INITIAL SUBMITTAL

ROBINSON EXAM 2001-301

MARCH 26 - APRIL 2, 2001

INITIAL SUBMITTAL - RO ONLY
WRITTEN EXAMINATION QUESTIONS

Question: 16

Which ONE (1) of the following conditions would be **REQUIRED** to be entered by the Reactor Operator in the Control Operator's Log?

a. Test data for an unsatisfactory Operations Surveillance Test

b. Entry into a Technical Specification LCO Action Statement

c. Name of on-shift person relieving an Auxiliary Operator who went home sick

d. Change in Secondary Chemistry Action Level

Answer:

d. Change in Secondary Chemistry Action Level

QUESTION NUMBER: 16
TIER/GROUP: *RO* 3 *SRO*
K/A: 2.1.18

Ability to make accurate, clear and concise logs, records, status boards, and reports.

K/A IMPORTANCE: *RO* 2.9 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: OMM-001-11-02

EXPLAIN the requirements for maintaining operations records and logs in accordance with OMM-001-11

REFERENCES: OMM-001-11

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since start and completion of OSTs are required entries, test data is not required.

b. Plausible since entries into TS LCOs actions are important information, but only entered by CRSS or SSO to ensure accuracy.

c. Plausible since this effects shift manning requirements, but not a required entry.

d. **CORRECT** This is a required log entry per OMM-001-11.

DIFFICULTY:
Comprehensive/Analysis [ ] *Knowledge/Recall* [X] *Rating* 2

Knowledge of logkeeping requirements for the reactor operator

REFERENCES SUPPLIED:

8.3.4 Control Operator's Logs

1. The CO's Log is maintained by the RO. The narrative log is a vital portion of the shift records and contains notations of plant conditions, operations, and events. It is maintained on a shift basis to record the plant status and events in chronological order. Log entries **SHALL** include, but are not limited to, the following:

> **NOTE:** To ensure accuracy and eliminate redundancy, TECH SPECS LCO Action Statement entries should be recorded only by the SSO or CRSS.

- Plant status.
- Changes in Generator output.
- Changes in Reactor power level.
- Starts/stops/trips of equipment controlled from the RTGB, both automatic and manual, with a brief description of the reason. (NCR 00012657, CR 9902062)

Example: Started Charging Pump "A" and stopped Charging Pump "C" to allow Maintenance to check packing for leakage.

- Change of auxiliary system and configuration.
- Surveillance tests started and completed. These may include OSTs, ESTs, MSTs, and special tests which are performed from the Control Room Complex, directs or affects Control Room Complex operations or affects plant production (the test data results need not be logged since they are recorded in the specific test).

Example: Completed partial OST-353 to return SI-844A to service. Test SAT.

- Reactor Trips.

8.3.4.1 (Continued)

- Instrument or equipment malfunctions or failures. The entry should include the time the component is removed from service, a brief description of the problem, any compensatory actions taken, and the number of any Work Request written.

Example: R-16 removed from service due to frequently spiking high. Notified E&RC to begin collecting grab samples. WR 9X-ABCD1 written.

- Unusual trends or conditions observed.
- Major in-plant electrical switching.
- Starting and stopping of Gaseous or Liquid Waste Releases (list Waste Release Permit Number).
- Setpoint changes which are performed.
- Declaration of and changes to Secondary Chemistry Action Levels.
- Annunciators received that are not the result of operator action or are not expected as a result of evolutions in progress (such as surveillance tests, clearing of equipment or equipment manipulation). It is acceptable to use a rough log for the accumulation of recurring annunciators and to document these annunciators as a single log entry near the end of shift. During plant transients when a large number of annunciators are received in a short period of time, this logging requirement can be waived.
- When annunciators are received and none of the actions specified in the APP are taken in response to the alarm because it is determined that none of the prescribed actions would be effective in eliminating the diagnosed cause, then the basis for not taking the prescribed actions should be logged. This basis should include the plant conditions, diagnosis of the event, conclusions of the diagnosis, and any alternate actions that are taken or justification for taking no actions at all.

Question: 17

Given the following conditions:

- RCS temperature is 362 °F.
- RCS pressure is 900 psig.
- RCP pump bearing temperatures are increasing.
- RCP seal injection and seal leakoff flows are:

RCP	SEAL INJECTION	SEAL LEAKOFF
'A'	5.8 gpm	1.2 gpm
'B'	6.7 gpm	0.9 gpm
'C'	6.5 gpm	1.3 gpm

Which ONE (1) of the following actions must be taken to permit opening CVC-307, PRI SEAL BYP ISO?

a. Increase RCS pressure more than 100 psig

b. Lower RCS temperature more than 12 °F

c. Increase RCP 'A' seal injection more than 0.2 gpm

d. Increase RCP 'B' seal leakoff more than 0.1 gpm

Answer:

c. Increase RCP 'A' seal injection more than 0.2 gpm

QUESTION NUMBER: 17
TIER/GROUP: *RO* 2/1 *SRO*
K/A: 003 2.1.32

Ability to explain and apply all system limits and precautions (Reactor Coolant Pump).

K/A IMPORTANCE: *RO* 3.4 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 3 *55.43(b) SRO*

OBJECTIVE: RCS-09

EXPLAIN the normal operation of the Reactor Coolant System control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: OP-101

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since 1000 psig is a trigger pressure, but pressure must be below 1000 psig, not above.

b. Plausible since 350 °F is used for many applications, such as RHR system operations and Mode changes, but has no effect on the operation of this valve.

c. **CORRECT** Required conditions are RCS pressure between 100 and 1000 psig, all seal leakoff valves open, any #1 seal leakoff flow < 1 gpm, all seal injection flows > 6 gpm. Only the seal injection flow requirement is not met.

d. Plausible since 1 gpm leakoff is a trigger value, but at least one leakoff flow must be below 1 gpm, not all above.

DIFFICULTY:
Comprehensive/Analysis [X] *Knowledge/Recall* [ ] *Rating* 3

Application of RCP precautions and limitations to determine required action

REFERENCES SUPPLIED:

4.2.1 (Continued)

10. The No. 1 Seal Bypass Valve is used when RCS pressure is less than 1000 psig, to prevent the RCP pump bearing temperature and the No. 1 seal leakoff temperature from reaching alarm levels. Prior to opening CVC-307, PRI SEAL BYP ISO, the following conditions shall all be satisfied:

    a. RCS pressure is between 100 and 1000 psig.

    b. All three No. 1 Seal Leakoff valves (CVC-303A, B, C) are open.

    c. Any No. 1 seal leakoff flow rate is less than 1 gpm.

    d. Seal injection flow rate to each RCP is greater than 6 gpm.

11. Any change greater than 10°F on No. 1 and No. 2 seal leak-off for unknown reasons should be investigated.

12. Only one RCP is to be started at any one time.

13. A Reactor Coolant Pump should not be operated continuously until the RCS has been thoroughly vented.

14. If Component Cooling Water flow to the RCP motor is lost, the RCP shall be stopped before either the upper or lower bearing temperature has increased to 200°F IAW AOP-014. (CR 95-02015 and ESR 95-01075)

15. Two Containment Fan Coolers are required for normal operation with RCS temperature greater than 140°F. The intent of this requirement is to maintain RCP motor winding temperature less than 248°F.

Question: 18

Given the following conditions:

- A Reactor Trip and SI has occurred from an unisolable main steam line break on SG 'A'.
- Diagnostic actions are in progress.
- SG 'A' has been isolated per EPP-11, "Faulted SG Isolation", and is dry.
- RCS temperature has been stabilized by dumping steam from the intact SGs following the dryout of the SG 'A'.

Which ONE (1) of the following would be the **FIRST** indication to the operators that a 250 gpm tube leak has subsequently developed in SG 'A'?

a. R-31A, Main Steamline Monitor

b. R-19A, SG Blowdown Radiation Monitor

c. Pressurizer level decreasing

d. SG 'A' level increasing

Answer:

c. Pressurizer level decreasing

QUESTION NUMBER: 18
TIER/GROUP: *RO* 1/2 *SRO*
K/A: 037AA1.11

Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: PZR level indicator

K/A IMPORTANCE: *RO* 3.4 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 5 *55.43(b) SRO*

OBJECTIVE: PATH-1-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in PATH-1 by explaining the basis of each.

