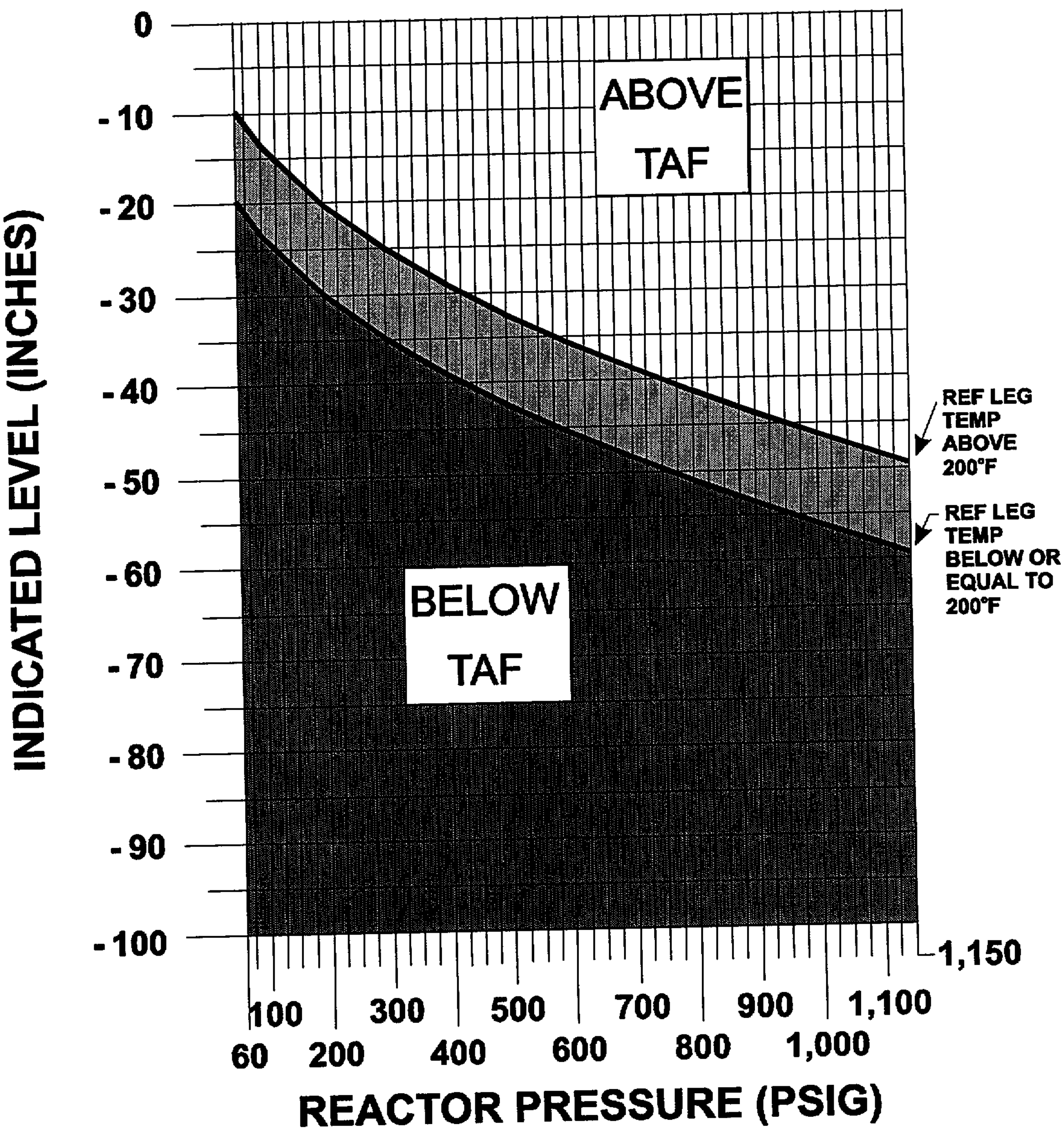
ATTACHMENT 6 (Cont'd)

FIGURE 16
SHUTDOWN RANGE LEVEL
INSTRUMENT (N027A, B) CAUTION



ATTACHMENT 6 (Cont'd)

FIGURE 17
UNIT 1 REACTOR WATER LEVEL AT TAF

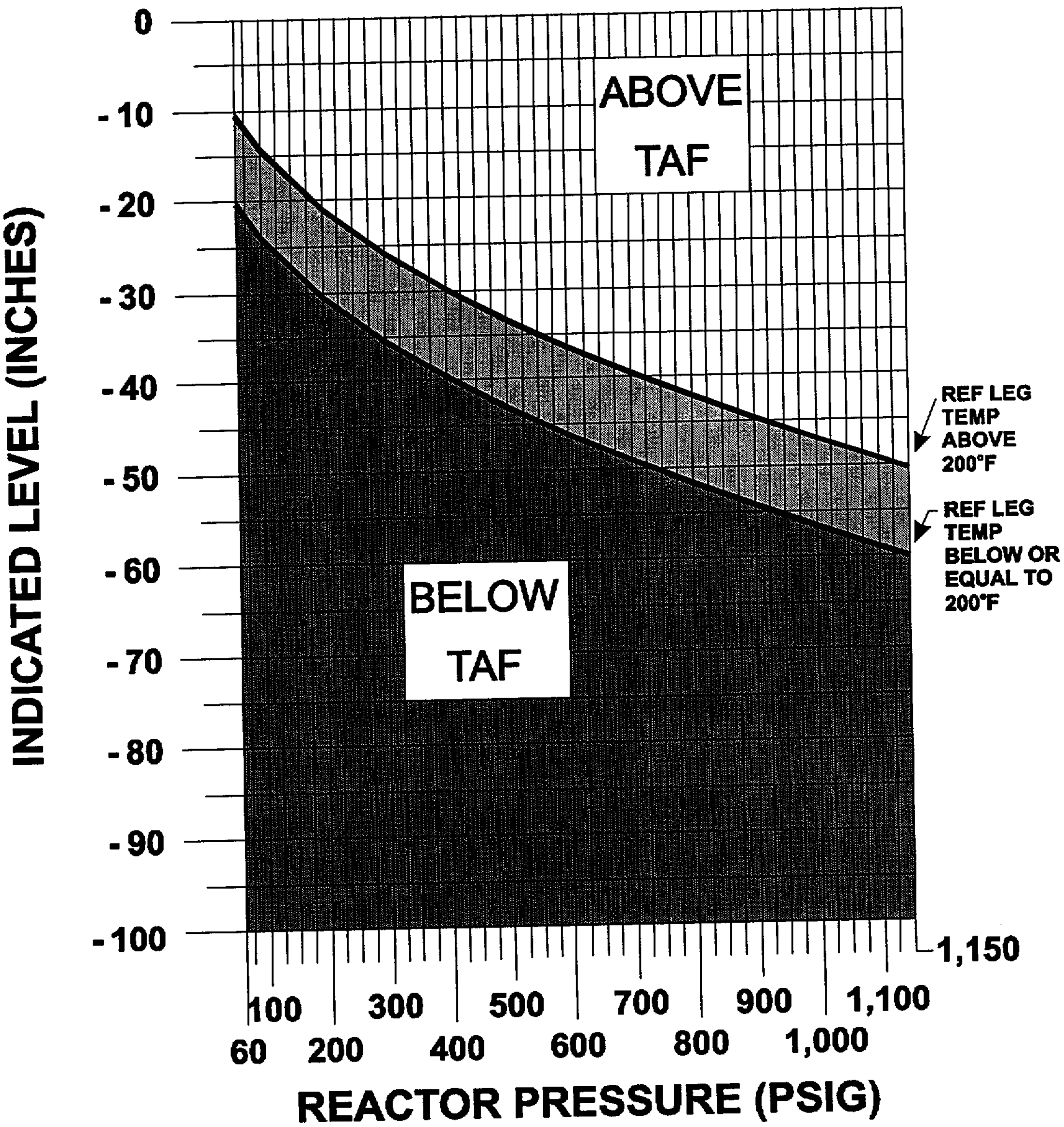


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 17A
UNIT 2 REACTOR WATER LEVEL AT TAF

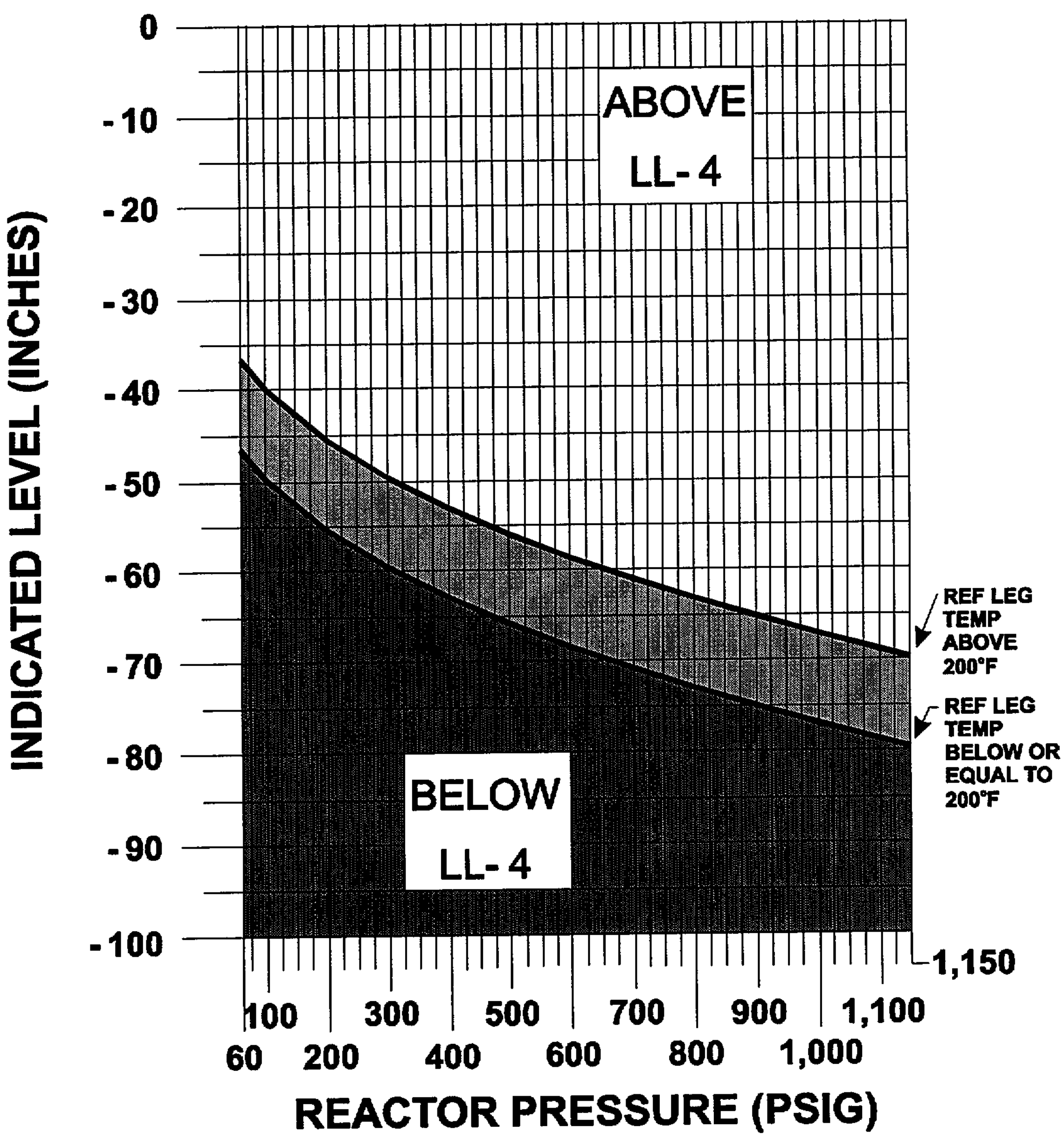


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18
UNIT 1 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

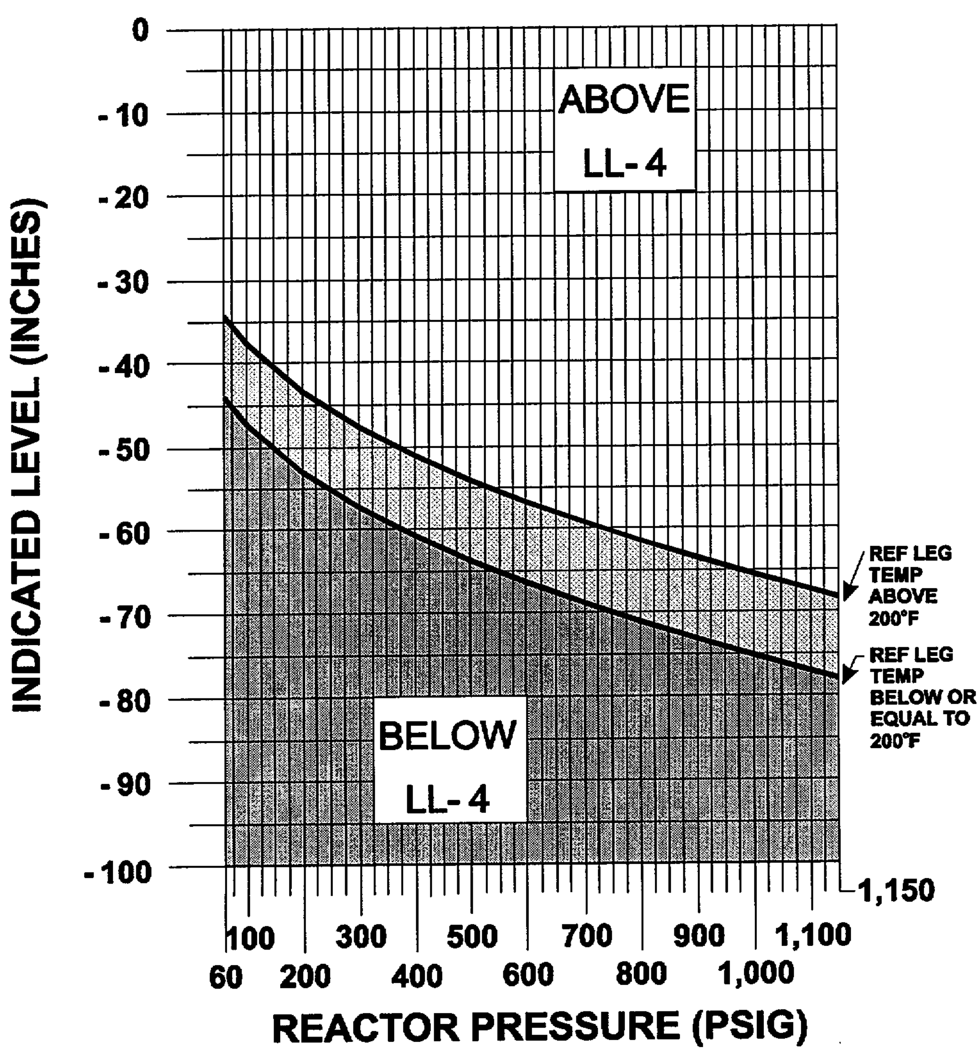


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -35 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18A
UNIT 2 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

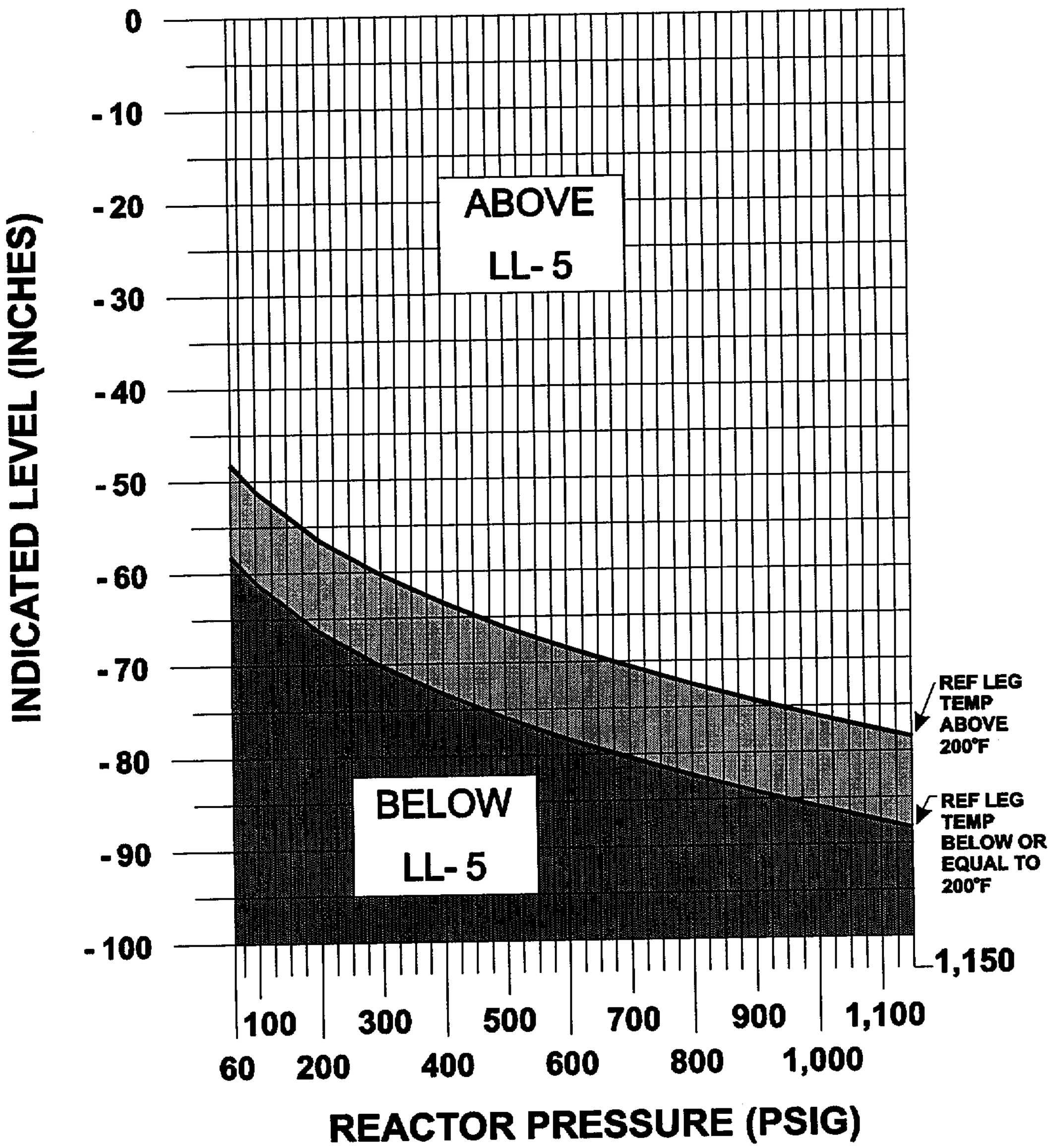


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -32.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19
UNIT 1 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

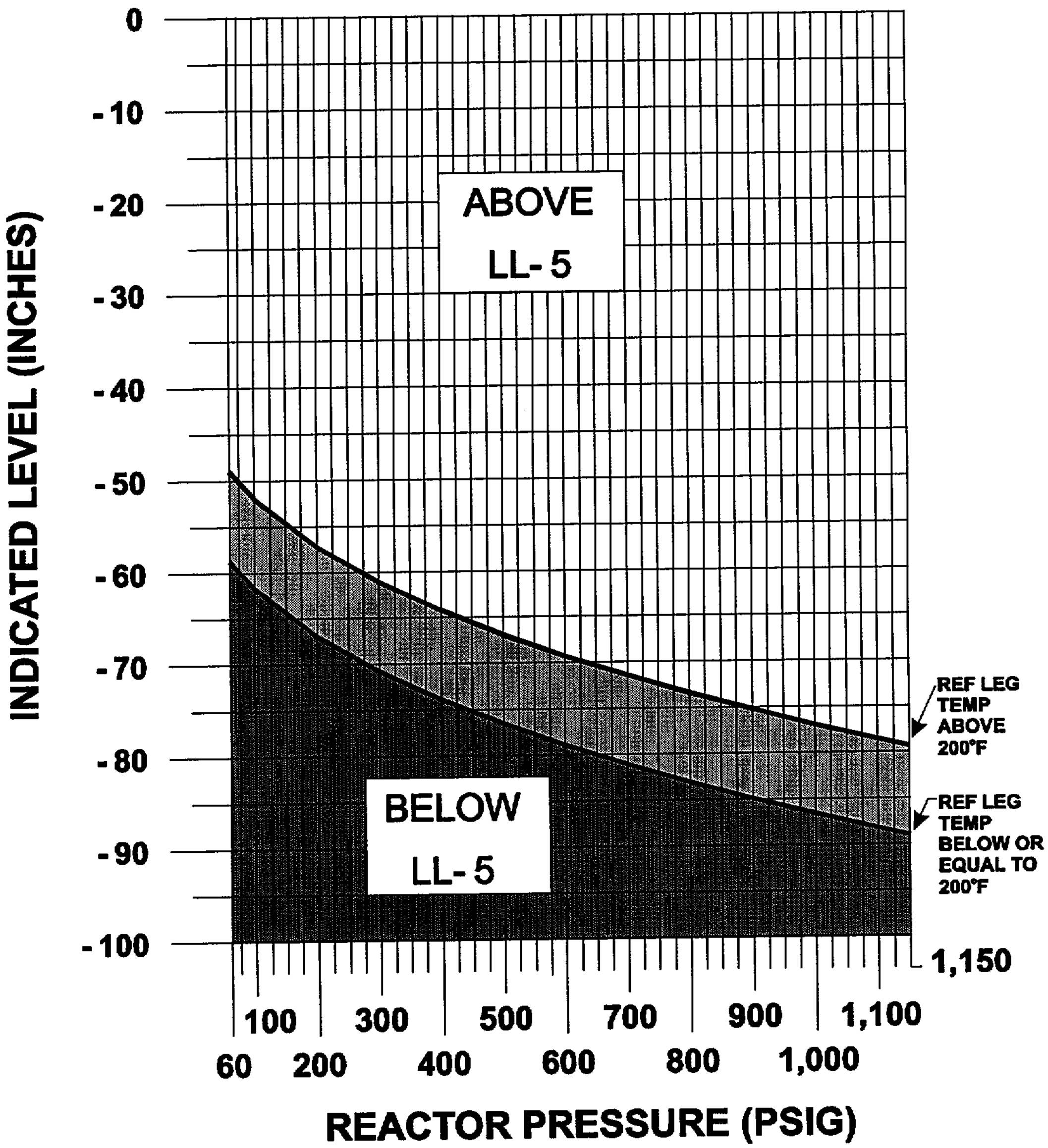


NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL-5 IS -47.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19A
UNIT 2 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

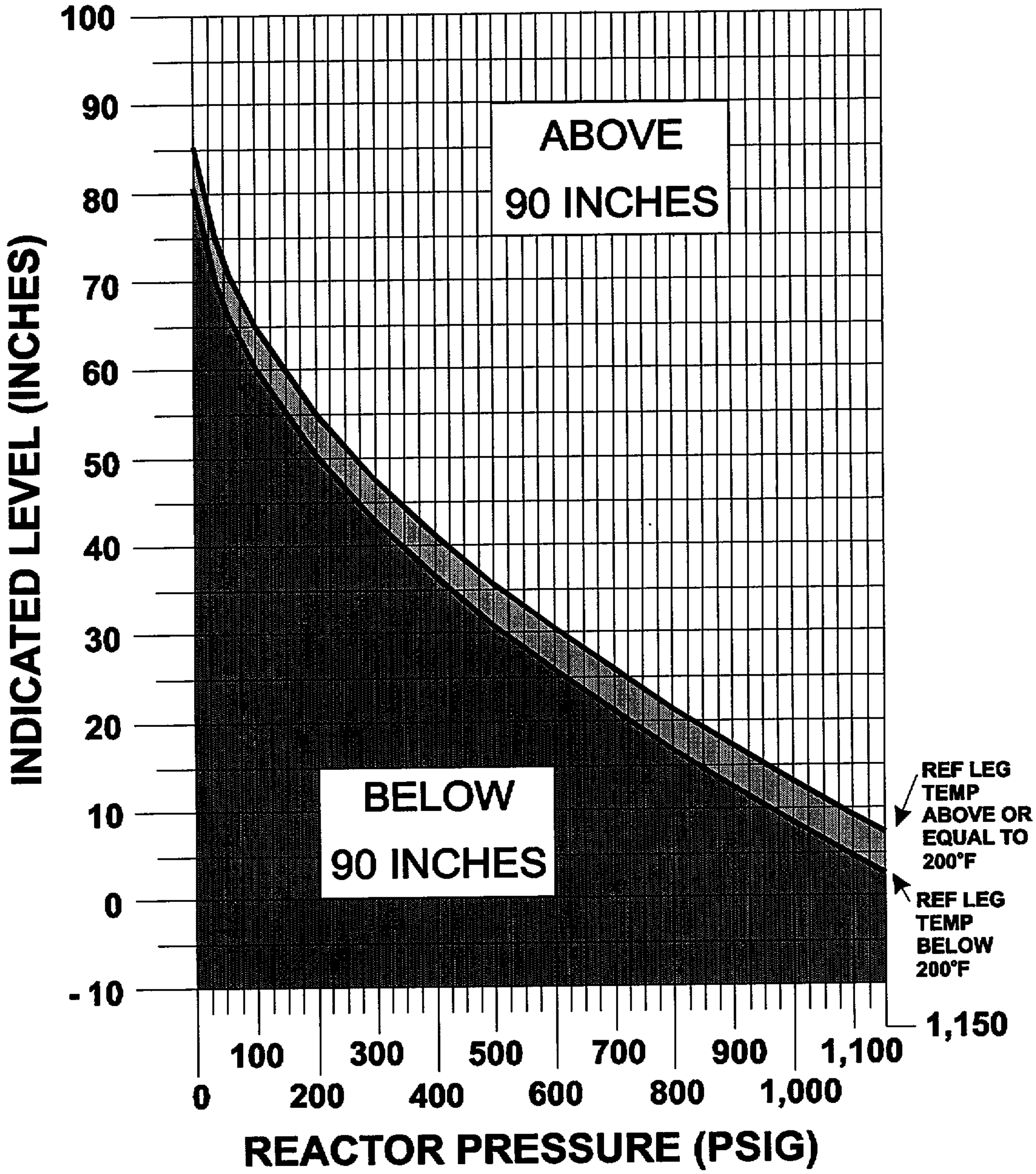


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-5 IS -47.5 INCHES.

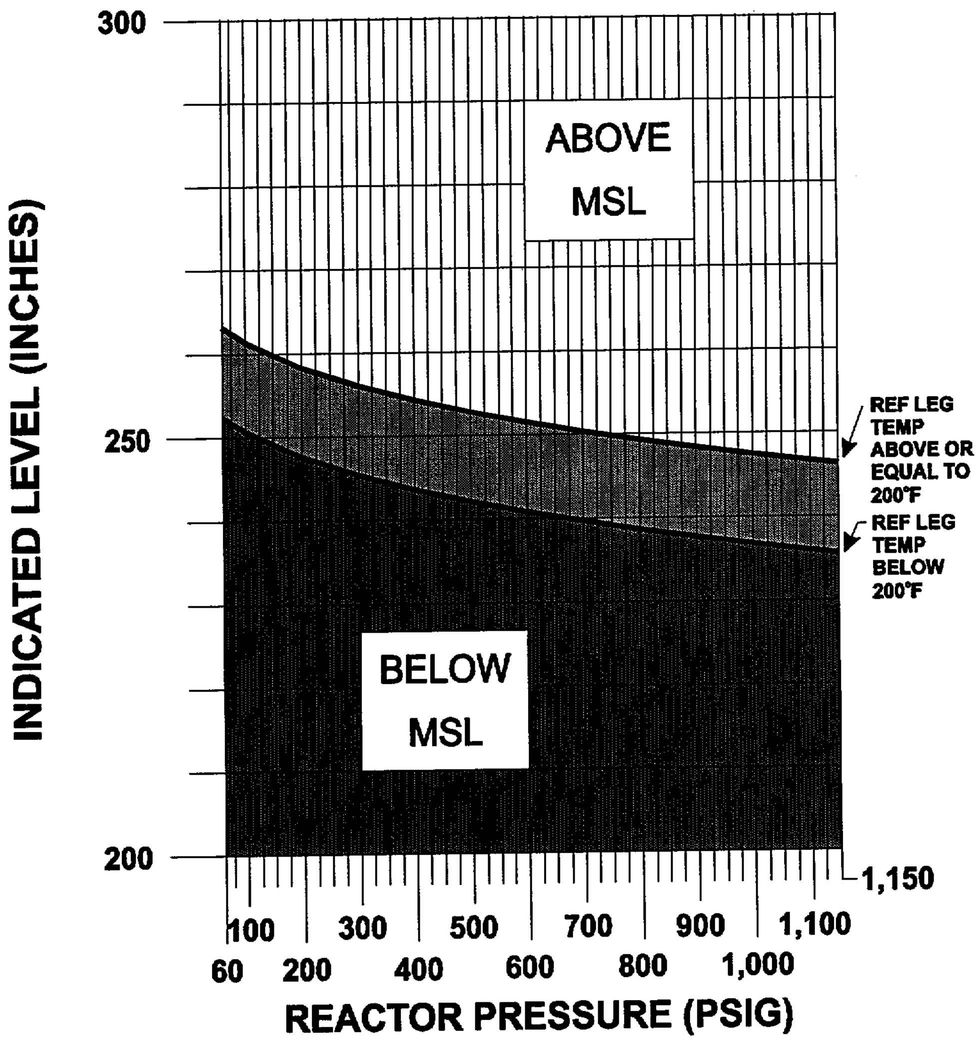
ATTACHMENT 6 (Cont'd)

FIGURE 20
REACTOR WATER LEVEL AT 90 INCHES



ATTACHMENT 6 (Cont'd)

FIGURE 21
REACTOR WATER LEVEL AT MSL
(MAIN STEAM LINE FLOOD LEVEL)



NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
MSL IS +250 INCHES.

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date: ____________

TABLE 1

TIME	APPROX. RX PWR	BYPASS VALVE % PWR	STEAM FLOW % PWR	LPRM % PWR	HEAT BALANCE % PWR	APRM GAFs ≤ 1.0	INIT.
	10%						
	20%						
N/A	TURBINE ON LINE	N/A	N/A	N/A	N/A	N/A	N/A

1. **USE** this attachment to validate the heat balance between 10% reactor power and placing the turbine on line, by performing the following:

   a. **OBTAIN** valid Heat Balance (Display 820, C067, C003, or 0PT-01.8D) **AND RECORD** heat balance % power in Table 1.

   b. **OBTAIN** LPRM % PWR (Display 815) **AND RECORD** in Table 1.

   c. **OBTAIN** Total Steam Flow (Mlb/hr).

Steam Line	(A)	(B)	(C)	(D)
(ERFIS)	C32FA014	C32FA015	C32FA016	C32FA017
(P603)	C32-R603A	C32-R603B	C32-R603C	C32-R603D

Total Steam Flow = (A) + (B) + (C) + (D) = ________

OR

USE Computer Point B041

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:____________

d. **OBTAIN** total bypass valve position (% open):

ERFIS computer point EHCXA002: __________%

OR

RTGB Panel XU1: (record the position (% open) of each)

Unit 1:

MS-BPV-1	MS-BPV-2	MS-BPV-3	MS-BPV-4
________%	________%	________%	________%

Total BPV % open = (The sum of adding BPV-1 through BPV-4) / 4 = __________%

Unit 2:

MS-BPV-1	MS-BPV-2	MS-BPV-3	MS-BPV-4	MS-BPV-5
________%	________%	________%	________%	________%
MS-BPV-6	MS-BPV-7	MS-BPV-8	MS-BPV-9	MS-BPV-10
________%	________%	________%	________%	________%

Total BPV % open = (The sum of adding BPV-1 through BPV-10) / 10 = __________%

> **NOTE:** Typing MAN runs an interactive program called MAN_ALTDSP that performs alternate power calculations based upon user supplied plant inputs. Remember to enter decimal points for all values. Use the equivalent % power output from this program for the comparison in the next step.

e. **LOG** on to the ERFIS terminal on the SRO's desk:

1) **TYPE:** SET HOST EC01B (EC02B)

OR

SET HOST EC01A (EC02A)

2) **TYPE:** GEPACUSER, at USERNAME prompt

3) **TYPE:** GEPAC, at PASSWORD prompt

4) **TYPE:** MAN (for manual input)

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:___________

> **NOTE:** The equivalent % power output from this program is to be used for the comparison in the next step.

f. **RECORD** BYPASS valve, and STEAM FLOW % power alternate indications on Table 1 of this attachment using the values obtained from MAN_ALTDSP in Step e.

g. **COMPARE** the Heat Balance (%) with the other alternate indications (%) recorded on Table 1.

Power Ascension may continue **IF**:

1) The heat balance is greater than all alternate indications (conservative as is).

OR

2) One or more alternate indications are within ± 5% of the heat balance (normal acceptance).

OR

3) There are no alternate indications within ± 5% of the heat balance provided the APRMs are adjusted greater than or equal to the next highest alternate indication above the heat balance (conservative action) in accordance with 1(2)OP-09.

OTHERWISE, STOP power ascension **AND CONTACT** Reactor Engineering to account for the differences in agreement.

h. **REPEAT** the above steps at 10% increments until the turbine is placed on line.

2. Definitions for Table 1:

HEAT BALANCE % POWER - A calculation of core thermal power obtained by solving an energy balance on the reactor vessel. Valid heat balance calculations may be obtained from Display 820 edits, Process Computer Points C003 and C067, or manually by performing 0PT-01.8D. Caution must be taken to ensure any failed sensors have valid substituted values.

Verification Of Reactor Power Level Using Alternate Indications

Unit ____ Date:____________

LPRM % POWER - An alternate indication of reactor power calculated only on the process computer which is obtained by averaging calibrated LPRM readings. This uses points U1NSSALTP and U2NSSALTP.

STEAM FLOW % POWER - An alternate indication of reactor power obtained by correlating the total steam flow to a valid heat balance. Total steam flow can be obtained from Plant Process Computer Point B041, the summation of ERFIS points C32FA014, C32FA015, C32FA016, C32FA017 (main steam lines A, B, C, D), or the summation of RTGB indications C32-R603A, B, C, D, on P603.

BYPASS VALVE % POWER - An alternate indication of reactor power obtained by correlating the total bypass valve position (percent open) to a valid heat balance. Total bypass valve position can be obtained from ERFIS point EHCXA002, or RTGB Panel XU1.

CALCULATION SHEET 1

Values Obtained From Recorder CAC-TR-778

80' elev

PT No. 1	____	____	____	____	____	____	____	____
	x 0.14	x 0.14	x 0.14	x 0.14	x 0.14	x 0.14	x 0.14	x 0.14
	A ____	A ____	A ____	A ____	A ____	A ____	A ____	A ____

28' elev

PT No. 3	____	____	____	____	____	____	____	____
	x 0.4	x 0.4	x 0.4	x 0.4	x 0.4	x 0.4	x 0.4	x 0.4
	B ____	B ____	B ____	B ____	B ____	B ____	B ____	B ____

13' elev

PT No. 4	____	____	____	____	____	____	____	____
	x 0.46	x 0.46	x 0.46	x 0.46	x 0.46	x 0.46	x 0.46	x 0.46
	C ____	C ____	C ____	C ____	C ____	C ____	C ____	C ____

Add the numbers obtained in lines A, B, and C, to obtain average DW temp.

Average DW Temp ____ ____ ____ ____ ____ ____ ____ ____

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 10 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel <u>AND</u> (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	(continued)	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days
SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	120 days cumulative operation in MODE 1
SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig.	Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES WHEN REACTOR STEAM DOME PRESSURE $\geq$ 800 psig[a][b] (seconds)
46	0.44
36	1.08
26	1.83
06	3.35

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) When reactor steam dome pressure is < 800 psig, established scram time limits apply.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 950 psig.	A.1	--------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ---------------------	
		Declare the associated control rod scram time "slow."	8 hours
	OR		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure $\geq$ 950 psig.	B.1 Restore charging water header pressure to $\geq$ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 950 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	<u>AND</u>	
	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action B.1 or C.1 and associated Completion Time not met.	D.1 --------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------- Manually scram the reactor.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ 940 psig.	7 days

BRUNSWICK UNIT 2, CYCLE 15

CORE OPERATING LIMITS REPORT

March 2001

Prepared By: Charles Stroupe **Date:** 3-5-01
Charles Stroupe

Approved By: G E Smith **Date:** 3-6-01
George E. Smith
Superintendent
BWR Fuel Engineering

LIST OF EFFECTIVE PAGES

Page(s)	Revision
1-34	0

TABLE OF CONTENTS

CAUTION
References to COLR Figures or Tables should be made using titles only; figure and table numbers may change from cycle to cycle.

LIST OF TABLES

LIST OF FIGURES

Introduction and Summary

This report provides the values of the power distribution limits and control rod withdrawal block instrumentation setpoints for Brunswick Unit 2, Cycle 15 as required by TS 5.6.5.

OPERATING LIMIT	REQUIREMENT
Average Planar Linear Heat Generation Rate (APLHGR) limits (with associated core flow and core power adjustment factors)	TS 5.6.5.a.1
Minimum Critical Power Ratio (MCPR) limits (with associated core flow and core power adjustment factors)	TS 5.6.5.a.2
Allowable Values for Function 2.b of TS 3.3.1.1, APRM Flow Biased Simulated Thermal Power –High	TS 5.6.5.a.3
Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions of TS 3.3.2.1	TS 5.6.5.a.4

Per TS 5.6.5.b and 5.6.5.c, these values have been determined using NRC approved methodology and are established such that all applicable limits of the plant safety analysis are met.

The limits specified in this report support single loop operation (SLO) as required by TS LCO 3.4.1 and inoperable Main Turbine Bypass System as required by TS 3.7.6.

In order to support the Thermal Hydraulic Instability (THI) E1A Stability Solution, the following is also included in this report:

OPERATING LIMIT	REQUIREMENT
Thermal Hydraulic Instability (THI) E1A Stability Solution Monitored Region and Restricted Region	TS 3.2.3 and 3.3.1.3, and TRMS 3.2
Thermal Hydraulic Instability (THI) E1A Stability Solution Exclusion Region	Implicit
"Setup" and "Non-Setup" scram values of the APRM Flow Biased Simulated Thermal Power-High Allowable Value ("Flow Biased Scram")	TS 3.2.3 and 3.3.1.1
"Setup" and "Non-Setup" control rod block values of the APRM Flow Biased - Upscale Allowable Value ("Flow Biased Rod Block")	TRMS 3.3

Four Siemens ATRIUM-10 (A10) Lead Qualification Assemblies will be loaded in the B2C15 core. Reference 4 concludes the A10 is bounded by the GE13 operating limits and licensing analyses, provided additional operating and design constraints are imposed on the GE13 fuel type used to monitor the A10. The additional operating requirements have been incorporated herein as applicable.

Preparation of this report was performed in accordance with Quality Assurance requirements as specified in Reference 1.

Single Loop Operation

Brunswick Unit 2, Cycle 15 may operate at any point in the cycle over the entire MEOD range with Single Recirculation Loop Operation (SLO) as permitted by TS 3.4.1 with applicable limits specified in the COLR for TS LCO's 3.2.1, 3.2.2 and 3.3.1.1:

LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR) Limits: per Reference 1 and Figures 9, 10 and 10a, the APLHGR Limits include a SLO limitation of 0.8 on the MAPLHGR(F) and MAPLHGR(P) multipliers.

LCO 3.2.2, Minimum Critical Power Ratio (MCPR) Limits: per Reference 1, Table 1 and Figures 11, 11a, 12 and 12a, the MCPR limits presented apply to SLO without modification.

LCO 3.3.1.1, Reactor Protection System Instrumentation Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High) Allowable Value: per Reference 1 and the THI E1A STABILITY SOLUTION, these limits apply to SLO without modification.

Inoperable Main Turbine Bypass System

Brunswick Unit 2, Cycle 15 may operate with an inoperable Main Turbine Bypass System in accordance with TS 3.7.6 with applicable limits specified in the COLR for TS LCO 3.2.1 and 3.2.2. Three or more bypass valves inoperable renders the System inoperable, although the Turbine Bypass Out-of-Service (TBPOOS) analysis supports operation with all bypass valves inoperable for the entire MEOD range and up to 110°F rated equivalent feedwater temperature reduction. The system response time assumed by the safety analyses from event initiation to start of bypass valve opening is 0.10 seconds, with at least 64% bypass flow achieved in 0.30 seconds. The applicable limits are as follows:

LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR) Limits: in accordance with Reference 1 as shown in Figure 10, TBPOOS requires a reduction in the MAPLGHR(P) limits between 25% and 30% power. The limits in Figure 10a between 25% and 30% power are valid for TBP operable or inoperable.

LCO 3.2.2, Minimum Critical Power Ratio (MCPR) Limits: in accordance with Reference 1, TBPOOS requires an increase in the MCPR(P) multiplier between 25% and 30% power, as shown in Figures 12. This increase is already identified in Figure 12a. TBPOOS also requires increased MCPR limits, included in Table 1.

APLHGR Limits

The limiting APLHGR value for the most limiting lattice (excluding natural uranium) of each fuel type as a function of planar average exposure is given in Figures 1 through 7. These values were determined with the SAFER/GESTR LOCA methodology described in GESTAR-II (Reference 2). Figures 1 through 7 are to be used only when hand calculations are required as specified in the bases for TS 3.2.1. Hand calculated results may not match a POWERPLEX calculation since normal monitoring of the APLHGR limits with POWERPLEX uses the complete set of lattices for each fuel type provided in Reference 3.

The core flow and core power adjustment factors for use in TS 3.2.1 are presented in Figures 9, 10 and 10a. For any given flow/power state, the minimum of MAPLHGR(F) determined from Figure 9 and MAPLHGR(P) determined from Figures 10 and 10a is used to determine the governing limit.

MCPR Limits

The ODYN OPTION A, ODYN OPTION B and non-pressurization transient MCPR limits for use in TS 3.2.2 for each fuel type as a function of cycle average exposure are given in Table 1. These values were determined with the GEMINI methodology and GEXL-PLUS critical power correlation described in GESTAR-II (Reference 2) and are consistent with a Safety Limit MCPR of 1.09 specified by TS2.1.1.2.

The core flow and core power adjustment factors for use in TS 3.2.2 are presented in Figures 11, 11a, 12 and 12a. For any given power/flow state, the maximum of MCPR(F) determined from Figure 11 or 11a and MCPR(P) determined from Figure 12 or 12a is used to determine the governing limit.