REFERENCES: PATH-1-BD
EPP-16

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number PATH-1-03 016

JUSTIFICATION:

a. Plausible since this would provide indication during power operations, but following a reactor trip the N-16 detectors on the steam lines would not be effective.

b. Plausible since this would provide indication during power operations, but following a reactor trip and safety injection blowdown would be isolated.

c. **CORRECT** Pressurizer level would decrease regardless of which SG had a tube rupture. Due to the plant conditions, none of the other 'normal' indications of a tube rupture would be available.

d. Plausible since this would provide indication if the SG were not faulted, but any leakage to the faulted SG will immediately flash to steam so no level increase would be noted.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of the effectiveness of diagnostic indications during abnormal conditions

REFERENCES SUPPLIED:

GRID	WOG STEP	BASIS/DIFFERENCES

WOG BASIS

PURPOSE: To identify any faulted SGs (failure in secondary pressure boundary)

BASIS:

An uncontrolled SG pressure decrease or a completely depressurized (i.e., near containment or atmospheric pressure) SG indicates failure of the secondary pressure boundary. Isolation is to be performed using E-2, FAULTED STEAM GENERATOR ISOLATION.

KNOWLEDGE:

"Uncontrolled" means not under the control of the operator, and incapable of being controlled by the operator using available equipment

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

B18 23 RNP STEP

ANY S/G COMPLETELY DEPRESSURIZED

WOG BASIS

See step 23 above.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

D18 23 RNP STEP

RESET SPDS AND INITIATE MONITORING OF CRITICAL SAFETY FUNCTION STATUS TREES (with transition to EPP-11)

WOG BASIS

See step 23 above.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

E-4 24 RNP STEP

R-19'S, R-31'S, AND R-15 RAD LEVELS NORMAL

GRID	WOG STEP	BASIS/DIFFERENCES

WOG BASIS

PURPOSE: To identify any ruptured SGs (failure in primary to secondary pressure boundary)

BASIS:

Abnormal condenser air ejector radiation, SG blowdown or steamline radiation indicates primary to secondary leakage. Optimal recovery in dealing with a steam generator tube rupture is provided in E-3, STEAM GENERATOR TUBE RUPTURE.

KNOWLEDGE:

"Normal" means the value of a process parameter experienced during routine plant operations.

RNP DIFFERENCES/REASONS

The path eliminates the negative of the ERG high level step. These radiation monitors, when in alarm, are used as indicators of S/G tube leakage. This satisfies RAIL item 91R0043.

Interpretation

Normal also includes instrument behavior observation over time. If levels are increasing or have gone up and then back down due to manual or automatic actions that isolate the RMS sample path, the levels are not considered normal.

SSD DETERMINATION

This is an SSD per criterion 11.

E-4 25 RNP STEP

R-2, R-32A, R32B RAD LEVELS NORMAL

E-5 RNP STEP

CV PRESSURE NORMAL

E-5 RNP STEP

CV SUMP LEVEL NORMAL

WOG BASIS

PURPOSE: To identify any failure in the RCS pressure boundary into the containment

BASIS:

Abnormal containment radiation, pressure, or recirculation sump level is indicative of a high energy line break in containment. Since the SGs have been determined to be non-faulted in an earlier step, then the break must be in the reactor coolant system. For smaller size breaks containment pressure and recirculation sump level may not increase for a period of time; however, containment radiation would be apparent. Guideline E-1, LOSS OF REACTOR OR SECONDARY COOLANT, is used for breaks in the RCS.

KNOWLEDGE:

"Normal" means the value of a process parameter experienced during routine plant operations.

RNP DIFFERENCES/REASONS

There are no significant differences.

SSD DETERMINATION

This is not an SSD.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. Open Foldout D

2. Perform The Following:

   a. Reset SPDS

   b. Initiate monitoring of Critical Safety Function Status Trees

Question: 19

While performing OST-012, "Power Range Calorimetric During Power Operation (Manual) Daily," which ONE (1) of the following will result in **ACTUAL** power being **HIGHER THAN INDICATED** power?

a. SG Blowdown is secured prior to starting the data collection

b. MDAFW Pump 'A' is operating with flow being delivered to a SG

c. Indicated feedwater temperature used is lower than actual

d. Indicated feedwater flow used is higher than actual

Answer:

b. MDAFW Pump 'A' is operating with flow being delivered to a SG

QUESTION NUMBER: 19
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 015K5.04

Knowledge of the operational implications of the following concepts as they apply to the NIS: Factors affecting accuracy and reliability of calorimetric calibrations

K/A IMPORTANCE: ***RO*** 2.6 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: NI-10

EXPLAIN the operation of the Nuclear Instrumentation System.

REFERENCES: OST-012

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number RNP-RO-2000 015

JUSTIFICATION:

a. Plausible since blowdown is a consideration in the calorimetric. Has no effect provided no changes are made to blowdown during the data collection period.

b. CORRECT AFW flow is not accounted for in the calorimetric. The amount of heat actually required to raise AFW temperature to saturation would be ignored, thereby causing calculated power (and indicated after adjustment) to be lower than actual.

c. Plausible since feed temperature is a consideration in the calorimetric. If indicated feed temperature was lower than actual, the calculation (and indicated power) would require more heat to raise temperature, so it would be higher than actual.

d. Plausible since feed flow is a consideration in the calorimetric. If indicated feed flow was higher than actual, more heat would be required to raise the additional feed flow to saturation, so calculated (indicated) would be higher than actual.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 4

Analysis of the effects of various inputs to the calorimetric calibration

REFERENCES SUPPLIED:

4.6 Rounding off the readings taken on the PR nuclear instruments shall be IAW the following guidelines (CR 93-15019):

4.6.1 **IF** indicated power is less than 100%, **THEN** round down to the nearest 0.5% increment (EXAMPLES: a reading of 99.7 would be recorded as 99.5, and 99.3 would be recorded as 99.0).

4.6.2 **IF** indicated power is greater than 100%, **THEN** round up to the nearest 0.5% increment (EXAMPLES: a reading of 100.2 would be recorded as 100.5, and 100.6 would be recorded as 101).

4.7 This procedure has been screened in accordance with PLP-037 criteria and determined to be a Case Three procedure.

4.8 When adjusting an NI Power Range gain with another OOS, extreme care must be utilized. Any electronic spiking has the potential of causing a Reactor Protection System actuation. (CR 97-01677)

4.9 Inaccuracies in the calorimetric increase in magnitude when less than 70% power. Reactor Engineering should be notified prior to adjusting any power range channel down when less than 70% to reduce the probability of non-conservative operation. It is allowable to raise indicated power to match calculated power. (CR 98-01362)

4.10 Operation of any MDAFW Pump with AFW Flow to a S/G will cause the calorimetric to be non-conservative. Do not perform a calorimetric when a MDAFW Pump is in service and flowing to a S/G. (CR 96-01991)

DATA SHEET POWER RANGE CALORIMETRIC DURING POWER OPERATION

ITEM	LOOP			FORMULA/SOURCE
	1	2	3	
AVG STM PRESS (PSIA)				AVG STEAM PRESS (PSIG) + 14.7
ENTHALPY OF STEAM (h_g)				STEAM TABLES (Saturated Steam @ Steam Generator Pressure)
ENTHALPY OF FEED (h_f)				STEAM TABLES (Saturated Liquid @ Feedwater Temperature)
BTU/LB CHANGE IN ENTHALPY FROM FEED TO STEAM (Δh_{g-f})				$\Delta h_{g-f} = [h_g - h_f]$
SQ. ROOT OF SPECIFIC WGT $(LB/FT^3)^{1/2}$ $\sqrt{\Upsilon}$				$1/\sqrt{V_f}$ FROM STEAM TABLES (@ Steam Generator Pressure and Feedwater Temperature)
THERMAL AREA FACTOR (Fa)				From Attachment 8.2
FEED FLOW DELTA P SQ. ROOT $(in)^{1/2}$ $\sqrt{\Delta P_f}$				SQUARE ROOT OF FEEDFLOW DIFFERENTIAL PRESSURE
FLOW CONST	32013	31896	31944	
FEEDWATER MASS FLOW RATE (LB/HR)	$X10^6$	$X10^6$	$X10^6$	$\dot{m}_f = [\sqrt{\Upsilon}]\,[Fa]\,[\sqrt{\Delta P_f}]$ x Flow Constant
GROSS LOOP POWER (BTU/HR)	$X10^6$	$X10^6$	$X10^6$	$\dot{m}_f \times \Delta h_{g-f}$
	TOTAL			
Q TOTAL (BTU/HR)	$X10^6$			Q(T)=Q(LOOP1) + Q(LOOP2) + Q(LOOP3)
Q NET (MW)				$[Q(T) - 30.677 \times 10^6]/3.4121 \times 10^6$
% POWER				$\dot{Q}$ NET / 23

LIMITS AT CALCULATED POWER: HIGH (CALCULATED + 2%) ______%

LOW (CALCULATED - 2%) ______%

Question: 20

Given the following conditions:

- Refueling Operations are schedule to commence.
- RCS Boron Concentration is currently 1825 ppm.