All MCPR limits presented in Table 1, Figure 11, Figure 11a, Figure 12 and Figure 12a were determined without EOC-RPT operable and apply to two recirculation pump operation and SLO without modification.

RBM Rod Block Instrumentation Setpoints

The nominal trip setpoints and allowable values of the control rod withdrawal block instrumentation for use in TS 3.3.2.1 (Table 3.3.2.1-1) are presented in Table 2. These values were determined consistent with the bases of the ARTS program and the determination of MCPR limits with the GEMINI methodology and GEXL-PLUS critical power correlation described in GESTAR-II (Reference 2).

THI E1A Stability Solution

The Enhanced Option 1A methodology was used to develop the THI E1A Stability Solution, which involves exclusion from certain areas of the power/flow map and specific restrictions for operating in other areas.

The COLR provides the Stability Regions on the power/(core) flow map in Figures 13-16. These Figures define the Monitored and Restricted Regions for compliance with TS 3.2.3, TS 3.3.1.3 and TRMS 3.2 (and indirectly TS 3.3.1.1 and TRMS 3.3), and include the Exclusion Region (for which definition in the COLR is not a TS requirement). Core flow nominal trip setpoint values on Figures 13-16 correspond to the nominal trip setpoint values translated into drive flow and installed in the Flow Control Trip Reference (FCTR) cards.

Automatic features of the THI E1A Stability Solution implementation use digital FCTR cards that incorporate Trip Reference setpoints which are equivalent or more restrictive than the pre-Stability Solution APRM flow-biased and clamped limits. The FCTR cards support TS 3.3.1.1 (automatic APRM Flow-biased Scram) and TRMS 3.2 (Restricted Region Entry Alarm, which uses the TRMS 3.3 Flow-biased Rod Block setpoint). Figures 17-20, E1A Setpoint Allowable Values Versus Aligned Drive Flow, are based on drive flow and not core flow to support the flow signal used for the FCTR cards. Also, Figures 17-20 allow quantification of Technical Specification compliance once the drive flow input is aligned in accordance with Table 3.

"Non-Setup" setpoints (Figures 13, 15, 17, 19) enforce the normal Exclusion and Restricted Regions described above. Setup setpoints (Figures 14, 16, 18, 20) are to be used only when FCBB ≤ 1.0 and allow operation in the Restricted Region. When operating in Setup, the Flow-biased Rod Block setpoints generally increase in power to the Flow-biased Scram or power/flow map boundaries. The Flow-biased Scram setpoint generally increases by an equivalent amount (within the power/flow map boundaries) to avoid spurious scrams from power spikes. The inherent stability from maintaining FCBB less than one justifies continued operation in the Restricted Region, but not in that portion of the power/flow map which, in Setup, becomes unprotected by the Flow-biased Scram. The alarm associated with the Rod Block ceases to be a RREA when in Setup, but signals to Operations a similar need to immediately move to a more stable region of the power/flow map.

For BNP the two loop operation (TLO) Flow-biased Scram and Rod Block setpoints, and TLO Stability Regions, are equivalent to the SLO counterparts over all applicable portions of the operating domain.

The E1A Stability Solution provides for distinct Flow-biased Scram and Rod Block setpoints for normal and reduced feedwater temperature conditions ("normal" and "alternate" setpoints) because the core is more susceptible to instabilities with decreasing feedwater temperature. Normal setpoints (Figures 13, 14, 17, 18) are to be used below 30% power or when feedwater temperature is within 50°F rated equivalent of nominal. Alternate setpoints (Figures 15, 16 19, 20) are to be used above 30% power when feedwater is reduced by more than 50°F rated equivalent ($50°F * (\% \text{ power}/100)^{0.385}$) in accordance with 2OP-32.

References

1) BNP Design Calculation 2B21-0585; "Preparation of the B2C15 Core Operating Limits Report," Revision 0, February 2001.

2) NEDE-24011-P-A; "General Electric Standard Application for Reactor Fuel," (latest approved version).

3) NEDC-31624P, "Loss-of-Coolant Accident Analysis Report for Brunswick Steam Electric Plant Unit 2 Reload 14 Cycle 15," Supplement 2, Revision 7, February 2001.

4) EMF-2168(P), "Brunswick ATRIUM-10 Lead Qualification Assemblies Safety Analysis," Revision 0, March 1999.

Table 1

MCPR Limits

(EOC-RPT Not Required)

Steady State, Non-pressurization Transient MCPR Limits	
Fuel Type	Exposure Range: BOC - EOC
GE13 and GE14	1.22
A10	1.36

Pressurization Transient MCPR Limits, OLMCPR (100%P): Turbine Bypass System Operable

MCPR Option	Fuel Type	Normal and Reduced Feedwater Temperature Exposure Range: BOC to EOFPC-2101 MWd/MT	Normal and Reduced Feedwater Temperature Exposure Range: EOFPC-2101 MWd/MT to EOC
A	GE13	1.40	1.46
	GE14	1.52	1.66
	A10	1.56	1.62
B	GE13	1.35	1.38
	GE14	1.41	1.49
	A10	1.50	1.53

Pressurization Transient MCPR Limits, OLMCPR (100%P): Turbine Bypass System Inoperable

MCPR Option	Fuel Type	Normal and Reduced Feedwater Temperature BOC to EOC
A	GE13	1.47
	GE14	1.68
	A10	1.63
B	GE13	1.39
	GE14	1.51
	A10	1.55

This Table is referred to by Technical Specifications 3.2.2, 3.4.1 and 3.7.6.

Table 2

RBM System Setpoints

Setpoint	Trip Setpoint	Allowable Value
Lower Power Setpoint (LPSP[a])	27.0	≤ 29.0
Intermediate Power Setpoint (IPSP[a])	62.0	≤ 64.0
High Power Setpoint (HPSP[a])	82.0	≤ 84.0
Low Trip Setpoint (LTSP[b])	≤ 115.1	≤ 115.5
Intermediate Trip Setpoint (ITSP[b])	≤ 109.3	≤ 109.7
High Trip Setpoint (HTSP[b])	≤ 105.5	≤ 105.9
t_{d2}	≤ 2.0 seconds	≤ 2.0 seconds

a Setpoints in percent of Rated Thermal Power.

b Setpoints relative to a full scale reading of 125. For example, ≤ 115.1 means ≤ 115.1/125.0 of full scale.

This Table is referred to by Technical Specification 3.3.2.1 (Table 3.3.2.1-1).

Table 3

Aligned Drive Flow

The Scram and Rod Block trip setpoints are provided by Flow Control Trip Reference (FCTR) cards. The FCTR cards have their drive flows calibrated each cycle by 0PT-50.10, "APRM FCTR Card Drive Flow Alignment". The calibration "aligns" the current cycle drive flow to the drive flow used when the E1A flow mapping solution was developed for BNP. The COLR presents the Scram and Rod Block trip setpoints as a function of aligned drive flow. This table provides an equation for deriving the aligned drive flow from the FCTR card input drive flow signal:

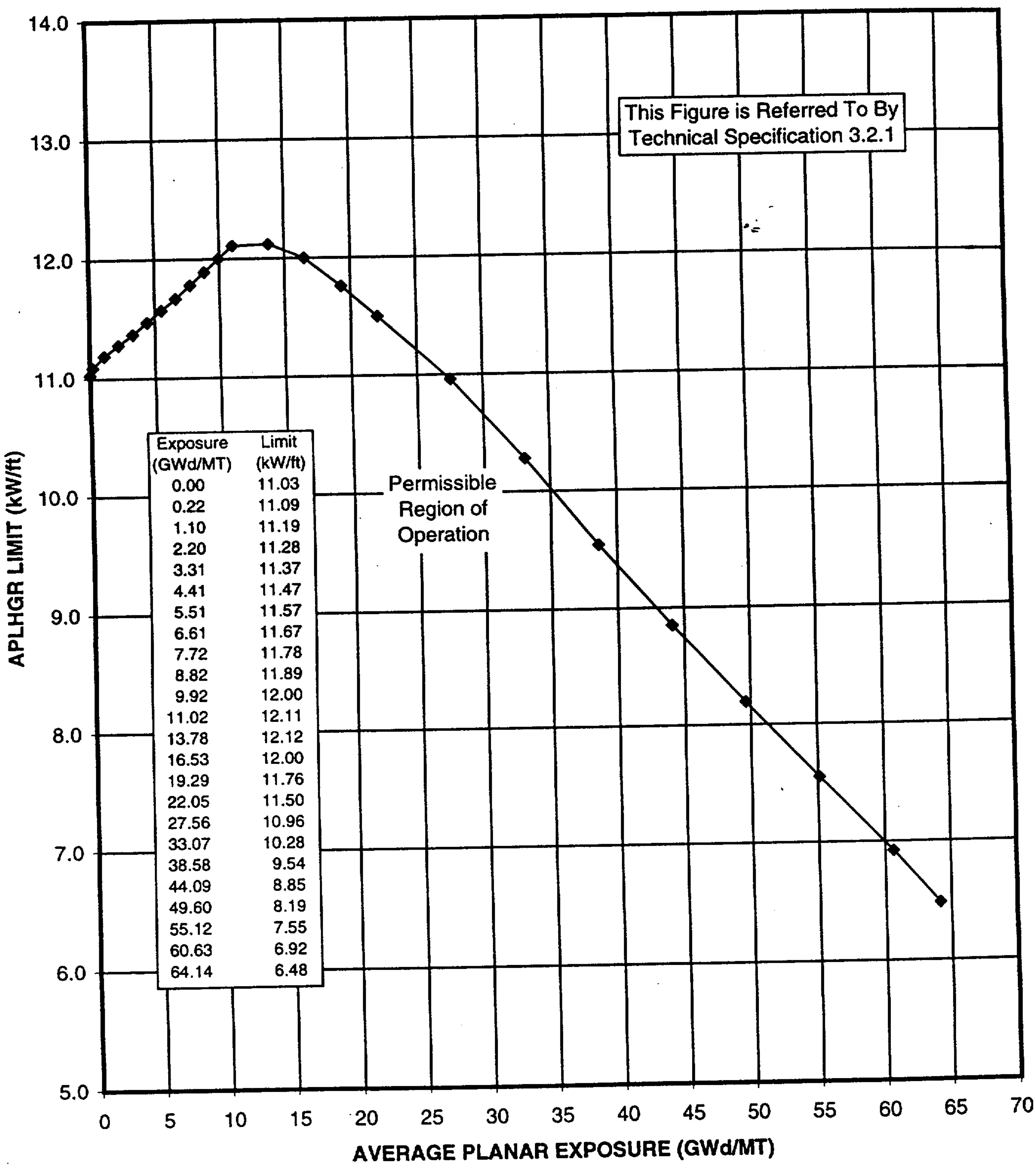
$$W_D = \frac{100.005 \cdot \Delta^{40} - 30.294 \cdot \Delta^{100} + 69.711 \cdot W_{\bar{D}}}{69.711 - (\Delta^{100} - \Delta^{40})}$$

where: W_D is the aligned drive flow to be used for Figures 17 through 20
Δ^{40} and Δ^{100} are the current values for the FCTR card alignment
$W_{\bar{D}}$ is the input drive flow signal

This Table supports Technical Specifications 3.2.3 and 3.3.1.1 and Technical Requirements Manual Specifications 3.2 and 3.3.

Figure 1

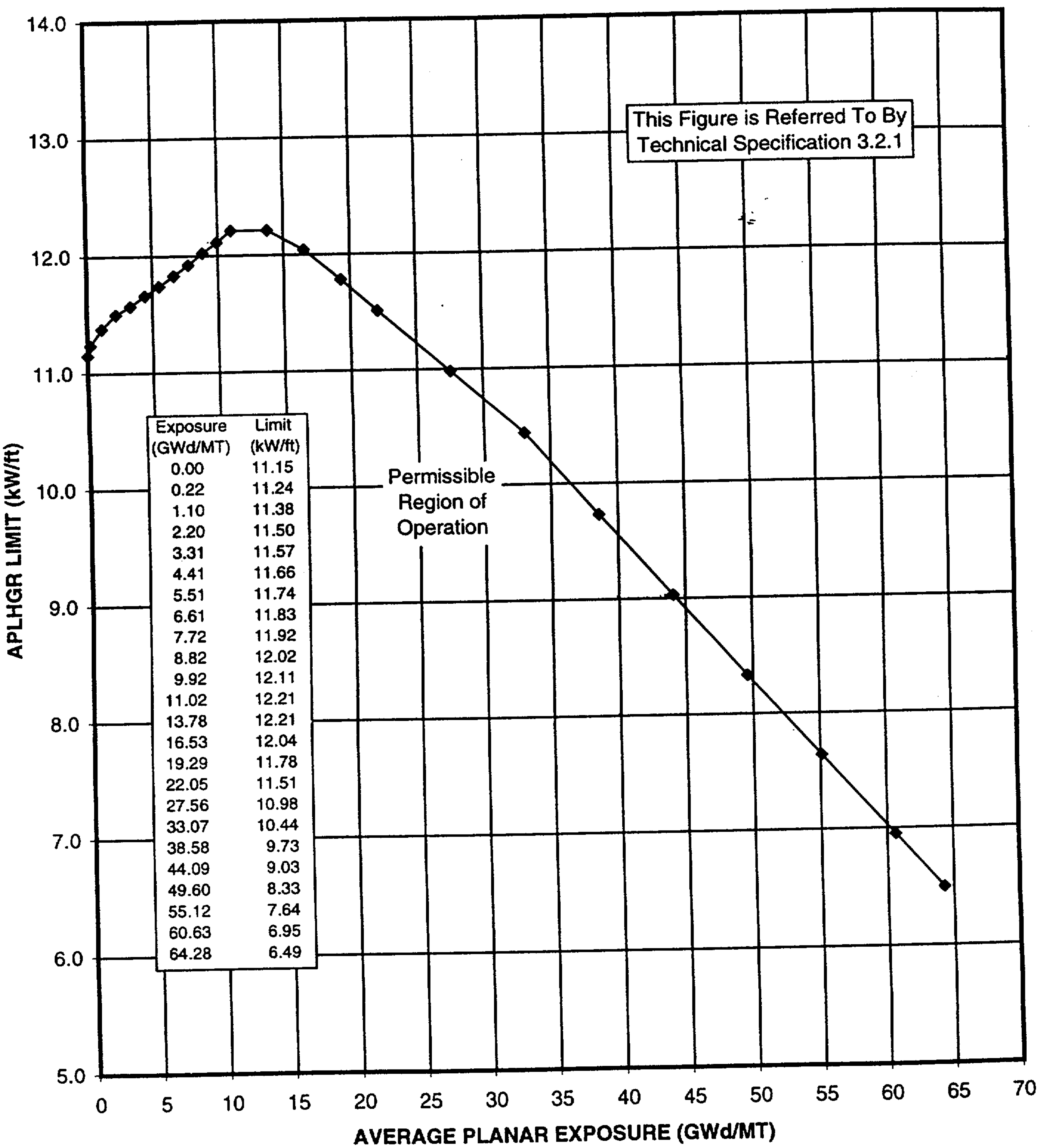
**Fuel Type GE13-P9DTB393-4G6.0/9G5.0-100T-146-T (GE13)
Average Planar Linear Heat Generation Rate (APLHGR) Limit
Versus Average Planar Exposure**



Exposure (GWd/MT)	Limit (kW/ft)
0.00	11.03
0.22	11.09
1.10	11.19
2.20	11.28
3.31	11.37
4.41	11.47
5.51	11.57
6.61	11.67
7.72	11.78
8.82	11.89
9.92	12.00
11.02	12.11
13.78	12.12
16.53	12.00
19.29	11.76
22.05	11.50
27.56	10.96
33.07	10.28
38.58	9.54
44.09	8.85
49.60	8.19
55.12	7.55
60.63	6.92
64.14	6.48

Figure 2

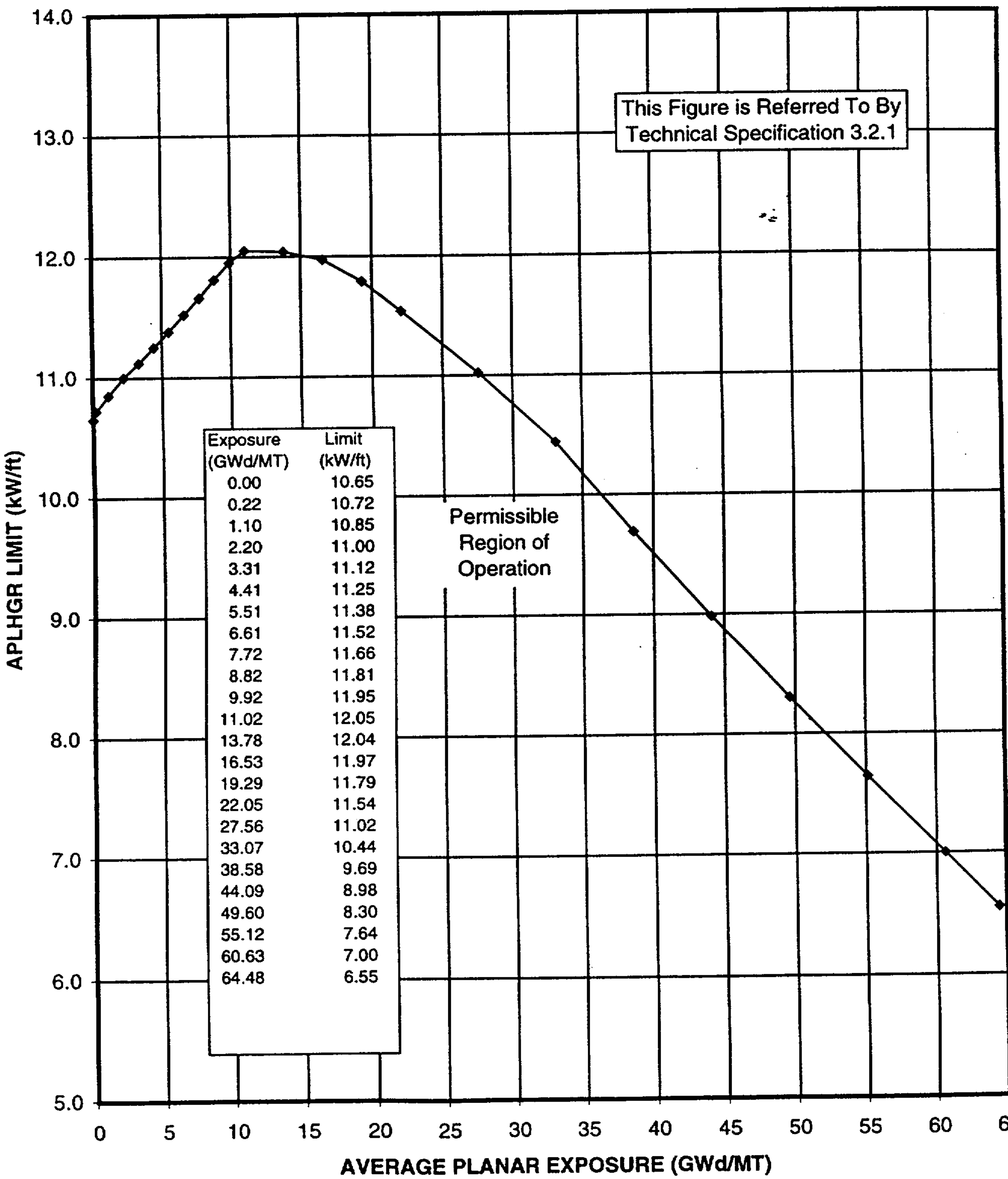
Fuel Type GE13-P9DTB395-12G5.0-100T-146-T (GE13) Average Planar Linear Heat Generation Rate (APLHGR) Limit Versus Average Planar Exposure



Exposure (GWd/MT)	Limit (kW/ft)
0.00	11.15
0.22	11.24
1.10	11.38
2.20	11.50
3.31	11.57
4.41	11.66
5.51	11.74
6.61	11.83
7.72	11.92
8.82	12.02
9.92	12.11
11.02	12.21
13.78	12.21
16.53	12.04
19.29	11.78
22.05	11.51
27.56	10.98
33.07	10.44
38.58	9.73
44.09	9.03
49.60	8.33
55.12	7.64
60.63	6.95
64.28	6.49

Figure 3

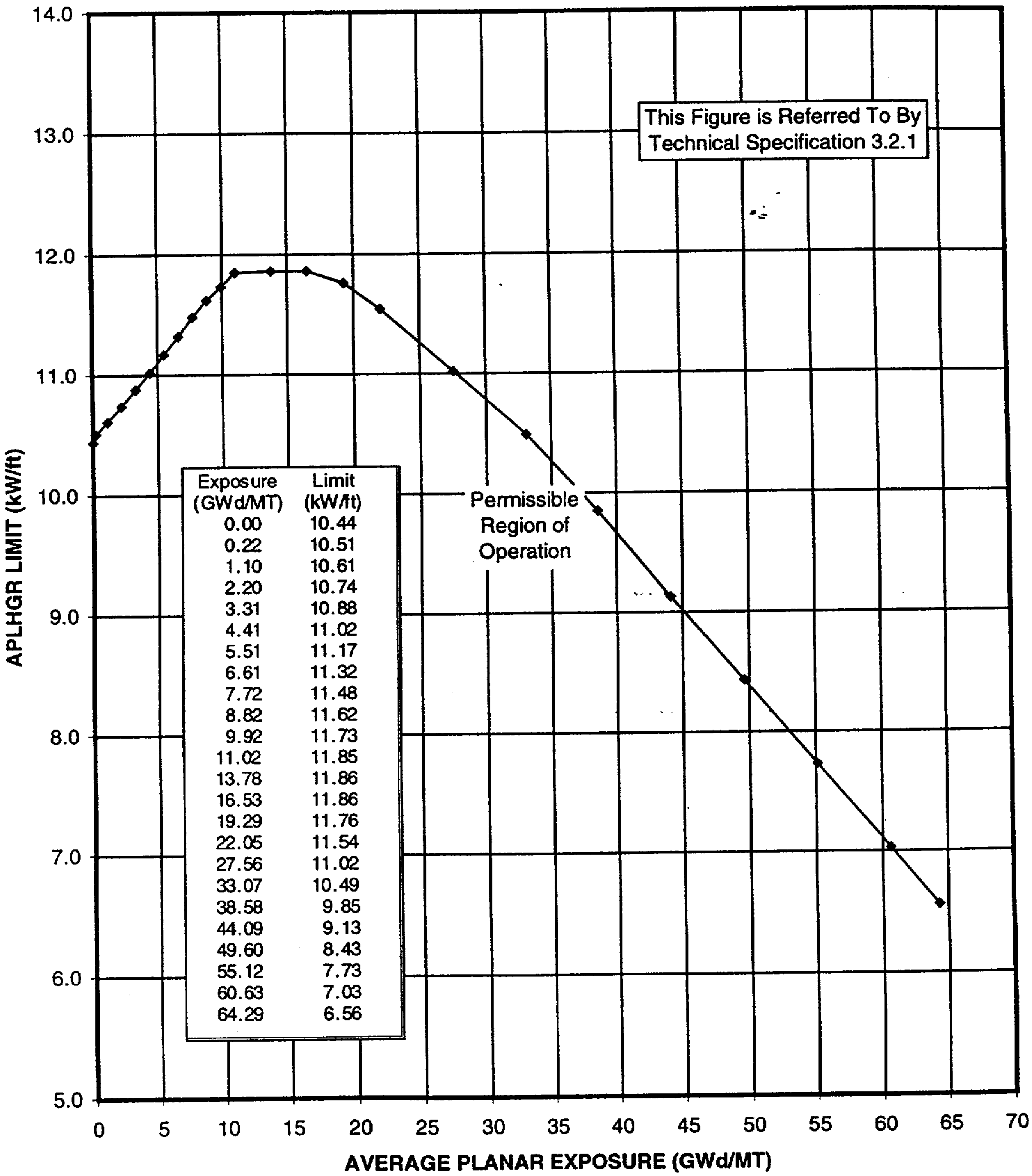
Fuel Type GE13-P9DTB403-5G6.0/7G5.0-100T-146-T (GE13) Average Planar Linear Heat Generation Rate (APLHGR) Limit Versus Average Planar Exposure



Exposure (GWd/MT)	Limit (kW/ft)
0.00	10.65
0.22	10.72
1.10	10.85
2.20	11.00
3.31	11.12
4.41	11.25
5.51	11.38
6.61	11.52
7.72	11.66
8.82	11.81
9.92	11.95
11.02	12.05
13.78	12.04
16.53	11.97
19.29	11.79
22.05	11.54
27.56	11.02
33.07	10.44
38.58	9.69
44.09	8.98
49.60	8.30
55.12	7.64
60.63	7.00
64.48	6.55

Figure 4

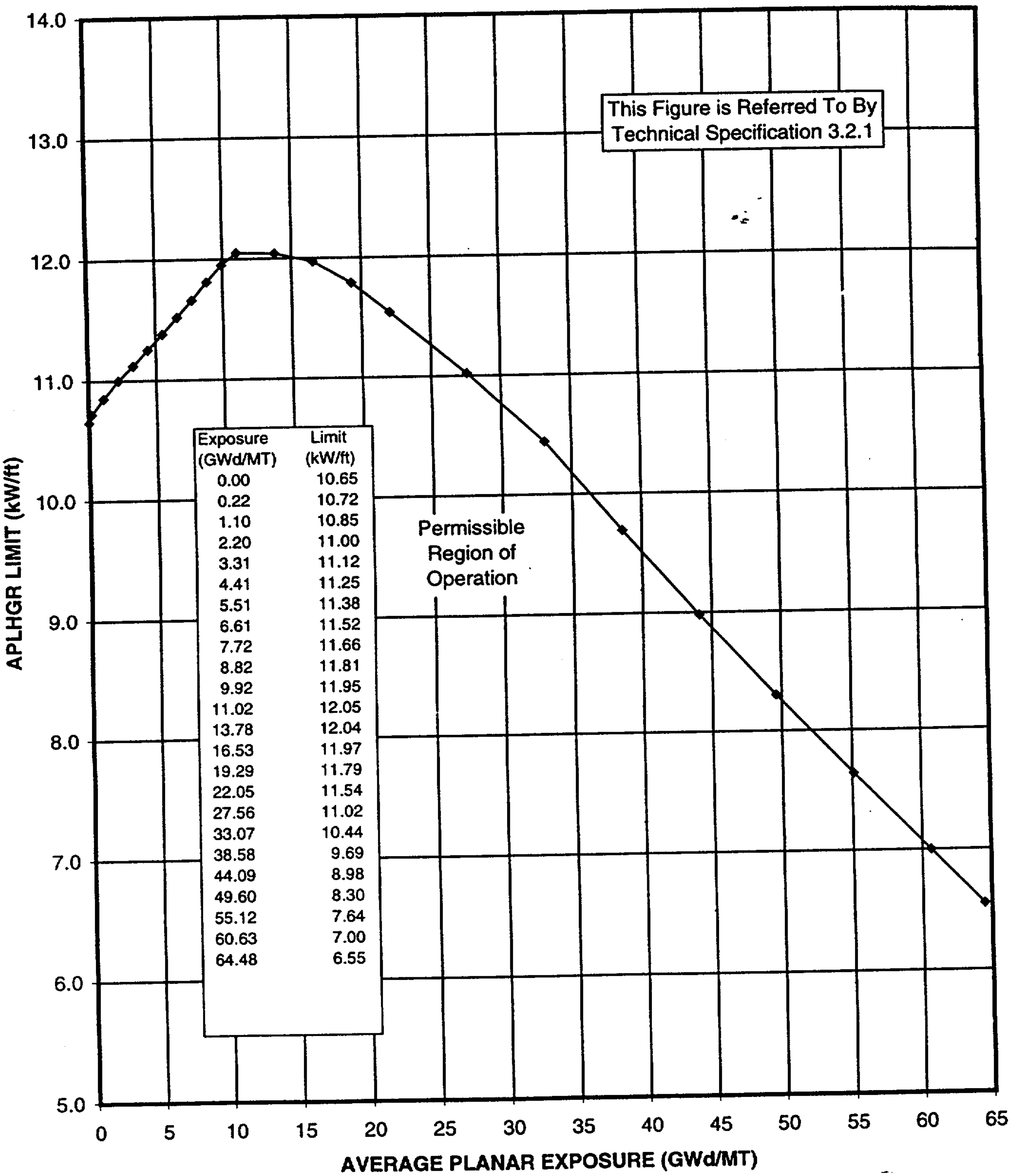
**Fuel Type GE13-P9DTB403-7G6.0/7G5.0-100T-146-T (GE13)
Average Planar Linear Heat Generation Rate (APLHGR) Limit
Versus Average Planar Exposure**



Exposure (GWd/MT)	Limit (kW/ft)
0.00	10.44
0.22	10.51
1.10	10.61
2.20	10.74
3.31	10.88
4.41	11.02
5.51	11.17
6.61	11.32
7.72	11.48
8.82	11.62
9.92	11.73
11.02	11.85
13.78	11.86
16.53	11.86
19.29	11.76
22.05	11.54
27.56	11.02
33.07	10.49
38.58	9.85
44.09	9.13
49.60	8.43
55.12	7.73
60.63	7.03
64.29	6.56

Figure 5

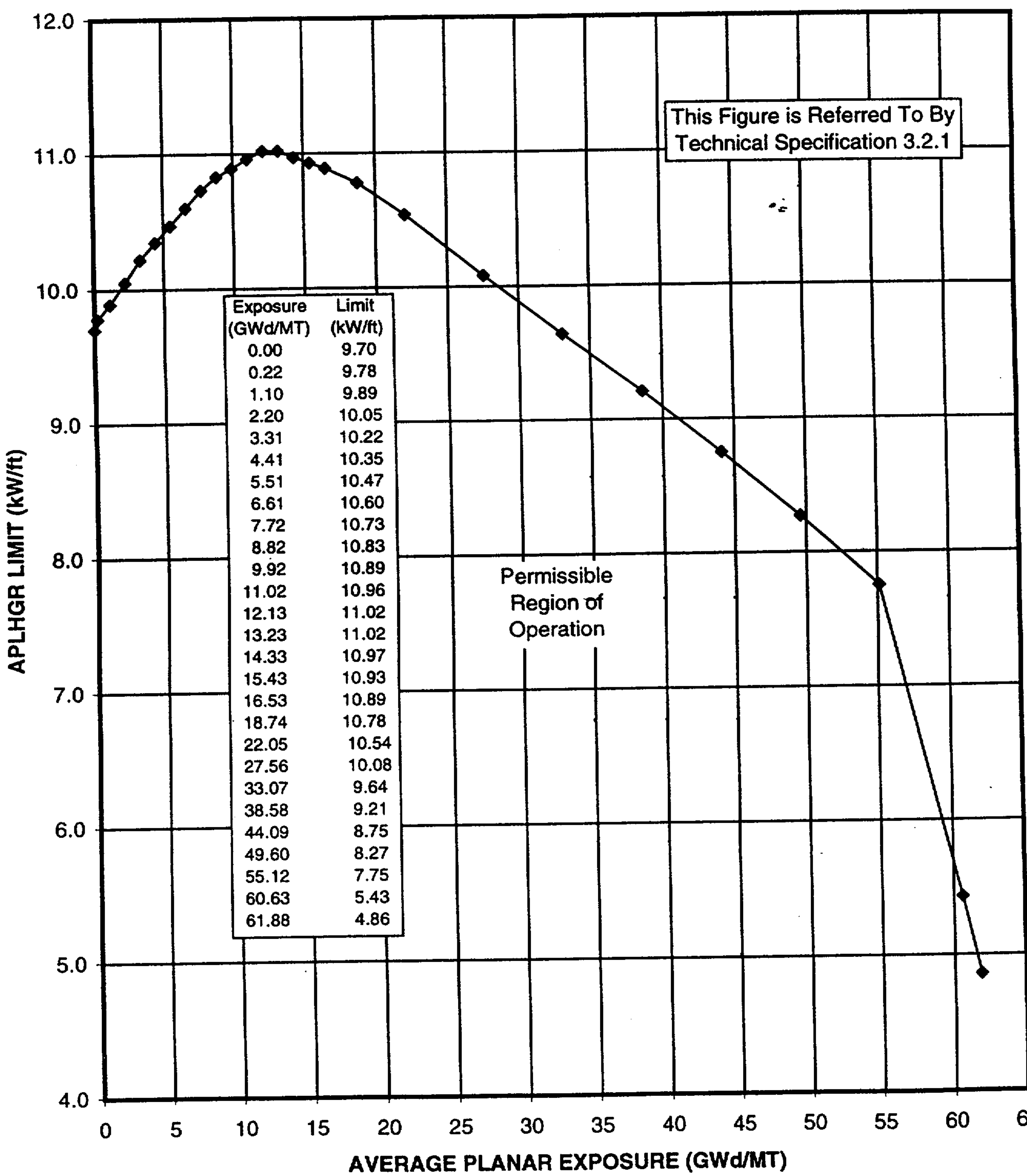
Fuel Type Atrium-10
Average Planar Linear Heat Generation Rate (APLHGR) Limit Versus Average Planar Exposure



Exposure (GWd/MT)	Limit (kW/ft)
0.00	10.65
0.22	10.72
1.10	10.85
2.20	11.00
3.31	11.12
4.41	11.25
5.51	11.38
6.61	11.52
7.72	11.66
8.82	11.81
9.92	11.95
11.02	12.05
13.78	12.04
16.53	11.97
19.29	11.79
22.05	11.54
27.56	11.02
33.07	10.44
38.58	9.69
44.09	8.98
49.60	8.30
55.12	7.64
60.63	7.00
64.48	6.55

Figure 6

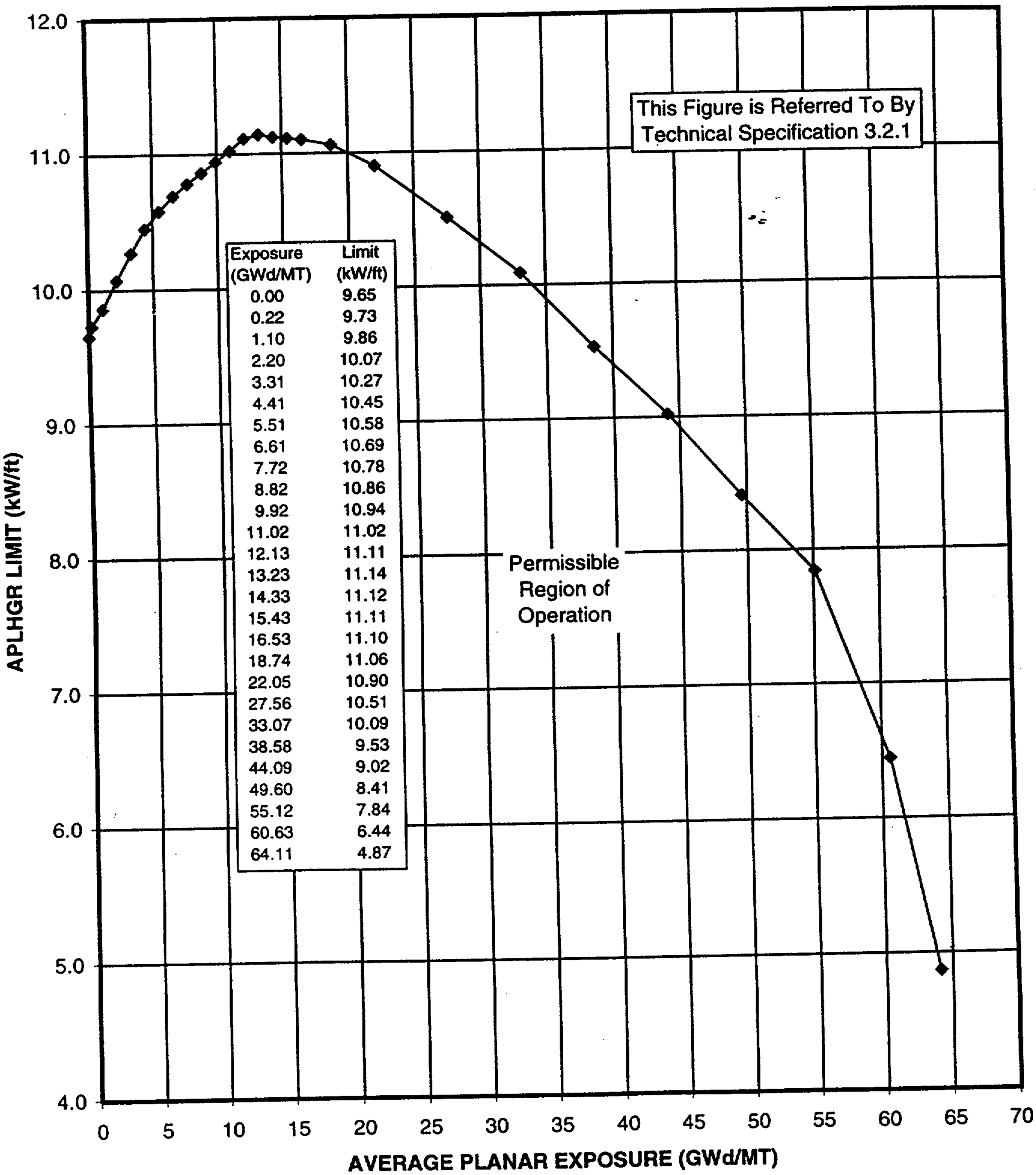
Fuel Type GE14-P10DNAB398-13GZ-100T-150-T (GE14) Average Planar Linear Heat Generation Rate (APLHGR) Limit Versus Average Planar Exposure



Exposure (GWd/MT)	Limit (kW/ft)
0.00	9.70
0.22	9.78
1.10	9.89
2.20	10.05
3.31	10.22
4.41	10.35
5.51	10.47
6.61	10.60
7.72	10.73
8.82	10.83
9.92	10.89
11.02	10.96
12.13	11.02
13.23	11.02
14.33	10.97
15.43	10.93
16.53	10.89
18.74	10.78
22.05	10.54
27.56	10.08
33.07	9.64
38.58	9.21
44.09	8.75
49.60	8.27
55.12	7.75
60.63	5.43
61.88	4.86

Figure 7

Fuel Type GE14-P10DNAB399-16GZ-100T-150-T (GE14)
Average Planar Linear Heat Generation Rate (APLHGR) Limit
Versus Average Planar Exposure



Exposure (GWd/MT)	Limit (kW/ft)
0.00	9.65
0.22	9.73
1.10	9.86
2.20	10.07
3.31	10.27
4.41	10.45
5.51	10.58
6.61	10.69
7.72	10.78
8.82	10.86
9.92	10.94
11.02	11.02
12.13	11.11
13.23	11.14
14.33	11.12
15.43	11.11
16.53	11.10
18.74	11.06
22.05	10.90
27.56	10.51
33.07	10.09
38.58	9.53
44.09	9.02
49.60	8.41
55.12	7.84
60.63	6.44
64.11	4.87

Figure 8

Not Used

Figure 9

Flow-Dependent MAPLHGR Limit, MAPLHGR(F)

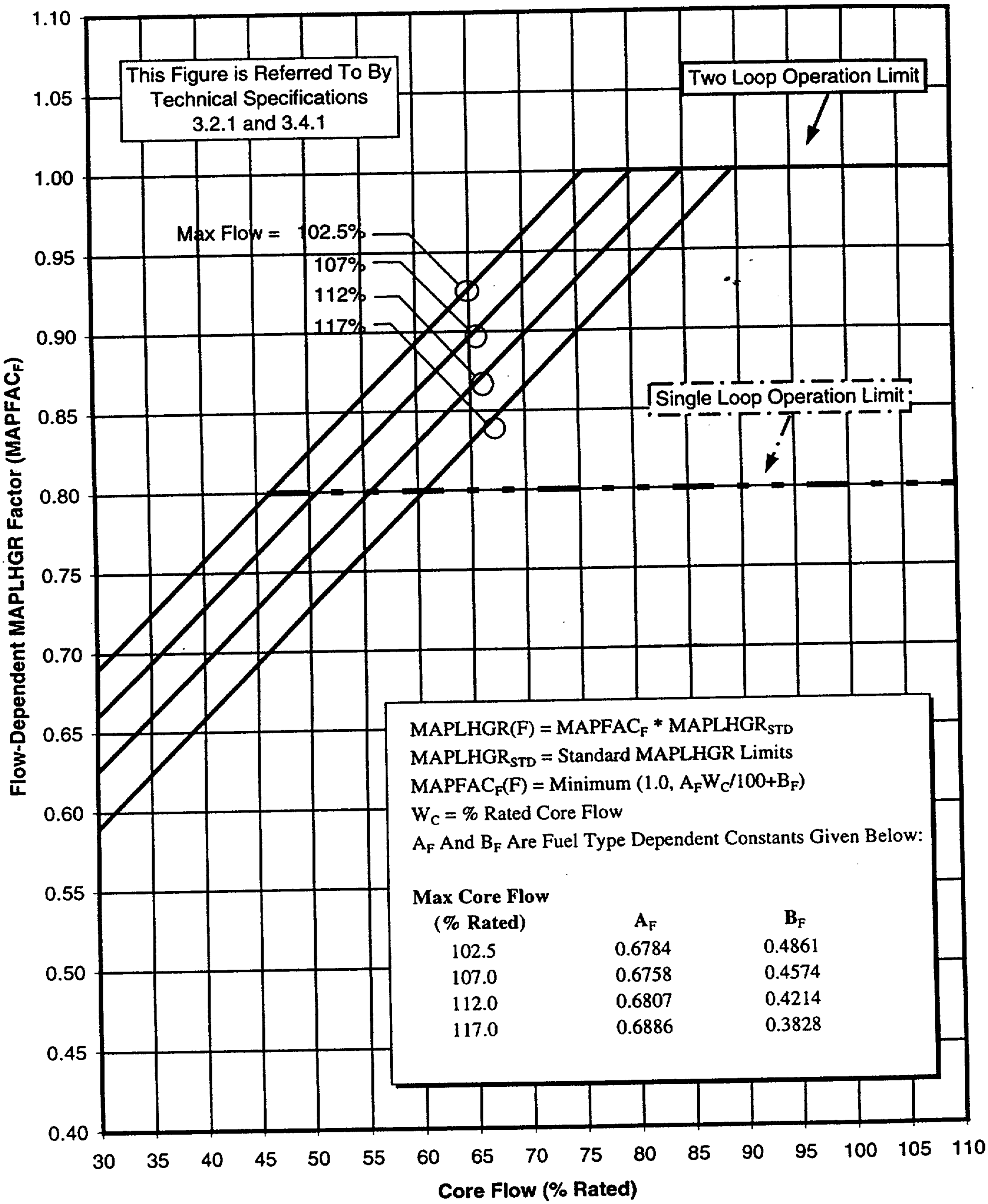


Figure 10

GE13 and A10 Power-Dependent MAPLHGR Limit, MAPLHGR (P)