Which ONE (1) of the following describes the required RCS boron concentration for refueling operations?

a. Boron concentration is adequate

b. Boron concentration must be increased by a minimum of 75 ppm

c. Boron concentration must be increased by a minimum of 125 ppm

d. Boron concentration must be increased by a minimum of 175 ppm

Answer:

c. Boron concentration must be increased by a minimum of 125 ppm

QUESTION NUMBER: 20
TIER/GROUP: ***RO*** 3 ***SRO***
K/A: 2.2.26

Knowledge of refueling administrative requirements.

K/A IMPORTANCE: ***RO*** 2.5 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: FH-12

STATE the Technical Specification Limitations and explain the bases for the FH System.

REFERENCES: TS 3.9.1
COLR 2.8

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number FH-12 003

JUSTIFICATION:

a. Plausible if misconception regarding required boron concentration as this is a reasonably high value, but actual required concentration is 1950 ppm.

b. Plausible if misconception regarding required boron concentration as this is a reasonably high value, but actual required concentration is 1950 ppm.

c. **CORRECT** Required boron concentration for refueling is 1950 ppm, so a boration is required to raise boron concentration an additional 125 ppm.

d. Plausible since this would meet required boron concentration, but minimum required is 1950 ppm.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of required boron concentration for refueling operations

REFERENCES SUPPLIED:

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	72 hours

2.7 Shutdown Margin Requirements (SDM) (ITS 3.1.1, 3.4.5, 3.4.6)

2.7.1 The Mode 1 and Mode 2 required SDM versus RCS boron concentration is presented in Figure 5.0.

2.7.2 The Mode 3 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% $\Delta k/k$.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.3 The Mode 4 SDM requirements are as follows:

a) With at least 2 reactor coolant pumps in operation, the SDM shall be greater than or equal to that specified in Figure 5.0.

b) With less than 2 reactor coolant pumps in operation and the rod control system capable of rod withdrawal, the SDM shall be greater than or equal to 4% $\Delta k/k$.

c) With less than 2 reactor coolant pumps in operation and with the rod control system not capable of rod withdrawal, the SDM shall be greater than or equal to that specified in Figure 5.0.

2.7.4 The minimum required SDM for Mode 5 is 1% $\Delta k/k$.

2.7.5 The minimum required SDM for Mode 6 is 6% $\Delta k/k$.

2.8 Refueling Boron Concentration (ITS 3.9.1)

2.8.1 In Mode 6 the minimum boron concentration shall be 1950 ppm.

FH-12 003

Which ONE (1) of the following describes the requried RCS boron concentration prior to refueling operations?

A. 1500 ppm

B. 1800 ppm

✓C. 1950 ppm

D. 2050 ppm

Question: 36

Given the following conditions:

- A reactor trip and SI have occurred.
- Containment pressure is 2 psig.
- All RCPs have been secured.
- EPP-007, "SI Termination," is being implemented.
- RVLIS Upper Range is 84%.
- Pressurizer Level is 56%.
- RCS Subcooling is 68 °F.
- SI, Phase A, and Phase B have been reset.
- OP-101 conditions have been met for starting an RCP.

Which ONE (1) of the following describes the conditions for starting an RCP?

a. All conditions have been met and an RCP may be started

b. Charging flow must be increased to raise RVLIS Upper Range at least an additional 6% before an RCP can be started

c. Charging flow must be increased to raise Pressurizer Level at least an additional 18% before an RCP can be started

d. Pressure must be increased and / or the RCS must be cooled down to raise RCS Subcooling at least an additional 6 °F before an RCP can be started

Answer:

c. Charging flow must be increased to raise Pressurizer Level at least an additional 18% before an RCP can be started

QUESTION NUMBER: 36
TIER/GROUP: *RO* 1/2 *SRO*
K/A: WE02EK3.2

Knowledge of the reasons for the following responses as they apply to the (SI Termination) Normal, abnormal and emergency operating procedures associated with (SI Termination).

K/A IMPORTANCE: *RO* 3.3 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: EPP-007-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in EPP-7 by explaining the basis of each.

REFERENCES: EPP-007

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number EPP-007-03 011

JUSTIFICATION:

a. Plausible if misconception of required values for RVLIS or pressurizer level, but neither are adequate for RCP start.

b. Plausible since RVLIS is addressed for conditions for starting an RCP, but must be > 100% or require adequate pressurizer level and subcooling.

c. **CORRECT** Required conditions for starting an RCP are RVLIS > 100% or both Pressurizer Level > 74% and Subcooling > 59 ^{o}F. Subcooling is met, but pressurizer level must be raised.

d. Plausible since subcooling is addressed for conditions for starting an RCP, but subcooling conditions are already met.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of required actions to ensure RCP start requirements are met

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
35.	Check RVLIS Upper Range - GREATER THAN OR EQUAL TO 100%	IF CV pressure has remained less than 4 psig, THEN Perform the following: • Control Charging flow to increase PZR level greater than 74%. • Increase subcooling to greater than 59°F. • Use PZR heaters to maintain PZR Liquid Temperature at saturation for RCS pressure. IF CV pressure has increased greater than 4 psig, THEN Go To Step 37.

EPP-007-03 011

Given the following plant conditions:

- A reactor trip and SI have occurred
- Crew has responded IAW the EOP network
- All RCP's have been secured
- EPP-007, SI TERMINATION, is in progress
- SI, Phase A, and Phase B have been reset

Which ONE (1) of the following describes the minimum plant conditions and the basis for starting an RCP?

A. RVLIS Upper Range > 100% **and** PZR level > 66%; Collapse void in the reactor vessel head.

✓B. RVLIS Upper Range > 100% **or** PZR level > 66%; Collapse void in the reactor vessel head.

C. RVLIS Full Range > 100% **and** RCS subcooling > 59 degrees; Establish saturated conditions in the PZR.

D. RVLIS Full Range > 100% **or** RCS subcooling > 59 degrees; Establish saturated conditions in the PZR.

Question: 37

Given the following conditions:

- Unit 2 is in mid loop operation to repair a S/G primary manway leak.
- The RCS is vented by two hot leg vents.
- RCS level is -71" and rising very slowly.
- RHR pump 'A' is in service at 3000 gpm.
- The operator notices that RHR flow and pressure is oscillating.

Which ONE (1) of the following actions would tend to stabilize RHR flow and pressure?

a. Start the RHR pump 'B' at 3000 gpm

b. Lower charging flow to stabilize RCS level

c. Lower RHR pump 'A' flow

d. Open the RV head vents

Answer:

c. Lower RHR pump 'A' flow

QUESTION NUMBER: 37

TIER/GROUP: ***RO*** 2/3 ***SRO***

K/A: 005K3.01

Knowledge of the effect that a loss or malfunction of the RHRS will have on RCS

K/A IMPORTANCE: ***RO*** 3.9 ***SRO***

10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: AOP-020-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of RHR events as directed in AOP-020.