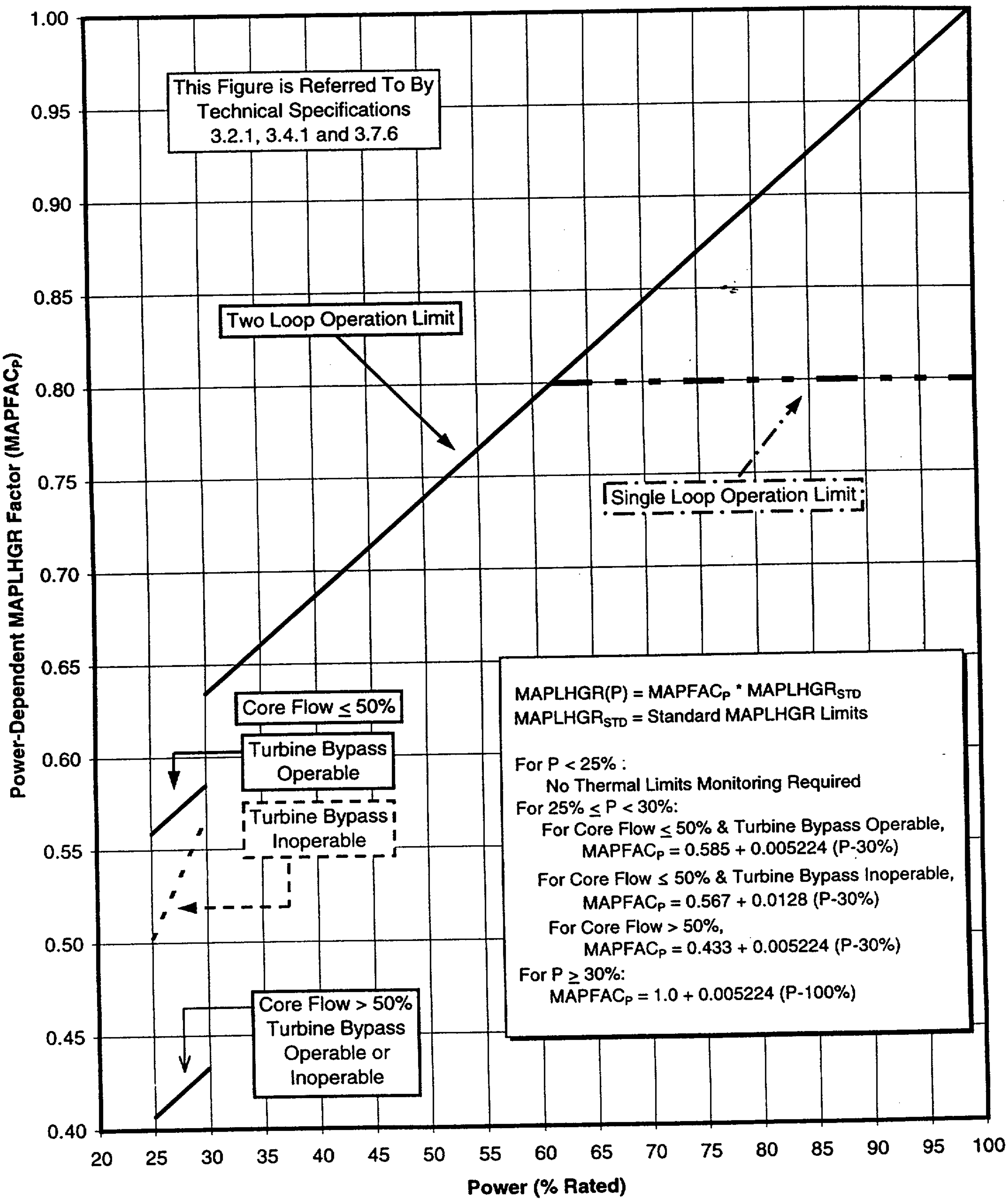


Figure 10a

GE14 Power-Dependent MAPLHGR Limit, MAPLHGR (P)

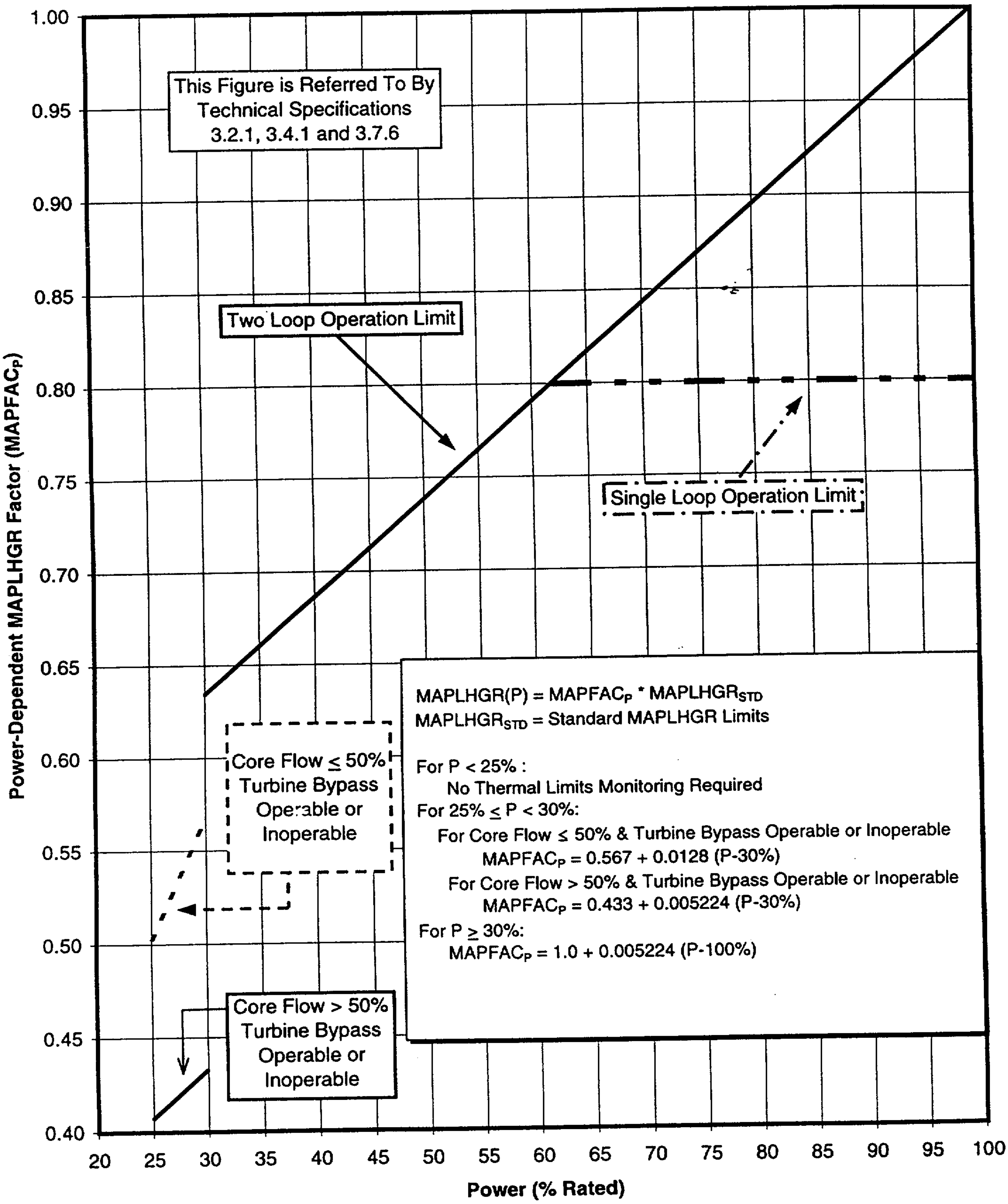


Figure 11

GE13 and GE14 Flow-Dependent MCPR Limit, MCPR(F)

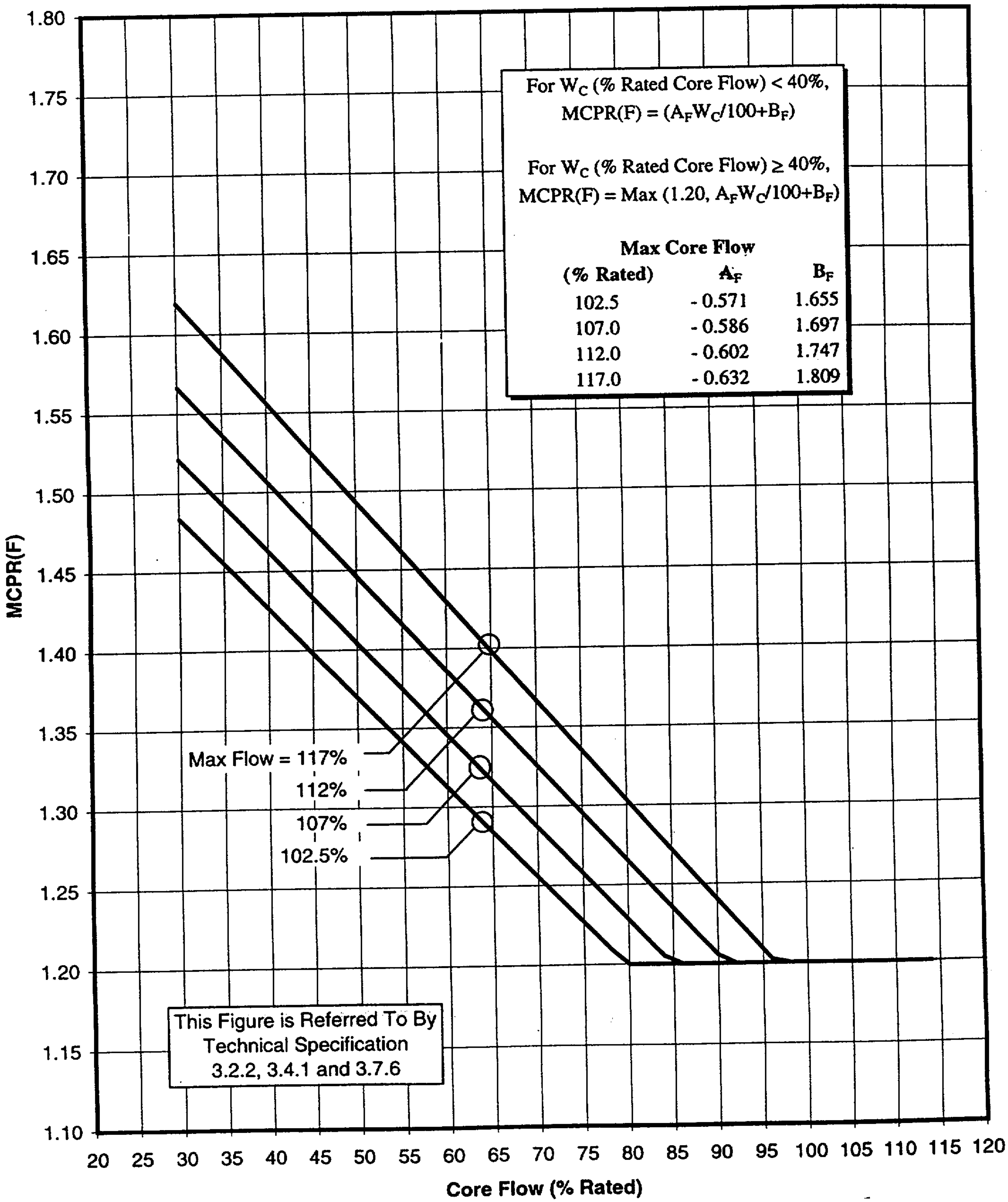


Figure 11a

A10 Flow-Dependent MCPR Limit, MCPR(F)

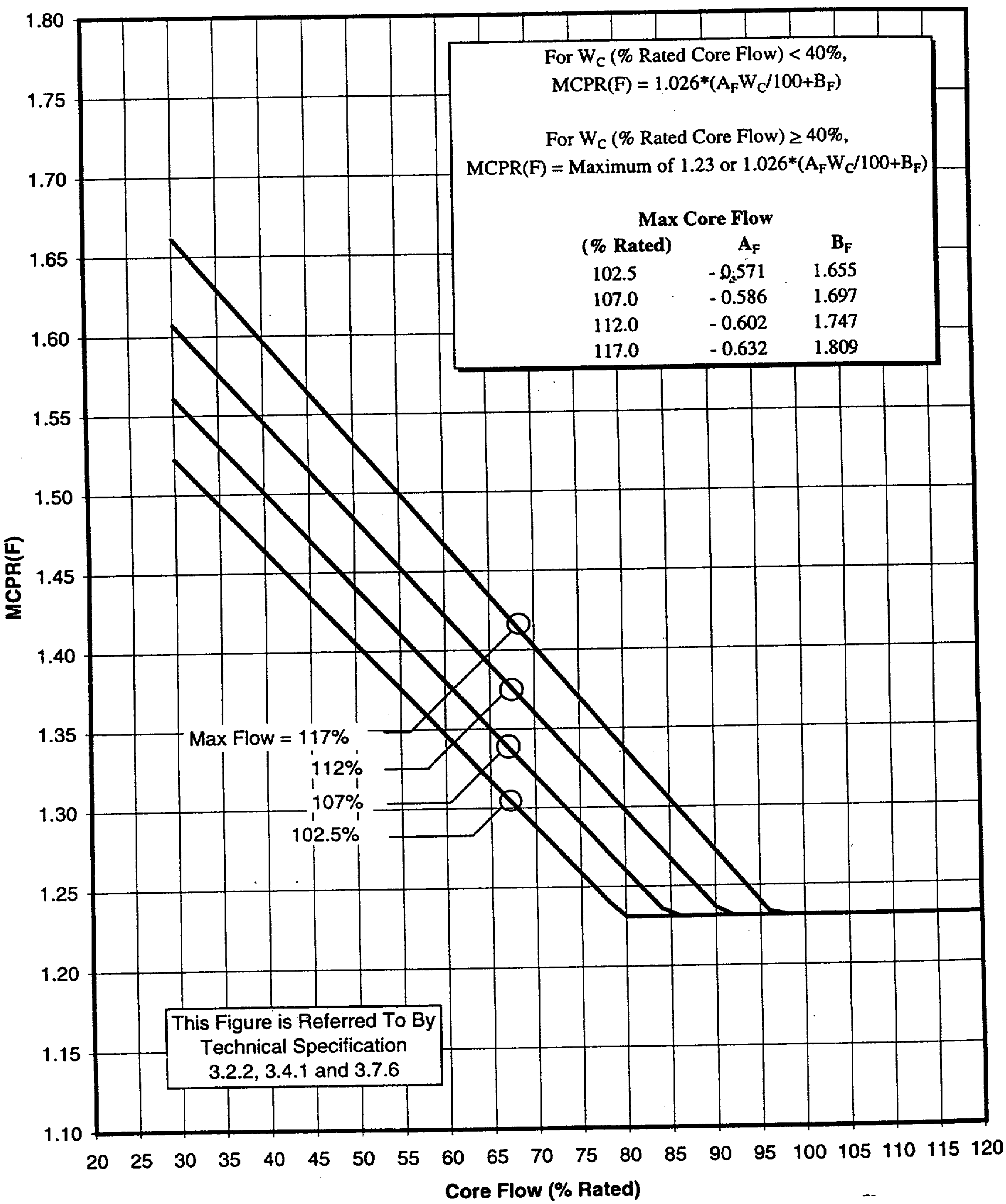


Figure 12

GE13 and Atrium-10 Power - Dependent MCPR Limit, MCPR (P)

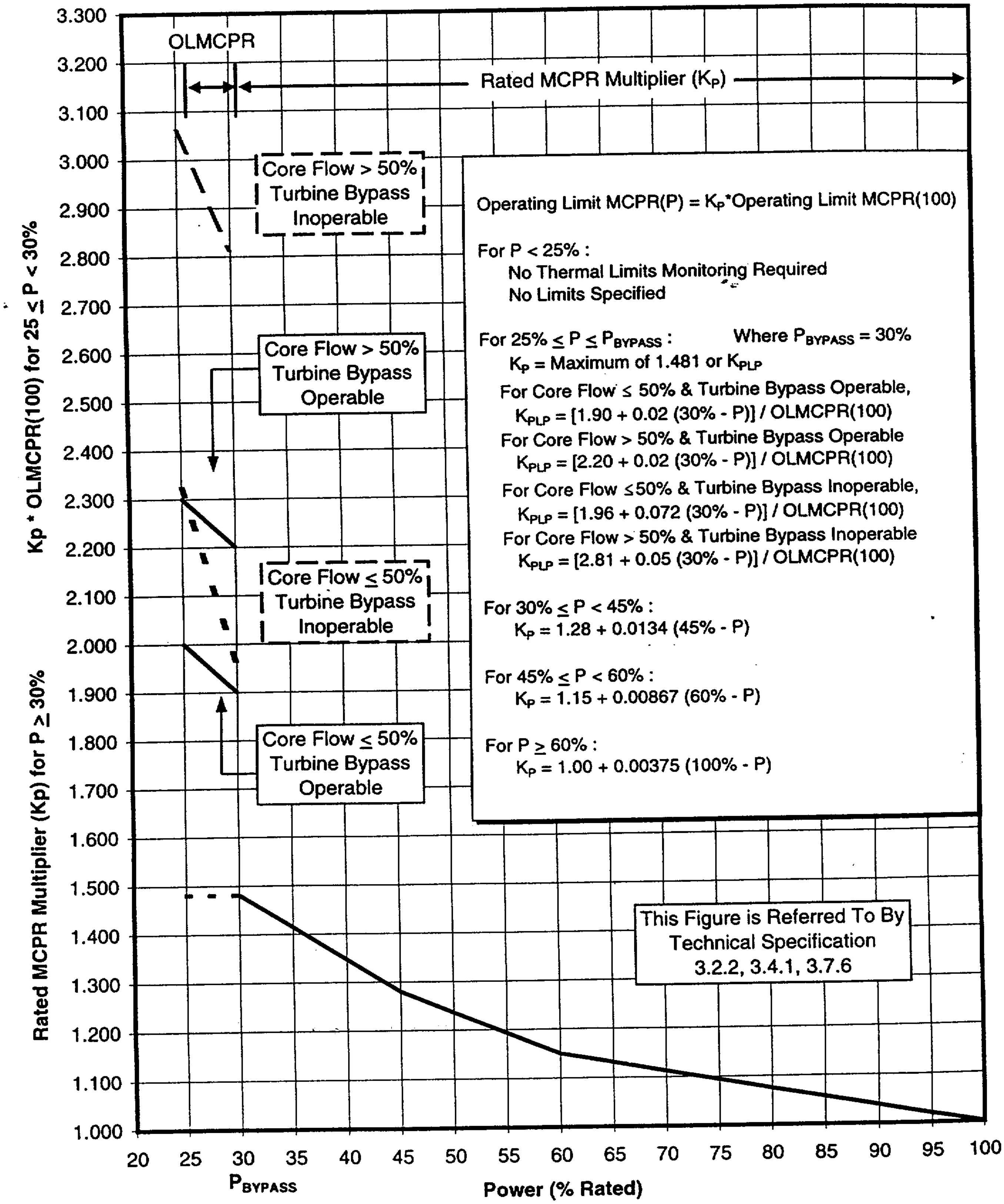


Figure 12a

GE14 Power - Dependent MCPR Limit, MCPR (P)

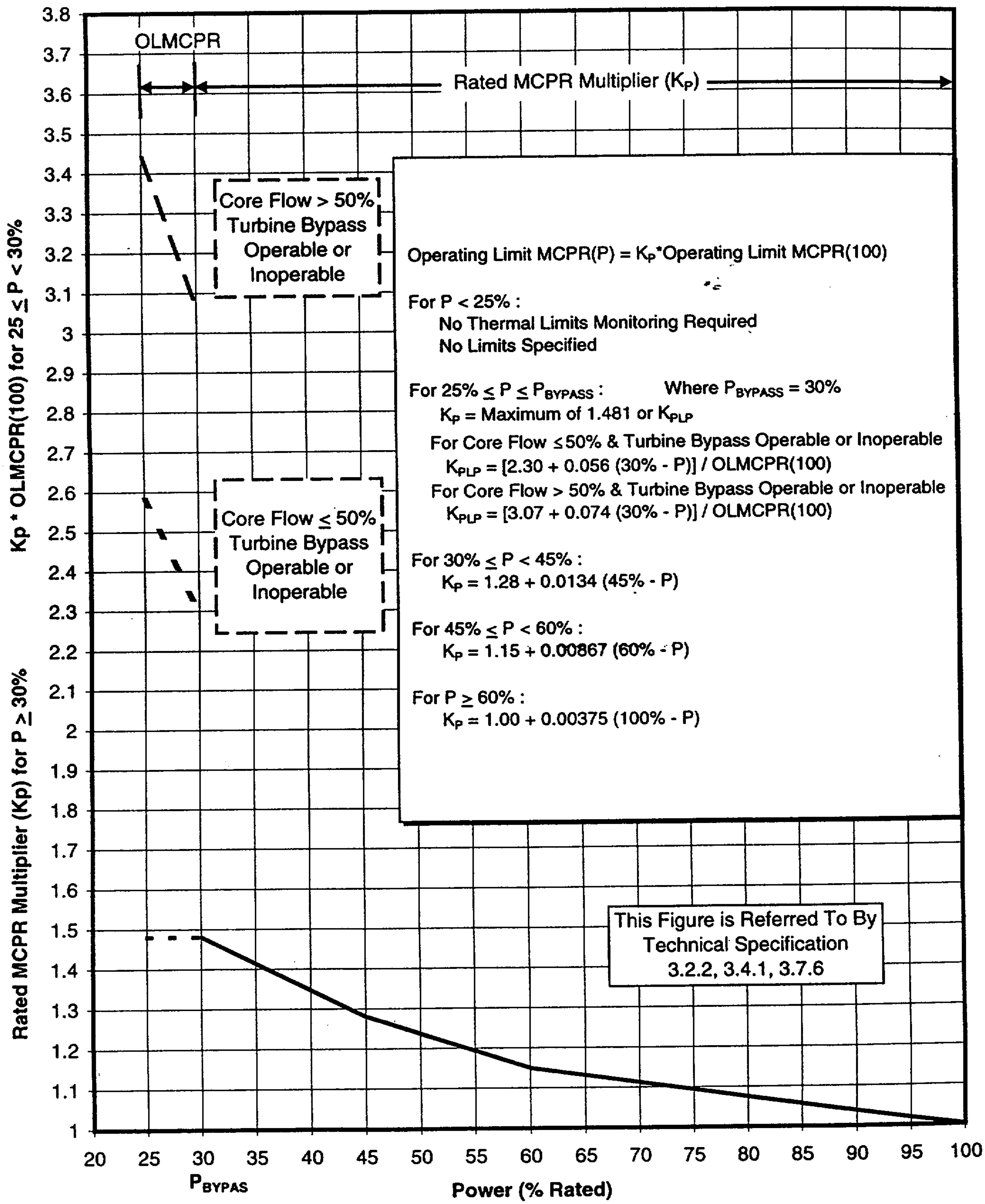
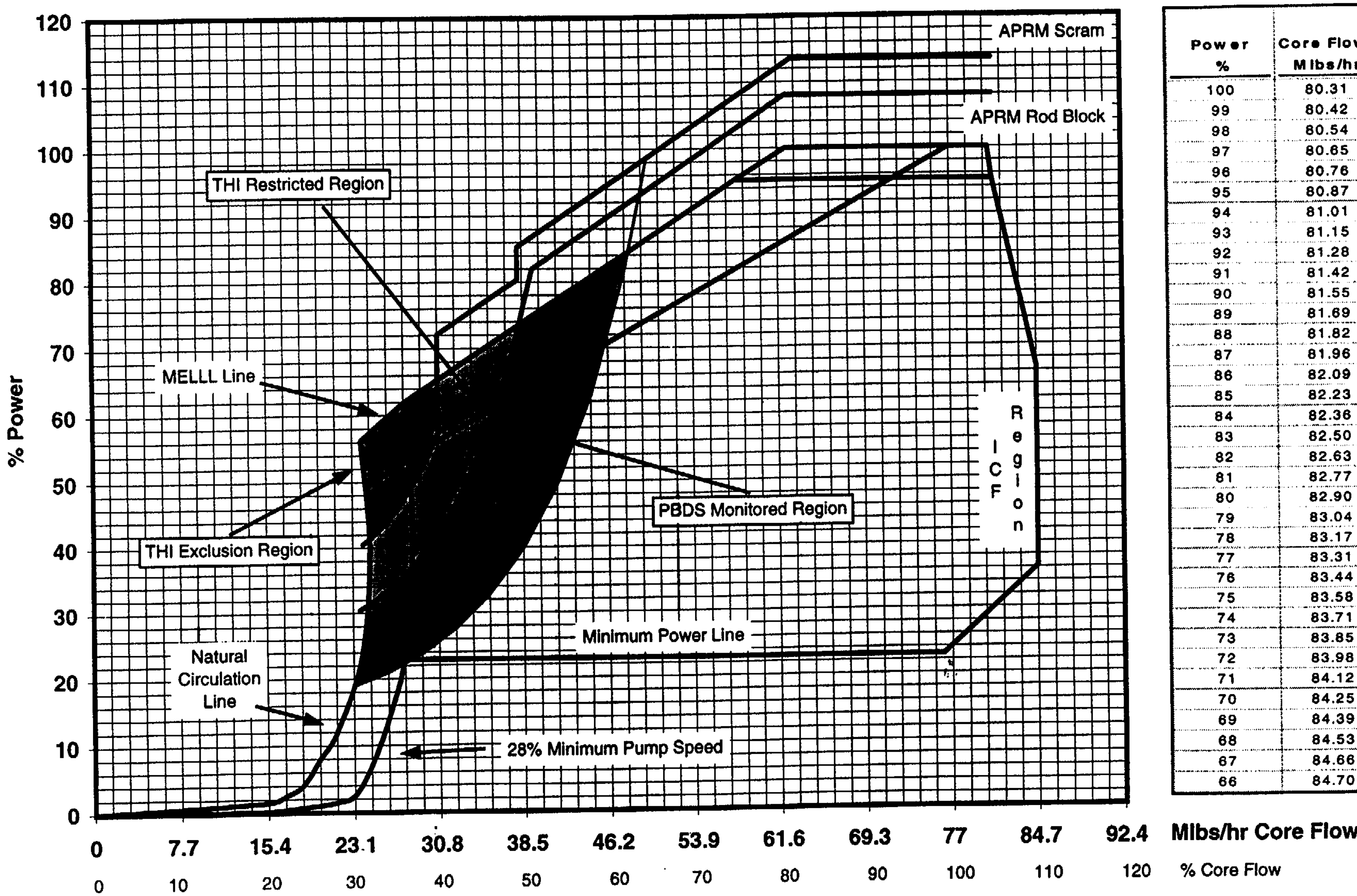


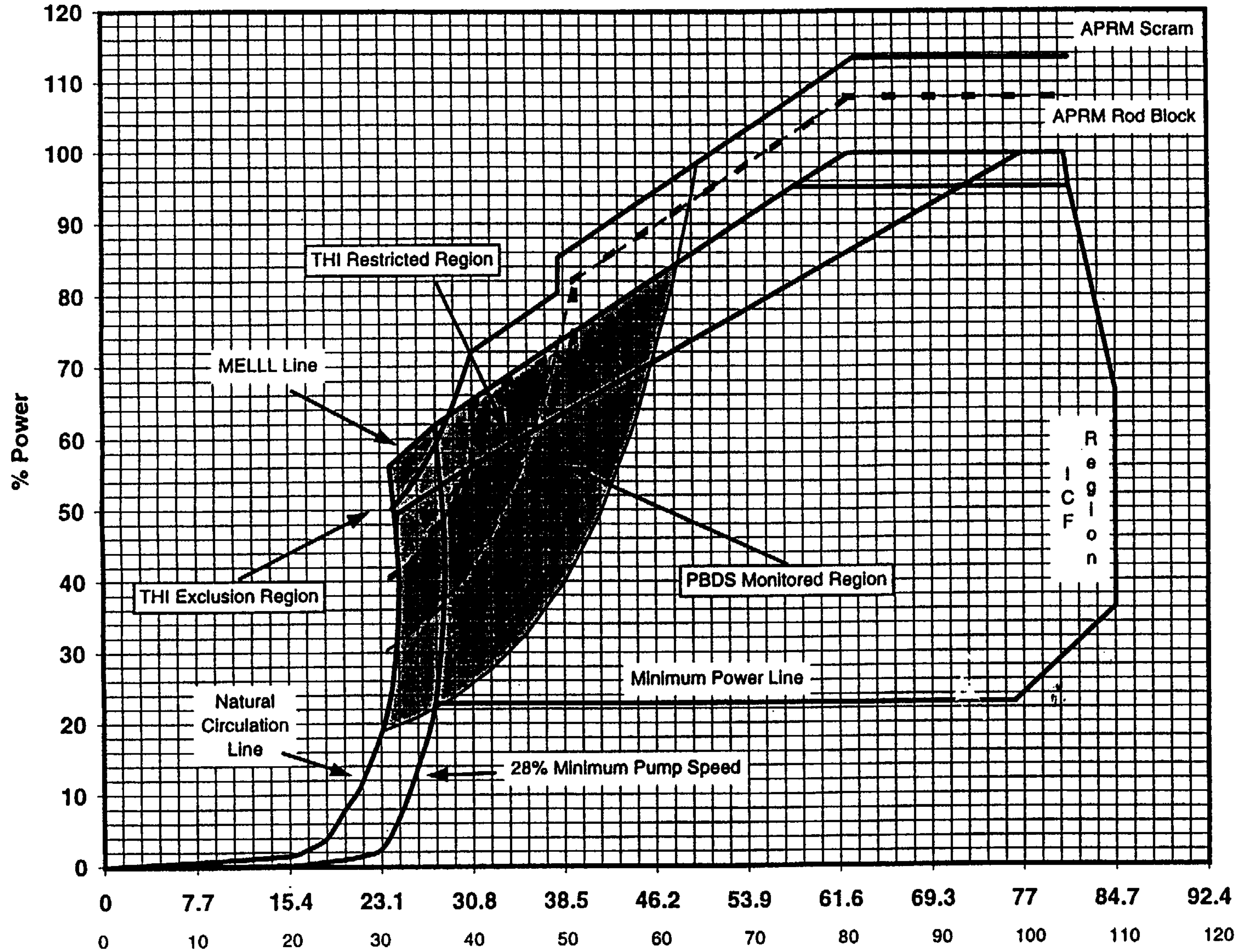
Figure 13
Power/Flow Map Stability Regions:
Normal TFW, Non-Setup



Power %	Core Flow, Mlbs/hr
100	80.31
99	80.42
98	80.54
97	80.65
96	80.76
95	80.87
94	81.01
93	81.15
92	81.28
91	81.42
90	81.55
89	81.69
88	81.82
87	81.96
86	82.09
85	82.23
84	82.36
83	82.50
82	82.63
81	82.77
80	82.90
79	83.04
78	83.17
77	83.31
76	83.44
75	83.58
74	83.71
73	83.85
72	83.98
71	84.12
70	84.25
69	84.39
68	84.53
67	84.66
66	84.70

This Figure supports Technical Specifications 3.2.3, 3.3.1.1 and 3.3.1.3 and the Technical Requirements Manual Specifications 3.2 and 3.3

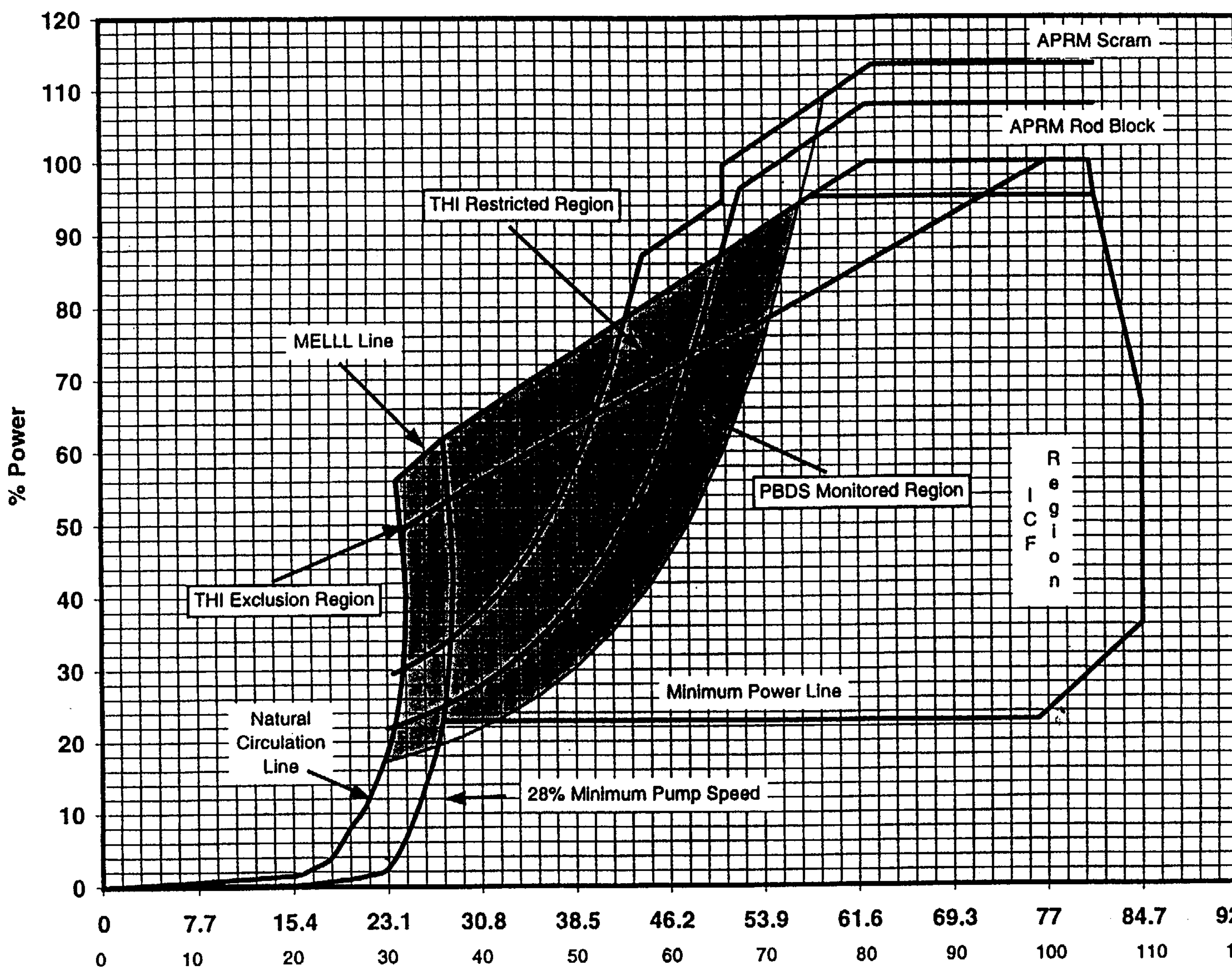
Figure 14
Power/Flow Map Stability Regions:
Normal TFW, Setup



Power %	Core Flow, Mlbs/hr
100	80.31
99	80.42
98	80.54
97	80.65
96	80.76
95	80.87
94	81.01
93	81.15
92	81.28
91	81.42
90	81.55
89	81.69
88	81.82
87	81.96
86	82.09
85	82.23
84	82.36
83	82.50
82	82.63
81	82.77
80	82.90
79	83.04
78	83.17
77	83.31
76	83.44
75	83.58
74	83.71
73	83.85
72	83.98
71	84.12
70	84.25
69	84.39
68	84.53
67	84.66
66	84.70

This Figure supports Technical Specifications 3.2.3, 3.3.1.1 and 3.3.1.3 and the Technical Requirements Manual Specifications 3.2 and 3.3

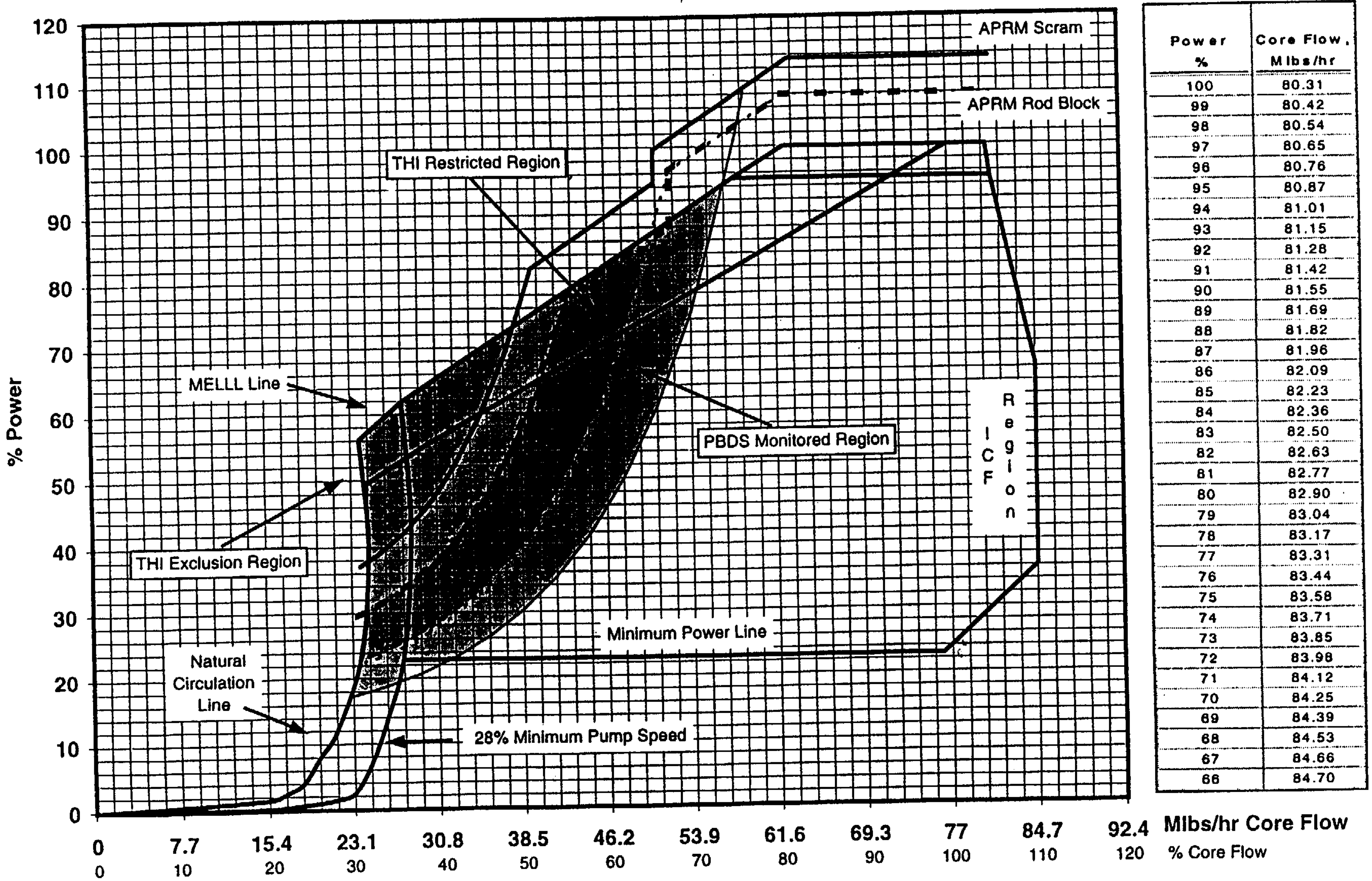
Figure 15
Power/Flow Map Stability Regions: Reduced TFW, Non-Setup



Power %	Core Flow, Mlbs/hr
100	80.31
99	80.42
98	80.54
97	80.65
96	80.76
95	80.87
94	81.01
93	81.15
92	81.28
91	81.42
90	81.55
89	81.69
88	81.82
87	81.96
86	82.09
85	82.23
84	82.36
83	82.50
82	82.63
81	82.77
80	82.90
79	83.04
78	83.17
77	83.31
76	83.44
75	83.58
74	83.71
73	83.85
72	83.98
71	84.12
70	84.25
69	84.39
68	84.53
67	84.66
66	84.70

This Figure supports Technical Specifications 3.2.3, 3.3.1.1 and 3.3.1.3 and the Technical Requirements Manual Specifications 3.2 and 3.3

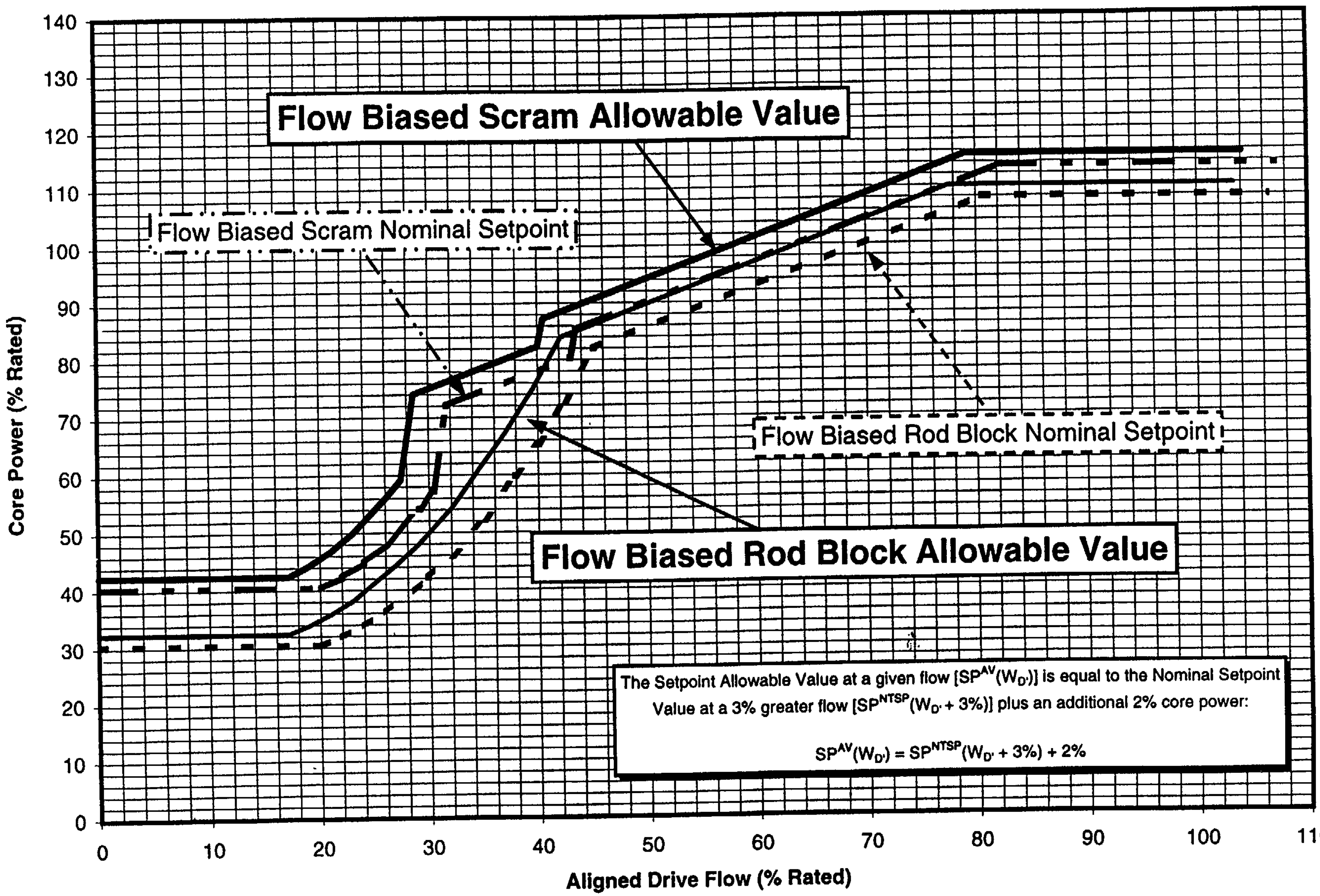
Figure 16
Power/Flow Map Stability Regions: Reduced TFW, Setup