REFERENCES: AOP-020

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-020-08 005

JUSTIFICATION:

a. Plausible if misconception that oscillations are due to inadequate heat removal, but oscillations are due to cavitation.

b. Plausible since RCS level is increasing, but level tends to increase as air enters RHR system.

c. **CORRECT** Cavitation is occurring due to too high a flow rate for the given level. Flow is to be reduced to 1500 gpm to eliminate cavitation.

d. Plausible if misconception that oscillations are due to voiding in head region, but oscillations are due to cavitation.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of conditions to determine response to RHR cavitation

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 3 of 15)

NOTE

Cavitation will be indicated by oscillations in RHR flow and pressure, accompanied by noises from the RHR Pumps.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Determine If RHR Pump Cavitation Is Occurring As Follows:	
	a. Check RCS level - ABOVE -72 INCHES (69% RVLIS FULL RANGE)	a. Perform the following: 1) Verify both RHR Pumps STOPPED. 2) Go To Section A, Loss Of RHR While At Reduced Inventory.
	b. Check the following RHR indications - CAVITATION PRESENT • FI-605, RHR TOTAL FLOW AND • Running RHR Pump Discharge Pressure Indication • RHR Pump A - PI-602A • RHR Pump B - PI-602B	b. Go To Step 10.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 4 of 15)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Perform the following:	
	a. Adjust FC-605 to reduce RHR flow to 1500 gpm.	
	b. Check Cavitation - ELIMINATED	b. Perform the following: 1) Verify both RHR Pumps STOPPED. 2) Go To Step 7.
	c. Return to procedure and step in effect	
7.	Check RCS Level - ABOVE OR EQUAL TO -36 INCHES	Go To Section A, Loss Of RHR While At Reduced Inventory.
8.	Check Reactor Vessel Head - OFF	Go To Section D, Loss Of RHR Inventory - Level Stable Or Increasing.
9.	Go To Section B, Loss Of RHR Inventory - Vessel Head Off	
10.	Check APP-001-A7, RHR HX LOW FLOW - ILLUMINATED	Go To Step 16.
11.	Adjust FC-605, RHR HX BYPASS FLOW Controller, To Restore Flow Between 3000 GPM And 3750 GPM	
12.	Check RHR Flow - GREATER THAN OR EQUAL TO 3000 GPM	Perform one of the following: • IF RHR flow is less than 500 gpm, THEN Go To Step 14. OR • IF RHR flow is greater than or equal to 500 gpm, THEN observe the NOTE prior to Step 15 and Go To Step 15.

Question: 38

Given the following conditions:

- The unit is operating at 100% power.
- 'B' EDG is under clearance to repair a leaky oil fitting.
- A tornado touches down in the switchyard.
- The transient resulting from the destruction causes a Phase Differential on the Main Generator.
- The Startup Transformer (SUT) is destroyed by the tornado.
- 'A' EDG is unable to start due to a faulty air lineup.
- After the initial transient, it is noted that **BOTH** of the Reactor Trip breaker indications are RED.

Which ONE (1) of the following describes the required operator action(s)?

a. Enter FRP-S.1, "Response to Nuclear Power Generation / ATWS," due to the ATWS

b. Enter PATH-1 due to the turbine trip and then FRP-S.1 due to the ATWS

c. Enter EPP-001, "Loss of All AC Power," due to the electrical conditions

d. Enter FRP-S.1 due to the ATWS, then EPP-001 due to the electrical conditions

Answer:

c. Enter EPP-001, "Loss of All AC Power," due to the electrical conditions

QUESTION NUMBER: 38

TIER/GROUP: *RO* 1/1 *SRO*

K/A: 055 2.4.1

Knowledge of EOP entry conditions and immediate action steps (Station Blackout).

	RO		*SRO*	
K/A IMPORTANCE:	*RO*	4.3	*SRO*	
10CFR55 CONTENT:	*55.41(b) RO*	10	*55.43(b) SRO*	

OBJECTIVE: EPP-001-02

RECOGNIZE the selected entry-level conditions of EPP-001.

REFERENCES: EPP-001

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number EPP-001-02 005

JUSTIFICATION:

a. Plausible since there is an ATWS and Subcriticality is the highest order CSFST, but EPP-1 states that CSFSTs are for information only and EPP-1 should not be exited to implement any FRPs.

b. Plausible since a reactor trip signal would have been generated, but EPP-1 states that CSFSTs are for information only and EPP-1 should not be exited to implement any FRPs.

c. **CORRECT** Loss of all AC power overrides and EPP-1 states that CSFSTs are for information only and EPP-1 should not be exited to implement any FRPs.

d. Plausible since there is an ATWS and Subcriticality is the highest order CSFST, but EPP-1 states that CSFSTs are for information only and EPP-1 should not be exited to implement any FRPs.

DIFFICULTY:

Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of hierarchy between loss of all AC and subcriticality

REFERENCES SUPPLIED:

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to respond to a loss of all AC power.

2. ENTRY CONDITIONS

   a. Upon any indication that all Main and Emergency AC Busses are de-energized.

   b. PATH-1, upon indication that E-1 and E-2 Busses are de-energized.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- Steps 1 AND 2 are Immediate Action steps.
- Critical Safety Function Status Trees are monitored for information only. This procedure is not exited to implement any Functional Restoration Procedure.
- Foldouts and concurrent AOPs should not be implemented during this procedure to prevent diluting available resources.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Reactor Trip: • Check REACTOR TRIP MAIN AND BYP BKRs - OPEN • Check Neutron flux - DECREASING	Trip Reactor.
2.	Check Both Turbine Stop Valves - CLOSED	Manually trip the Turbine. IF the Turbine will NOT trip, THEN close the MSIVs AND MSIV Bypasses.

Question: 39

Given the following conditions:

- A makeup to the Component Cooling Water (CCW) Surge Tank is being performed.
- CC-832, CC SURGE TANK MAKE-UP VALVE, is stroked full open.
- When tank level is 50%, the operator momentarily places the switch for CC-832 to close.

Assuming **NO** other operator actions are taken, which ONE (1) of the following describes the response of the CCW Surge Tank level?

a. CCW Surge Tank level will continue to rise to approximately 55% due to the stroke time of the valve

b. CCW Surge Tank level will stabilize at approximately 50%

c. CCW Surge Tank level will continue to rise to approximately 55% when the high level alarm will automatically close the valve

d. CCW Surge Tank level will eventually overflow out the vent valve

Answer:

d. CCW Surge Tank level will eventually overflow out the vent valve

QUESTION NUMBER: 39
TIER/GROUP: *RO* 1/1 *SRO*
K/A: 026AA1.05

Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm

K/A IMPORTANCE: *RO* 3.1 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 4 *55.43(b) SRO*

OBJECTIVE: CCW-08

EXPLAIN the component operation associated with each switch position for the CCW System switches and controls.

REFERENCES: OP-306
SD-013

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number CCW-07 002

JUSTIFICATION:

a. Plausible since level will continue to rise, but the valve will not continue to stroke closed and the tank will overflow.

b. Plausible since it is expected that level will stabilize when the valve is closed, but the valve will not continue to stroke closed and the tank will overflow.

c. Plausible since level will continue to rise, but the valve does not have any automatic close features.

d. **CORRECT** The makeup valve is a throttle valve. Momentarily placing it in the close position will only throttle it closed slightly. Makeup will continue and the tank will overflow.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of the operation of the CCW makeup system

REFERENCES SUPPLIED:

8.4 Infrequent Operation

8.4.1 Make-up to the Component Cooling Water System

1. Initial Conditions

   a. All the prerequisites of Section 4.0 are complete.

   b. Primary Make-up Water and Demineralized Water is in service IAW OP-915, Demineralized and Primary Water.

2. Instructions for Making-Up to the Component Cooling Water System Using Primary Make-up Water

NOTE: CC-832, CC SURGE TANK MAKE-UP VALVE is a throttle valve and will remain in position when the OPEN-CLOSE switch is released. Care should be used in opening this valve as the Surge Tank Level will rise rapidly if opened too far.

   a. Verify one Primary Water Pump is RUNNING.

   b. At the RTGB, momentarily place the OPEN-CLOSE Switch for CC-832, MAKEUP in the OPEN position.

   c. Verify a level increase on LI-614B, Comp Cool Surge Tank Level indicator.

8.4.1.2 (Continued)

d. **WHEN** the desired level has been established (normally 45 to 55%), **THEN** perform the following:

1) Stop the Primary Water pump, **AND** return switch position to AUTO.

NOTE: Holding CC-832 in the CLOSE position for one to two seconds after SHUT indication is received will ensure the valve is properly seated.

2) Close CC-832, MAKEUP.

e. Verify the Surge Tank Level is no longer increasing by observing LI-614B.

f. Notify E&C of addition of water to CCW Surge Tank (SCR 89-050).

3. Instructions for Making-up to the Component Cooling Water System Using Demineralized Water

a. Establish Communications between CC-711, DEMIN WTR MAKE-UP TO CC SURGE LINE and the Control Room.

b. Open CC-711.

c. Verify a level increase on LI-614B, COMP COOL SURGE TANK LEVEL INDICATOR.

d. **WHEN** the desired level has been established (normally 45 to 55%), **THEN** close CC-711.

e. Verify the Surge Tank Level is no longer increasing by observing LI-614B.

f. Notify E&C of addition of water to CCW Surge Tank (SCR 89-050).