Power %	Core Flow, Mlbs/hr
100	80.31
99	80.42
98	80.54
97	80.65
96	80.76
95	80.87
94	81.01
93	81.15
92	81.28
91	81.42
90	81.55
89	81.69
88	81.82
87	81.96
86	82.09
85	82.23
84	82.36
83	82.50
82	82.63
81	82.77
80	82.90
79	83.04
78	83.17
77	83.31
76	83.44
75	83.58
74	83.71
73	83.85
72	83.98
71	84.12
70	84.25
69	84.39
68	84.53
67	84.66
66	84.70

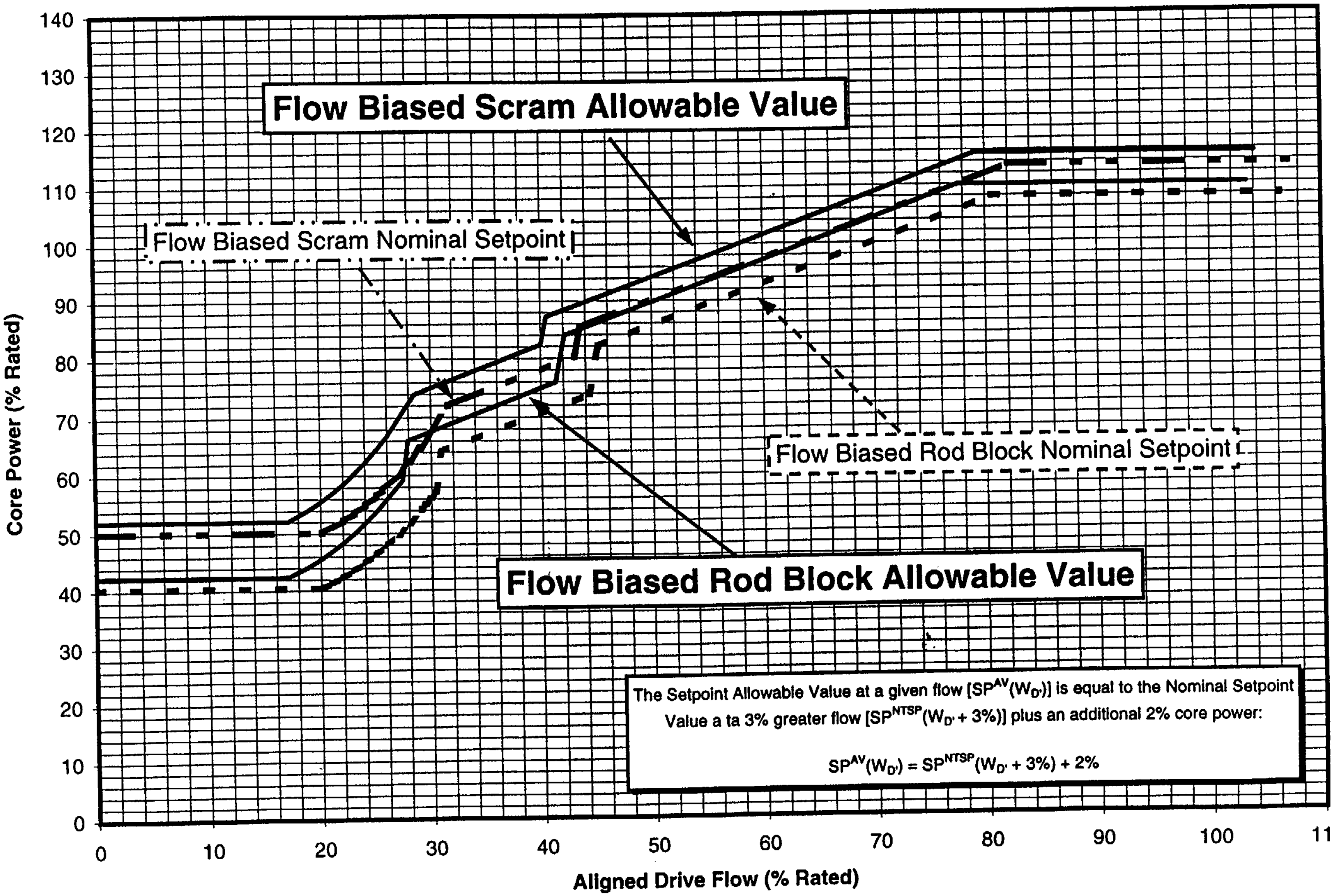
This Figure supports Technical Specifications 3.2.3, 3.3.1.1 and 3.3.1.3 and the Technical Requirements Manual Specifications 3.2 and 3.3

Figure 17
E1A Setpoint Allowable Values versus Aligned Drive Flow: Normal TFW, Non-Setup



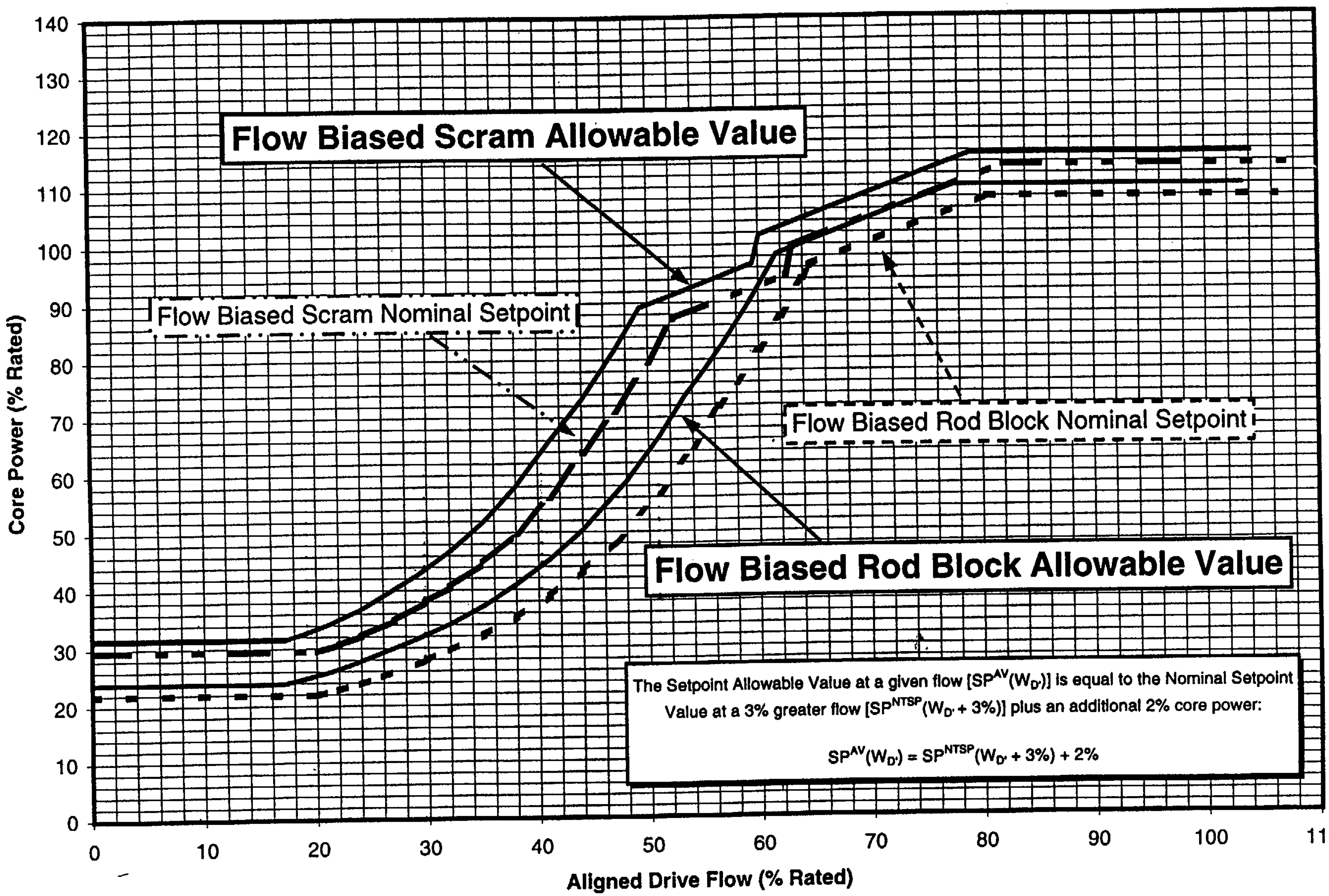
This Figure supports Technical Specification 3.3.1.1 and the Technical Requirements Manual Specifications 3.2 and 3.3

Figure 18
E1A Setpoint Allowable Values versus Aligned Drive Flow: Normal TFW, Setup



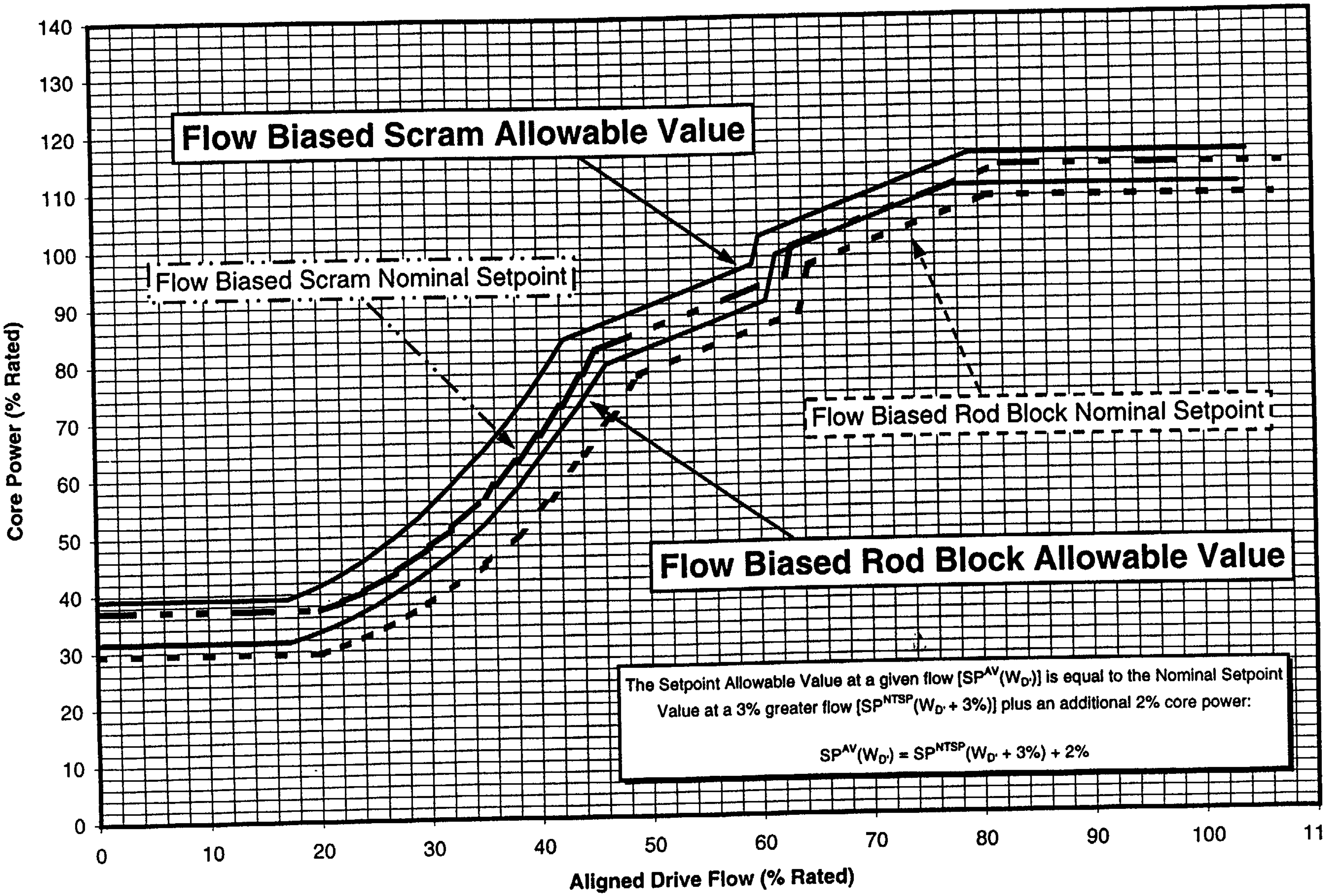
This Figure supports Technical Specification 3.3.1.1 and the Technical Requirements Manual Specifications 3.2 and 3.3

Figure 19
E1A Setpoint Allowable Values versus Aligned Drive Flow: Reduced TFW, Non-Setup



This Figure supports Technical Specification 3.3.1.1 and the Technical Requirements Manual Specifications 3.2 and 3.3

Figure 20
E1A Setpoint Allowable Values versus Aligned Drive Flow: Reduced TFW, Setup



This Figure supports Technical Specification 3.3.1.1 and the Technical Requirements Manual Specifications 3.2 and 3.3

PLANT OPERATING MANUAL

VOLUME XIII

PLANT EMERGENCY PROCEDURE

UNIT
0

0PEP-02.1

INITIAL EMERGENCY ACTIONS

REVISION 47

TABLE OF CONTENTS

1.0 PURPOSE

This procedure should be implemented by the Shift Superintendent or his alternate as described in Step 5.0 upon recognition of an off-normal condition to assist in determining whether an event should be classified as an emergency.

2.0 REFERENCES

2.1 0PEP-03.8.2, Personnel Accountability and Evacuation

2.2 0PEP-03.9.2, First Aid and Medical Care

2.3 0PEP-03.9.3, Transport of Contaminated Injured Personnel

2.4 0PEP-03.9.6, Search and Rescue

2.5 0PEP-03.1.3, Use of Communication Equipment

2.6 0PEP-02.1.1 Emergency Control - Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency

2.7 0RCI-06.1, Reportable Event Evaluation Criteria and Processing

2.8 0OI-01.07, Notifications

2.9 BSEP Technical Specifications

2.10 0E&RC-2020, Setpoint Determinations For Gaseous Radiation Monitors

2.11 0PFP-013, General Fire Plan

2.12 0PEP-03.4.7, Automation of Off-Site Dose Projection Procedures

2.13 BSEP Off-Site Dose Calculation Manual (ODCM)

2.14 NEI 97-03, Methodology for Development of Emergency Action Levels

2.15 NUREG-1022, Revision 1, Event Reporting Guidelines: 10 CFR50.72 and 50.73

3.0 GENERAL

3.1 This procedure should be implemented upon the identification of any off-normal condition.

3.2 Implementation of this procedure does not constitute an emergency but rather serves as a guideline for evaluation of plant conditions and comparisons with Emergency Action Levels (EALs).

3.3 Once implemented, this procedure shall remain in effect until:

3.3.1 All EAL criteria are determined to be less than event classification threshold values;

AND

3.3.2 The off-normal conditions have been resolved.

4.0 DEFINITIONS/ABBREVIATIONS

4.1 SEC - Site Emergency Coordinator

4.2 SRO - Senior Reactor Operator

4.3 Adequate core cooling - Heat removal from the reactor sufficient to prevent rupturing the fuel clad. Three viable mechanisms of adequate core cooling exist; in order of preference they are:

- Core submergence
- Steam cooling with injection of makeup water to the reactor
- Steam cooling without injection of makeup water to the reactor

4.4 Primary Containment Operability

4.4.1 All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an operable automatic containment isolation system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in technical specifications;

4.0 DEFINITIONS/ABBREVIATIONS

4.4.2 The primary containment air lock is operable, except as provided in technical specifications;

4.4.3 All equipment hatches are closed; and

4.4.4 The sealing mechanism associated with a penetration (e.g., welds, bellows, or O-rings) is operable.

4.4.5 Containment leakage rates are within the limits of technical specifications.

4.5 FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is not required if large quantities of smoke and heat are observed.

4.6 TOXIC GASES - A gas that is dangerous to life or health by reason of inhalation or skin contact (e.g. chlorine). Asphyxiants can also become toxic in large enough quantities (e.g. CO^2).

5.0 RESPONSIBILITIES

5.1 The Shift Superintendent or alternate has immediate and unilateral authority to carry out this procedure. He may delegate specific steps as necessary, but shall not delegate the responsibility for classification of an event.

5.2 A Senior Reactor Operator is a qualified alternate to implement this procedure if the Shift Superintendent is not available.

> **NOTE:** Attachment 2 at the end of this procedure provides a flowchart that addresses the SEC actions once an event has been declared.

6.0 INSTRUCTIONS

> **NOTE:** There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Normal reporting requirements (e.g., 10 CFR 50.72) are applicable in these cases. (ref. NEI 97-03).
>
> **NOTE:** "*" denotes decisions or actions which should be entered in the Shift SRO Log.

6.0 INSTRUCTIONS

> **NOTE:** The following actions are to be carried out in an expeditious manner for personnel and plant protection and emergency classification.

6.1 Ensure appropriate Emergency Operating Procedures and plant procedures are implemented concurrently.

6.2 If conditions require building or localized plant area evacuation:

* 6.2.1 Sound Building Evacuation alarm for 15 seconds and announce over the Plant PA System "(state emergency condition) in the (location). Evacuate the (location)."

> **EXAMPLE:** "Attention all personnel, there is a Radiation Alarm in the Radwaste Building, Evacuate the Radwaste Building."

6.2.2 Implement 0PEP-03.8.2, Personnel Accountability and Evacuation (Building or Area Evacuation Section); direct affected personnel to report to their work group supervisor and direct work group supervisors to inform the Shift Superintendent of any personnel not accounted for within 30 minutes.

6.2.3 Repeat the PA announcement.

6.3 If personnel injuries have occurred:

6.3.1 Notify the Fire Brigade.

* 6.3.2 Determine number of persons injured and their location(s).

6.3.3 Implement 0PEP-03.9.2, First Aid and Medical Care; 0PEP-03.9.3, Transport of Contaminated Injured Personnel; or 0PEP-03.9.6, Search and Rescue as appropriate.

6.3.4 Determine whether injuries involve radioactive contamination.

6.0 INSTRUCTIONS

> **CAUTION**
>
> Priority should be placed on lifesaving injury treatment over the need to decontaminate. See 0PEP-03.9.2 for guidance.

6.4 If a fire has been reported:

6.4.1 Sound the fire alarm.

6.4.2 Notify the Fire Brigade.

6.4.3 Make the following PA announcement:

"Fire in (location)"
"Fire in (location)"
"Fire in (location)"

"All personnel **NOT** involved in fire fighting or direct support activities are to evacuate the involved area immediately."

"Use of the PA is now restricted for emergency communications, except as directed by the Unit SCO for operational safety concerns."

"The Fire Brigade is to muster at (designated location)."

6.4.4 Implement 0PFP-013, General Fire Plan.

> **NOTE:** The revision dates, annotated in the top right corner of the EAL flowpaths, depict the date of the most recent change to the flowpath and the REP and 0PEP-02.1 revisions that were in effect at that time.

6.5 Using EAL flowpaths or Attachment 1, compare plant conditions (observed or indicated parameters and conditions) with the EALs and classify the emergency.

6.0 INSTRUCTIONS

6.5.1 The EAL flowpath can be entered at any point if the event is known. (Example: fuel handling accident.) This point should be noted to ensure that all other events are evaluated prior to exiting the flowpath.

If the event is not known, enter at Point A.

6.5.2 If no emergency action level threshold is exceeded go to Step 6.6.

6.5.3 If, at any time, an emergency classification is warranted, the Site Emergency Coordinator is to immediately declare the appropriate classification and carry out the associated actions in accordance with 0PEP-02.1.1, Emergency Control - Notification of Unusual Event, Alert, Site Area Emergency, General Emergency. (The highest level emergency classification for the conditions will be declared.)