5.9 CC-735, Thermal Barrier Outlet Isolation - MOV

(CDW-B-190628 Sh00230)

Valve CC-735 is operated from the two position (OPEN/CLOSED), spring return to the center, switch located on the RTGB. The valve is located outside containment downstream of FCV-626 in the Auxiliary Building pipe alley. The motor operator for the valve is powered from MCC-5. Upon receipt of a "P" signal, the valve will close.

5.10 CC-739, Excess Letdown HX. Outlet Isolation Valve, Air Operated Valve

(CWD-B-190628 Sh00229)

Valve CC-739 is operated from the RTGB using a two position (OPEN/CLOSED), spring return to center, switch. This valve is located in the Auxiliary Building pipe alley and provides CV isolation downstream of CV penetration P-22. CC-739 is an Air Operated Valve that receives operating air from the instrument air system through 125V DC solenoid valves. The solenoid valves receive power from the 125V DC Auxiliary Panel GC CKT#29. A safeguards actuation signal, "T" signal, will de-energize the solenoid valves causing CC-739 to close. Valve position indication is provided at the RTGB control switch and at the Containment Phase "A" Isolation indication on the vertical section of the RTGB.

5.11 RCV-609, Component Cooling Surge Tank Vent Isolation Valve, Air Operated Valve

(CWD-B-190628 Sh00204)

RCV-609 is currently gagged open to ensure CCW System does not over pressurize. The control switch for this valve is located on the RTGB. It is a two position (OPEN/CLOSED) spring return to center switch. The solenoid valves that control the air to the valve receive power from 125V Auxiliary DC Panel CKT#4. When Radiation Monitor RE-17 reaches its setpoint, the solenoid valves are deenergized to remove air pressure to RCV-609. When the activity in the CCW surge tank vent line is reduced to below the setpoint of RE-17, the solenoid valves for RCV-609 energize, opening the air supply to the air operator, and RCV-609 then strokes open. (Original design)

5.12 CC-832, CCW Makeup From Primary Water, MOV

(CWD-B-190628 Sh00203)

Valve CC-832 is operated from the RTGB using a two position (OPEN/CLOSED) switch. To open the valve the switch must be held in the OPEN position until the valve has reached its full stroke. To close the valve the switch must be held in the CLOSED position until the valve has reached the full closed position. It has throttle capability and will allow primary water to be admitted to the tank for makeup purposes. To throttle the valve, the control switch is held in the OPEN or CLOSED position until the valve has reached the desired throttle position. The valve has local position indicating lights. The motor operator for CC-832 is powered from MCC-6

The valve is located near the safeguard racks in the E-1/E-2 room, second level of the Auxiliary Building (outside of the RCA).

5.13 CC-749A/B, CCW From RHR Heat Exchangers A/B, MOVs

(CWD-B-190268 Sh00218/Sh00219)

Valves CC-749A and CC-749B are powered from MCC-5 and MCC-6 respectively and are operated by two position (OPEN/CLOSED), spring return to center, switches located on the RTGB. CC-749A and CC-749B are located in the RHR heat exchanger room, first level, Auxiliary Building. ESR 98-367 modifications added stem sensors to these valves.

6.0 SYSTEM OPERATION

6.1 Normal Operation

6.1.1 Plant Startup/Heat-up

At the beginning of the plant startup procedure, the water chemistry of the component cooling loop is checked and, if required, corrosion inhibitor is added to the loop. The surge tank water level is adjusted with the makeup valves. Normally, primary water is added to the CCW System by opening a motor operated valve from the RTGB. This valve (CC-832) is a throttle valve in that the motor only runs as long as the switch is held in the open or shut position. However, if necessary, demineralized water can be added to the component cooling loop by opening manual valve CC-711.

During plant heatup, while the RHR is in operation, two of the three CCW pumps are in operation. After the RHR System is removed from operation, one of the operating CCW pumps can be stopped.

6.1.2 Normal Plant Operation

Periodically, a sample of the CCW is taken by the plant E&RC technician to determine the solution pH and the concentration of inhibitor in the loop. If the solution is not within the specifications listed for CCW, the appropriate chemicals are added to the

CCW-07 002

Which ONE (1) of the following valves is operated from an RTGB control switch AND receives no other closure signals (no automatic closure signal)?

A. RCV-609, CCW Surge Tank Vent Isolation Valve

B. TCV-659A, "A" Charging Pump Oil Cooler Temperature Control Valve

C. CC-739, Excess Letdown HXer Outlet Isolation Valve

✓D. CC-832, CCW Makeup From Primary Water

Question: 40

Given the following conditions:

- RCS pressure is 1805 psig and decreasing.
- RCS temperature is 525 °F and decreasing.
- Tavg is 537°F and decreasing
- Steam Generator pressures and Steam Flows are:

SG	PRESSURE	STEAM FLOW
'A'	626 psig and decreasing	1.7×10^6 lbm/hr
'B'	745 psig and stable	0.05×10^6 lbm/hr
'C'	740 psig and stable	0.05×10^6 lbm/hr

Which ONE (1) of the following Safety Injection signals would be actuated?

a. High Steamline ΔP

b. Low Pressurizer Pressure

c. High Steam Line Flow with Low Tavg

d. High Steam Line Flow with Low Steam Line Pressure

Answer:

a. High Steamline ΔP

QUESTION NUMBER: 40
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 013A2.02

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences: Excess steam demand

K/A IMPORTANCE: ***RO*** 4.3 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: ESF-05

DESCRIBE the performance and design attributes of the major ESFAS components.

REFERENCES: SD-006
APP-004

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number ESF-04 006

JUSTIFICATION:

a. **CORRECT** A single steamline pressure being 100 psid lower than the header pressure will result in a safety injection.

b. Plausible since this is below the low pressure reactor trip, but is still above the low pressure safety injection.

c. Plausible since high steam flow coincident with low Tave results in a safety injection signal, but only if on 2/3 steam lines.

d. Plausible since high steam flow condition exists, but steamline pressure is above low pressure setpoint.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of safety injection actuation setpoints

REFERENCES SUPPLIED:

4.1.2 Reactor Coolant Temperature (ESF-Figure-1)4.1.2 Reactor Coolant Temperature

The RCS Low Tavg signal (2 of 3 channels below 543°F) is used to initiate the Safety Injection signal, when coincident with high steam flow; and close the Main Steam Isolation Valves, when coincident with high steam flow (i.e., generate the Steam Line Isolation Signal).

4.1.3 Steam Flow (ESF-Figure-1)4.1.3 Steam Flow

Hi Steam Flow (37.25% flow at no load to 20% load, increases linearly to 109% at full load) detected by at least one sensor on two of three steam lines, coincident with low Tavg (543°F) or low steam line pressure (614 psig), generates a Safety Injection signal and closes all MSIVs. Two flow controllers on each steam line are used to sense high steam line flow. This circuit is designed to detect steam line breaks downstream of the MSIVs.

4.1.4 Steam Line Pressure (ESF-Figure-1 & 3)4.1.4 Steam Line Pressure

Steam Line Pressure measurement is utilized for steam line break protection. Low steam line pressure (614 psig) in two of three main steam lines or Low Tavg (543°F) in two of three loops, coincident with high steam line flow in two-of-three main steam lines, will initiate the Steam Line Isolation and Safety Injection signals. This is to protect against: a steam line break upstream of the main steam check valves, a feed line break, and/or an inadvertent opening of a SG safety.

In addition, each steam line pressure measurement is compared with a main steam header pressure measurement to determine if a high steam line differential pressure exists. A coincidence of two-of-three steam line differential pressures (100 psid) in any one steam line, that is, steam line pressure lower than main steam header pressure, will initiate a Safety Injection signal.

The steam header pressure is electronically limited to a minimum value of 585 psig. Therefore, this SI signal must be blocked before a plant cooldown is started to prevent SI actuation when S/G pressures drop below 485 psig(approximately 467°F). The steam line differential pressure circuit detects faults upstream of the MSIVs. Since the steam line check valves prevent reverse flow to the faulted S/G, excessive steam line differential pressure does not close the MSIVs.

4.1.5 Containment Pressure (ESF-Figure-4 & 5)4.1.5 Containment Pressure

ESF-04 006

Given the following plant conditions:

- A plant cooldown is in progress in accordance with GP-007, Plant Cooldown From Hot Shutdown to Cold Shutdown
- RCS Pressure is 1900 psig and appropriate SI signals have been blocked IAW GP-007
- Tavg is 515°F
- A RCS leak is identified in the CV

Which ONE (1) of the following contains valid signals which could result in a Containment Ventilation Isolation under these conditions?

A. Hi Steamline Delta-P; an alarm on R-12, Containment Noble Gas Monitor

B. Low pressurizer pressure Safety Injection; an alarm on R-14C, Plant Effluent Noble Gas Monitor

✓C. Manual actuation of Containment Isolation Phase A; an alarm on R-12, Containment Noble Gas Monitor

D. Manual actuation of Containment Isolation Phase A; an alarm on R-14C, Plant Effluent Noble Gas Monitor

ALARM

S/G A STM LINE HI ΔP SFGRD/TRIP

AUTOMATIC ACTIONS

1. Safeguards Actuation

CAUSE

1. Steam Line Rupture upstream of MSIV and Check Valve
2. Failure to block SI with S/G "A" pressure less than 485 psig

OBSERVATIONS

1. S/G "A" Steam Flow (FI-474, FI-475)
2. Reactor trip breaker position

ACTIONS

1. **IF** the Reactor has tripped, **THEN** refer to the EOP Network.
2. **IF** the Reactor is **NOT** tripped **AND** a plant transient is in progress, **THEN** trip the Reactor **AND** refer to the EOP Network.
3. **IF** the Reactor is **NOT** tripped **AND** the plant is stable, **THEN** perform the following:
    1) Scan the RTGB for confirmation that a trip is **NOT** required.
    2) Inform the CRSS **OR** SSO of plant conditions to assist in diagnosis.
    3) **IF** no supporting indications show a plant trip is required, **THEN** the plant may remain at power for troubleshooting and repairs.

DEVICE/SETPOINTS

PC-474B (PT-468 - PT-474) / 100 psid

PC-475 (PT-466 - PT-475) / 100 psid

PC-476 (PT-464 - PT-476) / 100 psid

– Differential pressure of greater than or equal to 100 psi between the main steam header and a steam line (2/3 Channels on any steam line).

REFERENCES

1. EOP Network
2. 5379-2757 and 5379-2758, Logic Diagrams
3. CWD B-190628 SH 399 Cable AB
4. 5379-3232, Safeguards System

ALARM

HI STM FLO LO TAVG/LO SLP SFGRD/TRIP

AUTOMATIC ACTIONS

1. Safeguards Actuation
2. Main Steam Line Isolation

CAUSE

1. Steam Line Break downstream of the MSIVs and Check Valves

OBSERVATIONS

1. Reactor trip breaker position

ACTIONS

1. **IF** the Reactor has tripped, **THEN** refer to the EOP Network.
2. **IF** the Reactor is **NOT** tripped **AND** a plant transient is in progress, **THEN** trip the Reactor **AND** refer to the EOP Network.
3. **IF** the Reactor is **NOT** tripped **AND** the plant is stable, **THEN** perform the following:
   1) Scan the RTGB for confirmation that a trip is **NOT** required.
   2) Inform the CRSS **OR** SSO of plant conditions to assist in diagnosis.
   3) **IF** no supporting indications show a plant trip is required, **THEN** the plant may remain at power for troubleshooting and repairs.

DEVICE/SETPOINTS

1. High Steam Flow (1/2 flows on 2/3 lines)
   1) FC-474, FC-475 / 37.25% - 109% (Ramped from 20% to 100% Turbine PWR)
   2) FC-484, FC-485 / 37.25% - 109% (Ramped from 20% to 100% Turbine PWR)
   3) FC-494, FC-495 / 37.25% - 109% (Ramped from 20% to 100% Turbine PWR)
2. Low T_{avg} (2/3 channels)
3. TC-412E, TC-422E, TC-432E / 543°F
4. Low Steam Line Pressure (2 of 3 Channels)
5. PC-474A, PC-485A, PC-496A / 614 psig

REFERENCES

1. EOP Network
2. 5379-2758, Logic Diagram
3. 5379-3435, Block Diagram
4. CWD B-190628 SH 399 Cable AE
5. Calculation RNP/INST-1045

ALARM

PZR LO PRESS SFGRD/TRIP

AUTOMATIC ACTIONS

1. Safeguards Actuation

CAUSE

1. LOCA
2. Steam Break

OBSERVATIONS

1. PZR Pressure (PI-455, PI-456, PI-457)
2. Reactor trip breaker position

ACTIONS

1. **IF** the Reactor has tripped, **THEN** refer to the EOP Network.
2. **IF** the Reactor is **NOT** tripped **AND** a plant transient is in progress, **THEN** trip the Reactor **AND** refer to the EOP Network.
3. **IF** the Reactor is **NOT** tripped **AND** the plant is stable, **THEN** perform the following:
   1) Scan the RTGB for confirmation that a trip is **NOT** required.
   2) Inform the CRSS **OR** SSO of plant conditions to assist in diagnosis.
   3) **IF** no supporting indications show a plant trip is required, **THEN** the plant may remain at power for troubleshooting and repairs.

DEVICE/SETPOINTS

1. PC-456D, PC-457D, PC-455E / 1715 psig (2/3 Channels)

REFERENCES

1. EOP Network
2. 5379-2757, Logic Diagram
3. 5379-3439, Block Diagram
4. CWD B-190628 SH 399 Cable A

Question: 56

Given the following conditions:

- The plant is operating at 43% power.
- An electrical transient causes a momentary underfrequency condition on 4 KV Bus 1.
- Moments later, an undervoltage condition is also sensed on 4 KV Bus 1.
- The RCP powered from 4 KV Bus 1 trips.
- The other two RCPs remain running.

Which ONE (1) of the following identifies the signal which **DIRECTLY** generated the reactor trip?

a. Bus underfrequency

b. Bus undervoltage

c. Low flow

d. Pump breaker trip

Answer:

d. Pump breaker trip

QUESTION NUMBER: 56
TIER/GROUP: ***RO*** 2/2 ***SRO***
K/A: 012K6.04

Knowledge of the effect of a loss or malfunction of the concepts as the apply to the RPS: Bypass-block circuits

K/A IMPORTANCE: ***RO*** 2.9 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: RPS-11

EXPLAIN the reactor trips associated with the RPS System. Include purpose and setpoints.

REFERENCES: SD-011

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since UF on 2/3 buses will cause all RCPs to trip, but does not directly cause a reactor trip.

b. Plausible since UV on 2/3 buses will cause a reactor trip, but a single bus UV will only cause the related RCP to trip.

c. Plausible since a low flow signal would be generated in the single loop, but the UV condition would trip the reactor previous to the low flow signal so power would be below P-7 before the low flow condition was sensed.

d. **CORRECT** An undervoltage condition will cause the pump breaker to trip. The pump breaker tripping above P-8 (40%) will cause a reactor trip.

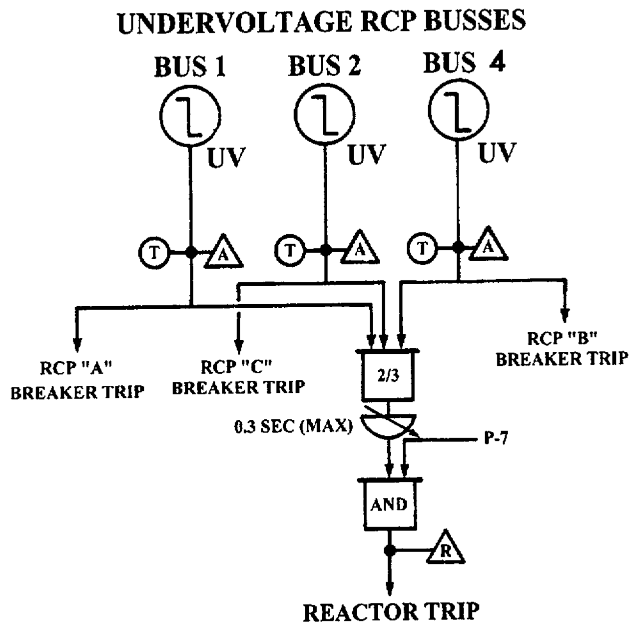
DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