6.6 Continue to monitor and evaluate plant conditions in accordance with previous steps until off-normal conditions are returned to normal.

6.7 Review RCI-06.1 and 0OI-01.07 to determine reporting requirements.

6.8 A turnover checklist may be used to ensure that all essential tasks are completed; however, such a checklist shall not be used to replace this procedure.

NOTE: When operations are restored to within normal operating parameters and safe in the judgment of the Shift Superintendent, terminate use of this procedure.

NOTE: Notify the Maintenance Rule Program Engineer of any Emergency Action Level entry due to equipment failure.

Emergency Action Levels

1.0 Abnormal Primary Leak Rate

1.1 Notification of Unusual Event

Reactor Coolant System total leakage greater than 25 gpm averaged over the previous 24-hour period using the sum of drywell equipment drain integrator (G16-FQ-K603) and drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 25 gpm within eight hours, or plant shutdown is not achieved within required time period.

Unidentified Reactor Coolant System leakage greater than 5 gpm averaged over the previous 24-hour period using the drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 5 gpm within eight hours, or plant shutdown is not achieved within required time period.

1.2 Alert

Small break LOCA with primary system leakage greater than 50 gpm. A LOCA is indicated by a significant loss of reactor inventory to the drywell resulting in increased drywell pressure, temperature, and/or sump pump usage indicated by:

- Low or falling Reactor Coolant System pressure with rising drywell pressure and temperature (C32-R608, CAC-PI-2685-1, CAC-TR-4426-1A, CAC-TR-4426-1B, CAC-TR-4426-2A and CAC-TR-4426-2B).

1.3 Site Area Emergency

- Loss of coolant accident requiring the initiation of Low Pressure Coolant Injection, Core Spray, or the Automatic Depressurization System, **AND REQUIRED FOR ADEQUATE CORE COOLING.**

OR

- Loss of two-out-of-three fission product barriers listed in Step 2.4.1 of this attachment.

1.0 Abnormal Primary Leak Rate (Cont'd)

1.4 General Emergency

- Site Area Emergency indicated above **AND** inability to provide makeup water to the Reactor Coolant System (i.e., failure of HPCI, Core Spray A and B, RHR Loops A and B, RCIC, condensate, and feedwater) as indicated by falling or low reactor vessel level with attempts to inject water not successful.

OR

- Loss of two-out-of-three fission product barriers listed in Step 2.4.1 of this attachment with a potential to lose the third barrier.

2.0 Steam Line Break or Safety/Relief Valve Failure

2.1 Notification of Unusual Event

2.1.1 Reactor Coolant System pressure ≥ 1250 psig.

OR

2.1.2 Inability to close an SRV with Reactor Coolant System pressure ≤ 900 psig.

2.2 Alert

Steam line break downstream of MSIVs or upstream of feedwater isolation valves as indicated by:

A. Reactor trip with:

1. Low RCS pressure (C32-R608 or B21-PI-R605A or B21-PI-R605B)

OR

2. Low steam pressure (C32-R609)

OR

3. Low reactor vessel water level (C32-R608)

OR

4. High steam flow (C32-R603)

AND

B. Shift Superintendent/Site Emergency Coordinator's opinion or evidence on P601 and P603 of continuing steam flow with steam line break outside of primary containment.

2.0 Steam Line Break or Safety/Relief Valve Failure (Cont'd)

2.3 Site Area Emergency

- Alert indicated above and inability to isolate the leak.

OR

- Loss of two-out-of-three fission product barriers listed in Step 2.4.1 of this attachment.

2.4 General Emergency

2.4.1 Loss of any two of the three fission product barriers below with a potential loss of the third barrier:

A. Failed fuel causing RCS activity greater than 40 μCi/ml I-131 dose equivalent

B. Loss of primary coolant boundary

1. Loss of coolant accident (Step 1.2 of this Attachment - Alert)

2. Major steam line break (Step 2.2 of this Attachment - Alert)

C. Loss of primary containment operability. A release path has been established.

3.0 Abnormal Core Conditions and Core Damage

3.1 Notification of Unusual Event

Failed fuel as indicated by:

3.1.1 Liquid

A. Reactor Coolant System (RCS) activity greater than 4.0 μCi/ml I-131 dose equivalent

B. RCS activity greater than 0.2 μCi/ml I-131 dose equivalent but less than limit above for more than 48 hours

3.1.2 Gaseous

A. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than 1.2×10^4 mR/hr

B. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) increase of greater than 2.4×10^3 mR/hr in 30 minutes.

3.2 Alert

3.2.1 Liquid

Reactor coolant activity greater than 40 μCi/ml I-131 dose equivalent

3.2.2 Gaseous

Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than 1.2×10^5 mR/hr

3.0 Abnormal Core Conditions and Core Damage (Cont'd)

3.3 Site Area Emergency

- Reactor Coolant System activity is greater than 400 μCi/ml I-131 dose equivalent.

OR

- Loss of two-out-of-three fission product barriers listed in Step 2.4.1.of this attachment.

3.4 General Emergency

3.4.1 Any two functional high range drywell radiation monitors (D22-RI-4195, 4196, 4197, and 4198) reading greater than 5000 R/hr

OR

3.4.2 Reactor Coolant System activity is greater than 4000 μCi/ml I-131 dose equivalent

OR

3.4.3 Loss of two-out-of-three fission product barriers listed in Step 2.4.1 of this attachment with a potential for loss of the third barrier.

4.0 Abnormal Radiological Effluent or Radiation Levels

4.1 Notification of Unusual Event

4.1.1 Liquid Release

Any unplanned release from the liquid waste system resulting in activity levels in the discharge canal greater than those in 10CFR20, Appendix B, Table II, Column 2.

4.1.2 Gaseous Release

Any gaseous release which exceeds the dose limit specified in ODCM 7.3.7 (i.e., exceeding the noble gas instantaneous dose rate limit as evaluated by 0E&RC-2020.

4.1.3 Any building evacuation based on confirmed radiological conditions (i.e., greater than 10 dac airborne [except precautionary evacuations]).

4.2 Alert

4.2.1 Liquid Release

Any liquid release resulting in activity concentration levels in the discharge canal that are greater than 10 times those given in 10CFR20, Appendix B, Table II, Column 2 (10 times the concentration listed in Unusual Event).

4.2.2 Gaseous Release

Any gaseous release which exceeds 10 times the dose rate limit specified in ODCM 7.3.7 (i.e., exceeding 10 times the noble gas instantaneous dose rate limit as evaluated by 0E&RC-2020.

4.0 Abnormal Radiological Effluent or Radiation Levels (Cont'd)

4.2.3 In-Plant Leak or Spill

A. Any area radiation monitor or continuous air monitor off-scale high and radiological conditions are confirmed.

B. Any site evacuation based on confirmed radiological conditions.

C. Reactor Building closed cooling water monitor (D12-RM-K606) off-scale high and high activity is confirmed by sampling.

4.3 Site Area Emergency

4.3.1 Projected dose exceeding 50 mRem Whole body (TEDE) **OR** exceeding 250 mRem Thyroid (CDE) at site boundary.

4.3.2 Measured dose rate exceeding 100 mR/hr at site boundary.

4.3.3 Measured I-131 dose equivalent concentration exceeds 3.9E-7 μCi/cc at the site boundary.

4.4 General Emergency

4.4.1 Offsite release resulting in a dose exceeding one (1) Rem Whole Body (TEDE) **OR** five (5) Rem Thyroid (CDE) at the Site Boundary as indicated by dose projection or field data.

4.4.2 Measured I-131 Dose Equivalent concentration exceeding 3.9E-6 μCi/cc at the site boundary.

5.0 Loss of Shutdown Functions: Decay Heat and Reactivity

5.1 Notification of Unusual Event

N/A

5.2 Alert

5.2.1 Complete loss of ability to maintain plant in cold shutdown:

A. Loss of essential service water loops, or Loss of RHR Loops A and B.

AND

B. Loss of Condenser Condensate System.

AND

C. Either:

1. Coolant temperature exceeds 212°F,

OR

2. Uncontrolled temperature rise approaching 212°F.

5.2.2 Failure of the Reactor Protection System to initiate and complete a scram, indicated on Panel A-5, which brings the reactor to a subcritical condition as indicated by full core display panel P603 and neutron monitoring instruments (APRM and IRM).

5.3 Site Area Emergency

Failure of the Reactor Protection System to initiate and complete a scram as indicated by Section 5.2.2 above.

AND

Failure of standby liquid control to bring the reactor to a subcritical condition.

5.0 Loss of Shutdown Functions: Decay Heat and Reactivity (Cont'd)

5.4 General Emergency

5.4.1 Site Area Emergency as indicated in Section 5.3 above lasting greater than 30 minutes.

AND

5.4.2 Loss of main condenser heat removal capability indicated by MSIVs shut or loss of vacuum on condenser vacuum indicator.

AND EITHER

A. Failure of all low pressure coolant injection trains indicated on panel P601.

OR

B. Failure of all service water trains necessary for decay heat removal indicated on panel P601 (RHR Service Water) and Panel XU2 (Nuclear and Conventional Service Water).

6.0 Electrical or Power Failures

6.1 Notification of Unusual Event

6.1.1 Inability to power either 4 kV E Bus from off-site power.

OR

6.1.2 Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

6.2 Alert

6.2.1 Loss of all vital DC power.

OR

6.2.2 Inability to power either 4 kV E Bus from off-site power.

AND

A. Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

6.3 Site Area Emergency

Either Alert condition in Section 6.2.1 or 6.2.2 listed above **AND** lasting longer than 15 minutes.

6.4 General Emergency

N/A

7.0 Fire

7.1 Notification of Unusual Event

Fire within the protected area lasting longer than ten minutes.

7.2 Alert

Fire which could potentially affect vital safety-related equipment.

7.3 Site Area Emergency

Any fire that impairs the operability of any vital equipment which, in the opinion of the Site Emergency Coordinator, is essential to maintain the plant in a safe condition.

7.4 General Emergency

Any fire which in the opinion of the Site Emergency Coordinator could cause massive common damage to plant systems.

8.0 Control Room Evacuation

8.1 Notification of Unusual Event

N/A

8.2 Alert

Evacuation of Control Room anticipated or required with control of shutdown established from local stations.

8.3 Site Area Emergency

Evacuation of Control Room **AND** local control of shutdown is not established in 15 minutes.

8.4 General Emergency

N/A

9.0 Loss of Monitors or Alarms or Communication Capability

9.1 Notification of Unusual Event

9.1.1 Loss of communications capability as determined by the Communication Failures Decision Matrix (Section 9.5).

9.1.2 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 with the affected unit in Operational Condition 1, 2, or 3 for > 15 minutes;

AND

Compensatory (non-alarming) indications are available.

9.2 Alert

9.2.1 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 with the affected unit in Operational Condition 1, 2, or 3 for > 15 minutes;

AND

Either;

- Compensatory (non-alarming) indications are **NOT** available.

OR

- A plant transient is in progress.

9.0 Loss of Monitors or Alarms or Communication Capability (Cont'd)

9.3 Site Area Emergency

9.3.1 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 with the affected unit in Operational Condition 1, 2, or 3;

AND

- Compensatory (non-alarming) indications are NOT available.

AND

- A plant transient is in progress.

AND

- Plant safety function indications (reactor power, reactor level, reactor pressure, containment parameters) are **NOT** available.

9.4 General Emergency

N/A

9.0 Loss of Monitors or Alarms or Communication Capability (Cont'd)

9.5 COMMUNICATION FAILURES DECISION MATRIX (DECLARATION OF A NOTIFICATION OF UNUSUAL EVENT)

NOTE: See 0PEP-3.1.3 for alternate communication means.

		NOTIFICATION OF UNUSUAL EVENT
1.	Complete Loss of Selective Signaling	N
2.	Loss of NRC Emergency Notification System (ENS)	N
3.	Loss of Bell South Network	N
4.	Loss of CP&L Network (Caronet)	N
5.	Loss of Selective Signaling Phone and ENS	N
6.	Loss of Selective Signaling Phone and Bell South Network (Long Distance Calling)	N
7.	Loss of Selective Signaling Phone and CP&L Network (Caronet)	N
8.	Loss of ENS and Bell South Network	N
9.	Loss of ENS and CP&L Network (Caronet)	N
10.	Loss of BOTH Bell South and CPL Network (Caronet)	Y
11.	Loss of Selective Signaling Phone, ENS, and Bell South Network (Long Distance Calling)	N
12.	Loss of Selective Signaling Phone, ENS, and CP&L Network (Caronet)	N
13.	Loss of All Phone Communication: Selective Signaling Phone, ENS, Bell South, [Long Distance Calling] and CP&L Network (Caronet)	Y

10.0 Fuel Handling Accident

10.1 Notification of Unusual Event

N/A

10.2 Alert

10.2.1 Fuel handling accident involving damage to new or spent fuel indicated by:

A. Observation/report **AND** alarm on:

1. Process Reactor Building ventilation RAD monitor D12-K609A, B or D12-RR-R605.

OR

2. Reactor Building roof ventilation monitor CAC-AIQ-1264-3.

OR

3. Refuel floor area monitor ARM channel 1-28 or 2-28.

10.3 Site Area Emergency

10.3.1 Major damage to spent fuel indicated by:

A. Observation of substantial damage to multiple fuel assemblies, or observation that water level has dropped below the top of the fuel.

AND

B. Indications or alarms listed in Attachment 1, Section 10.2.1.A above.

10.4 General Emergency

N/A

11.0 Security Threats

11.1 Notification of Unusual Event

Declaration of a security alert as defined by the Security Contingency Plan.

11.2 Alert

Declaration of a security emergency as defined by the Security Contingency Plan.

11.3 Site Area Emergency

Physical attack on the plant involving imminent occupancy of the Control Room, auxiliary shutdown panels, and other vital areas.

11.4 General Emergency

Physical attack on the plant has resulted in unauthorized personnel occupying the Control Room and other vital areas.

12.0 Specific LCOs

12.1 Notification of Unusual Event

12.1.1 Loss of containment operability requiring shutdown by Technical Specifications and shutdown is not achieved within required time period.

12.1.2 Loss of engineered safety feature requiring shutdown by Technical Specifications and shutdown is not achieved within required time period.

12.2 Alert

N/A

12.3 Site Area Emergency

N/A

12.4 General Emergency

N/A

13.0 Hazards to Plant Operations

13.1 Notification of Unusual Event

13.1.1 Aircraft crash within site boundaries with the potential to endanger safety-related equipment.

13.1.2 Unplanned explosion within the site boundaries with the potential to endanger safety-related equipment.

13.1.3 Release of toxic or flammable gas that could endanger personnel.

13.1.4 Turbine rotating component failure causing rapid plant shutdown.

13.2 Alert

13.2.1 Explosion, aircraft crash, or missile resulting in major damage to structures housing safety-related systems.

13.2.2 Unplanned and uncontrolled entry of toxic or flammable gases into vital areas in sufficient quantities to endanger personnel or the operability of safety-related equipment.

13.2.3 Turbine failure causing penetration of its outer casing.

13.3 Site Area Emergency

13.3.1 Explosion, aircraft crash, or missile resulting in major damage to safe shutdown equipment with plant not in cold shutdown.

13.3.2 Uncontrolled entry of flammable or toxic gases into vital areas where lack of access constitutes a safety problem with plant not in cold shutdown.

13.4 General Emergency

Any major internal or external event substantially beyond design basis which could cause massive common damage to plant systems.

14.0 Natural Events

14.1 Notification of Unusual Event

14.1.1 Alarm on seismic monitor **AND** confirmation of earthquake.

14.1.2 Hurricane warning issued.

14.1.3 Tornado on site.

14.2 Alert

14.2.1 Earthquake registering greater than 0.08g on seismic instrumentation.

14.2.2 Any adverse weather conditions that causes a loss of function of two or more safety trains.

14.2.3 Tornado striking inside protected area resulting in major damage to structures housing safety-related systems.

14.2.4 Hurricane winds on site estimated:

A. ≥ 130 mph at 30 ft above ground level

B. ≥ 180 mph at 300 ft above ground level

14.3 Site Area Emergency

14.3.1 Earthquake registering greater than 0.16g on seismic instrumentation with plant not in cold shutdown.

14.3.2 Flood, low water, or hurricane surge greater than design levels or failure to protect vital equipment at lower levels and plant not in cold shutdown.

14.0 Natural Events (Cont'd)

14.3.3 Plant not in cold shutdown with hurricane winds on site estimated:

A. ≥ 130 mph at 30 ft above ground level

B. ≥ 180 mph at 300 ft above ground level

14.4 General Emergency

Any major natural event substantially beyond design basis which could cause massive common damage to plant systems.

15.0 Shift Superintendent/Site Emergency Coordinator Judgments

When any condition exists which indicates a necessity for an increased level of awareness or readiness above previous plant conditions, the Shift Superintendent/Site Emergency Coordinator should use his judgment to declare the appropriate emergency status for the plant.

15.1 Notification of Unusual Event

Plant conditions exist that warrant increased awareness by plant staff such as exceeding any Technical Specification safety limit.

15.2 Alert

Plant conditions exist that reflect a significant degradation in the safety of the reactor, but releases from this event would be small.

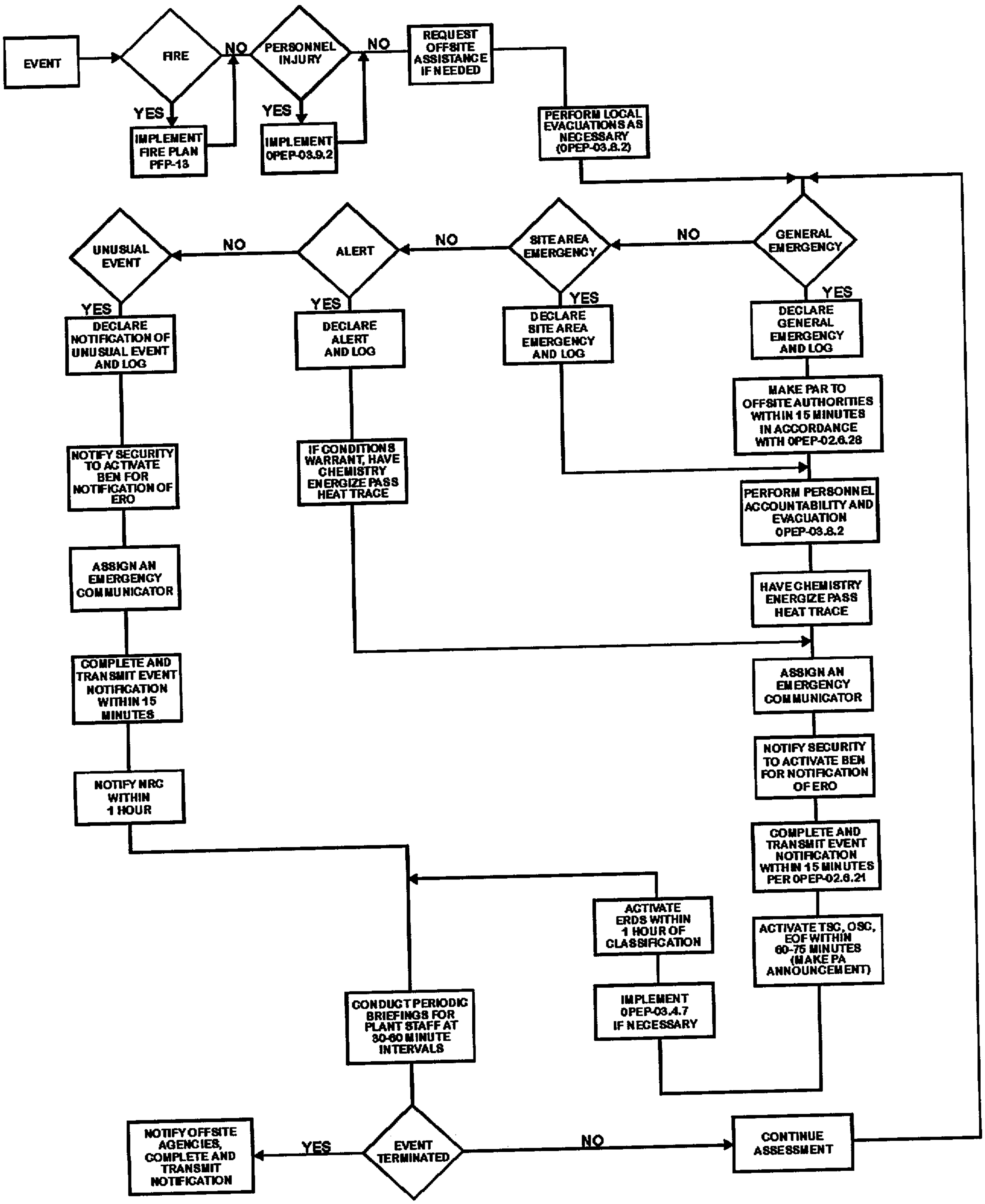
15.3 Site Area Emergency

Plant conditions exist that involve major failures of equipment and that will lead to core damage. Unless corrective action is taken, significant radiation releases may occur.

15.4 General Emergency

Plant conditions exist that make a release of a large amount of radioactivity in a short time possible; any core melt situation.

Site Emergency Coordinator Actions Flow Chart



REVISION SUMMARY

Revision 47 of 0PEP-02.1 consists of the following changes:

- Eliminated EAL reference to coolant gross activity (100/E-bar) based on Tech Spec bases change which removed requirement.

NOTE: Emergency Action Level Flowcharts, Pages 1 and 2 will be Revision 47.