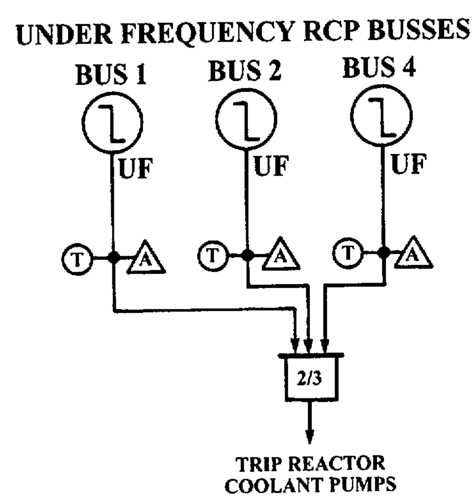
Analysis of plant conditions to determine cause of reactor trip as result of electrical perturbation

REFERENCES SUPPLIED:

UNDERVOLTAGE & UNDERFREQUENCY RCP BUS LOGIC

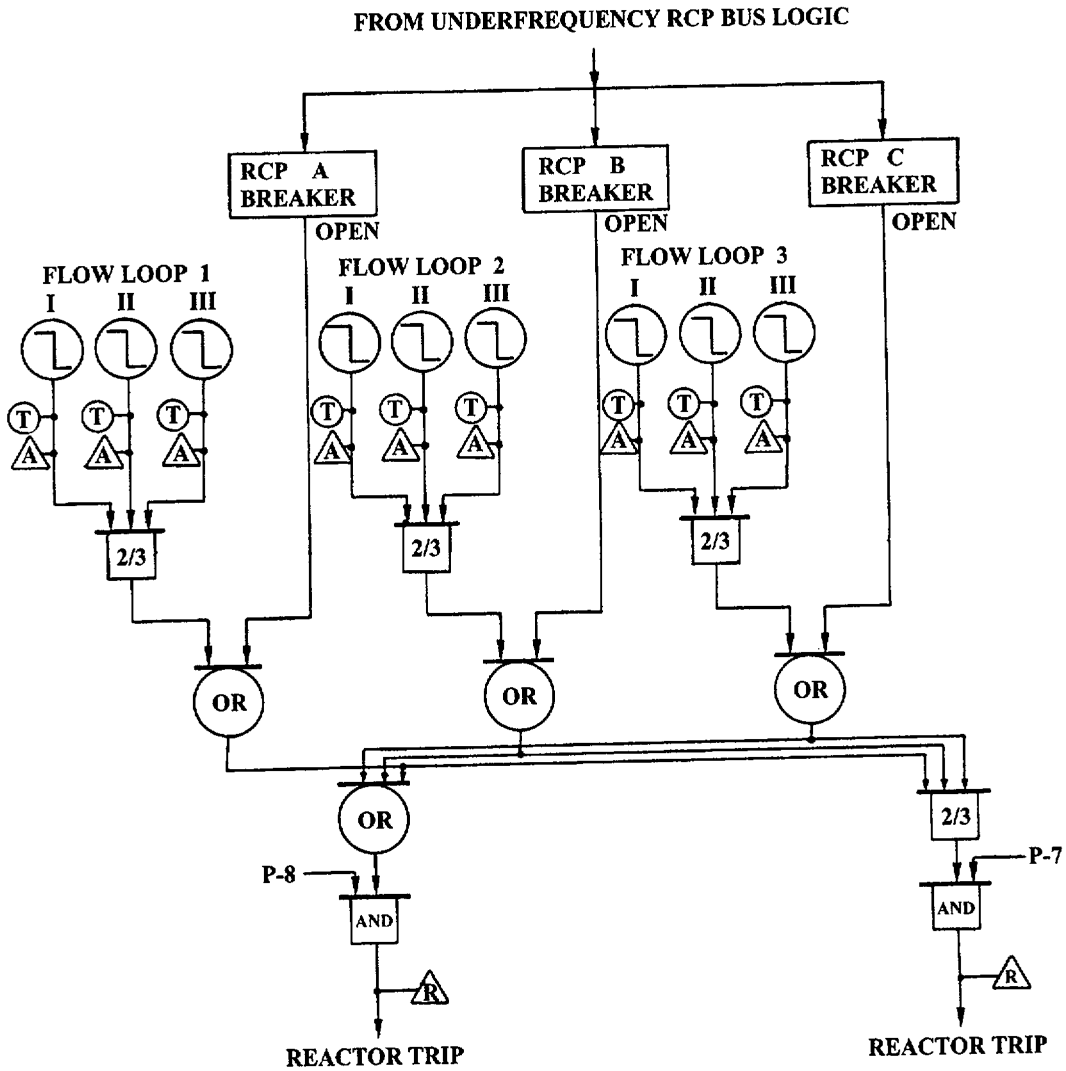
RPS-FIGURE-28 (Rev . 0)





RPS-FIGURE-27 (Rev . 0)

LOW PRIMARY COOLANT FLOW REACTOR TRIP LOGIC



Question: 57

Given the following conditions:

- An inadvertent reactor trip and safety injection have occurred.
- The SI and Phase A signals have just been reset.

Which ONE (1) of the following describes the expected position of the Normal and Emergency Inlet Dampers for the Containment Air Recirculation Fans (HVH-1 through 4) following resetting of these signals?

	NORMAL INLET DAMPERS	**EMERGENCY INLET DAMPERS**
a.	Open	Open
b.	Open	Closed
c.	Closed	Open
d.	Closed	Closed

Answer:

c.	Closed	Open

QUESTION NUMBER: 57
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 022A3.01

Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

K/A IMPORTANCE: ***RO*** 4.1 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 9 ***55.43(b) SRO***

OBJECTIVE: CVHVAC-05

DESCRIBE the performance and design attributes of the major CV HVAC, PACV and Hydrogen Recombiner Systems.

REFERENCES: SD-037

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number CVHVAC-07 003

JUSTIFICATION:

a. Plausible since until a recent modification to prevent the normal inlets from opening when SI was reset, these dampers would open.

b. Plausible since until a recent modification to prevent the normal inlets from opening when SI was reset, these dampers would open.

c. **CORRECT** On the SI normal inlet dampers automatically close and emergency inlet dampers remain open since they are failed open. When the signals are reset these components remain in the post-SI position.

d. Plausible since the normal inlet dampers remain closed, but the emergency inlet dampers will remain open.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of the operation of the Containment Ventilation systems to an SI

REFERENCES SUPPLIED:

The function of the Containment Air Recirculation Cooling system under normal power operating conditions is to remove heat from the containment atmosphere at a rate of 1.75 x 10^6 Btu/hr and thereby maintain the average dry-bulb temperature of the containment atmosphere below 120°F, except for short durations tolerable for the insulation of the electric motors, wiring, and miscellaneous electrical devices. In order to enhance CV temperature control during the summer, piping needed to inject chilled water has been installed down stream of SW Booster pump "B" (ESR 97-00383).

Condensate from the cooling coils is collected by floor drains and directed to a level column for RCS leakage detection. ESR -98-00314 modified the Condensate Measuring System (CMS). The modification replaced the four level detectors with digital indicators and installed a local annunciator with reflash ability. For example, if a CMS alarm is received on the RTGB, an acknowledge pushbutton on the local panel must be depressed to enable the reflash capability. The alarm on the RTGB will be locked in, however, any new alarms received will cause the RTGB alarm to reflash and a subsequent local alarm. The alarms will clear themselves when the alarm condition no longer exists. APP-002-E2 and OST -901 have been revised to reflect these changes.

The cooling coils of each HVH unit are supplied with 800 gpm of cooling water from the Service Water system.

Each fan is designed to supply at least 65,000 cfm at design basis accident (DBA) conditions at approximately 20 in. static pressure, 263°F, 0.162 lb/ft^3 density. The fans are direct-driven centrifugal type. Cooling coils are plate fin-tube type. Each air handling unit is capable of removing 4.0 x 10^7 Btu/hr from the containment atmosphere under DBA conditions when supplied with 750 gpm of service (cooling) water. The design maximum cooling water inlet temperature is 95°F which results in a maximum outlet temperature of 195°F under DBA conditions.

Each air handling unit has a normal air inlet damper (85,000 cfm) and an emergency air inlet butterfly valve (65,000 cfm).

The emergency air inlet butterfly valve is secured open and supplies air to the unit under all operating conditions. The instrument air tubing for the emergency air inlet butterfly valve solenoids was disconnected and removed during RFO 18 by Engineering Service Request (ESR) 97-00382. The solenoids themselves were abandoned in place.

The operation of the normal air inlet damper was modified during RFO's 17 and 18 by ESRs 97-00382 and 95-00783. A key locked three position selector switch and a new relay were installed for the normal air inlet damper on each unit. The selector switch positions are : LEFT - OPEN, CENTER - CLOSED and RIGHT - RESET. The selector switch maintains position in OPEN or CLOSED but it is spring returned to CENTER - CLOSED from RESET. [These switches are located on the DC Relay Racks in the Computer Room.] The normal air inlet dampers close on a SI SIGNAL and the new relays prevent them from automatically reopening when the SI SIGNAL is reset. These

3.1.4 HVH-9A and HVH-9B (Reactor Concrete Shield Cooling)

Number of units:	2
Fans - per unit:	1
Manufacturer:	Industrial Air
Air flow rate - per fan:	15,450 cfm
Power requirements - per fan:	25 HP

The units contain supply ductwork connected to the recirculation cooling unit's distribution header, booster fans, and exhaust ductwork.

3.1.5 HVH-1, HVH-2, HVH-3, and HVH-4 (Containment Air Recirculation Cooling)

Number of units:	4
Fans - per unit:	1
Manufacturer:	Westinghouse
Air flow rate - normal power operation:	
- with Emergency air inlet butterfly valves open	65,000 cfm
- with Emergency air inlet butterfly valves and normal air inlet dampers open	>85,000 cfm
Air flow rate - post-accident operation:	65,000 cfm
Design power requirements - per fan	
- normal power operation:	117 BHP
- post-accident operation:	244 BHP
Rated power requirements - per fan	350 HP
Cooling coils - per unit:	6
Cooling capacity - per unit	
- normal power operation:	1.75×10^6 Btu/hr
- post-accident operation:	40×10^6 Btu/hr
Cooling water flow rate - per unit:	800 gpm

Each air handling unit includes a space for roughing filters, water supplied cooling coils, and a centrifugal fan enclosed in a sheet metal casing. Supply air is drawn through the space for roughing filters during shutdown operating conditions. During normal power and accident conditions, the space for filters is not used and air is drawn directly through an air operated butterfly valve located on the unit's casing. Air discharges to the recirculation cooling unit's distribution header where it is distributed through ductwork to individual areas in containment.

Question: 58

Given the following conditions:

- The unit has experienced a loss of off-site power.
- The reactor trip and turbine trip have been verified.
- EPP-1, "Loss of ALL AC Power," was implemented until the inside AO restored power to 480V Bus E-2 per Attachment 6 of EPP-1.
- A transition has been made back to PATH-1.
- SI has **NOT** occurred and is **NOT** required.

Which ONE (1) of the following describes how power will be supplied to the Charging Pumps?

	FROM 'B' EDG	**FROM DSDG**
a.	Charging Pump 'B'	Charging Pump 'A'
b.	Charging Pump 'C'	Charging Pump 'B'
c.	Charging Pump 'B'	Charging Pump 'C'
d.	Charging Pump 'C'	Charging Pump 'A'

Answer:

d.	Charging Pump 'C'	Charging Pump 'A'

QUESTION NUMBER: 58
TIER/GROUP: *RO* 2/1 *SRO*
K/A: 004K2.03

Knowledge of bus power supplies to the Charging pumps

K/A IMPORTANCE: *RO* 3.3 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 8 *55.43(b) SRO*

OBJECTIVE: CVCS-06

LIST power supplies for the major CVCS components as listed in the EDPs.

REFERENCES: EDP-002

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number PATH-1-03 001

JUSTIFICATION:

a. Plausible since the power supply for 'A' CCP is correct, but incorrect power supply listed for 'B' CCP.

b. Plausible since the power supply for 'C' CCP is correct, but incorrect power supply listed for 'B' CCP.

c. Plausible since power supply for both pumps seems logically correct, but incorrect power supply listed for both.

d. **CORRECT** C' CCP will be supplied by 'B' EDG and 'A' CCP will be supplied by DSDG.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of emergency power supplies for charging pumps

REFERENCES SUPPLIED:

9.0 480V-E2

480V-E2 POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: E-1/E-2 ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
23A	**CHARGING PUMP C** CHG-PMP-C	163B	52/23A
23B	**SAFETY INJECTION PUMP C** SI-PMP-C	239	52/23B
23C	**FEED TO MCC-6** MCC-6	1188	52/23C
24A	**SERVICE WATER PUMP C** SW-PMP-C	833	52/24A
24B	**CV RECIRC FAN, HVH-4** HVH-4	514	52/24B
24C	**AUX FEEDWATER PUMP B** AFW-PMP-B	655	52/24C
25A	**CV RECIRC FAN, HVH-3** HVH-3	513	52/25A
25B	**SERVICE WATER PUMP D (NORMAL SUPPLY)** SW-PMP-D	834B	52/25B
25C	**CV SPRAY PUMP B** CV-SPRAY-PMP-B	290	52/25C
26A	**FEED TO MCC-18** MCC-18	1189	52/26A
26B	**RESIDUAL HEAT REMOVAL PUMP B** RHR-PMP-B	216	52/26B
26C	**COMPONENT COOLING WATER PUMP C** CCW-PMP-C	209	52/26C
27A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
27B	**EMERGENCY DIESEL GENERATOR B TO 480V BUS E-2** 480V-E2	895	52/27B

7.0 480V-DS

480V-DS POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: 4160V SWITCHGEAR ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
32A	**FEED TO 480V BUS DS** 480V-DS	1015	52/32A
32B	**DEDICATED SHUTDOWN DIESEL GENERATOR TO 480V BUS DS (ALT POWER)** 480V-DS	1016	52/32B
33A	**CONTROL POWER TRANSFORMER (*)** CPT/480V-DS	N/A	N/A
33B	**SERVICE WATER PUMP D (ALT POWER)** SW-PMP-D	834C	52/33B
33C	**COMPONENT COOLING WATER PUMP A** CCW-PMP-A	201	52/33C
33D	**RESIDUAL HEAT REMOVAL PUMPS (ALT POWER)** RHR-PMP-A, B	1752	52/33D
34A	**POTENTIAL TRANSFORMER** PT/480V-DS	N/A	N/A
34B	**CHARGING PUMP A** CHG-PMP-A	161	52/34B
34C	**FEED TO MCC-5 (ALT POWER)** MCC-5	N/A	52/34C
34D	**FEED TO PP-51** PP-51	N/A	52/34D

* Compartment 33A also contains the Charging Pump A total run time meter and the Component Cooling Water Pump A total run time meter.

Question: 59

Given the following conditions:

- The unit is experiencing a loss of all feedwater event and FRP-H.1, "Response to Loss of Secondary Heat Sink," has been entered.
- **NO** AFW flow is available.
- Containment pressure is 0.4 psig.

Which ONE (1) of the following describes when the operator is required to trip the RCPs and immediately initiate feed and bleed?

a. Five highest core exit TC temperatures are 658 °F, 656 °F, 649 °F, 648 °F, and 645 °F and are all rising

b. RCS hot leg temperatures are 652 °F, 646 °F, and 648 °F and are all rising

c. Pressurizer levels are indicating 93%, 97%, and 94% and are all stable

d. SG wide range levels are 18%, 22%, and 36% and are all stable

Answer:

d. SG wide range levels are 18%, 22%, and 36% and are all stable

QUESTION NUMBER: 59
TIER/GROUP: *RO* 1/2 *SRO*
K/A: WE05EA2.2

Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

K/A IMPORTANCE: *RO* 3.7 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: FRP-H.1-08

Given plant conditions, EVALUATE the appropriate actions to mitigate consequences of a loss of secondary heat sink as directed by steps in FRP-H.1.

REFERENCES: FRP-H.1

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number FRP-H.1-03 012

JUSTIFICATION:

a. Plausible since this would be an indication that heat is not being adequately removed from the RCS, but trigger event is low SG level.

b. Plausible since this would be an indication that heat is not being adequately removed from the RCS, but trigger event is low SG level.

c. Plausible since this would be an indication that heat is not being adequately removed from the RCS, but trigger event is low SG level.

d. CORRECT Any 2 SGs below 26% wide range level requires immediate tripping of the RCPs and initiation of feed and bleed.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of feed and bleed initiation criteria

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Feed flow is not re-established to any faulted S/G if an intact S/G is available.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Total Feed Flow - LESS THAN 300 GPM DUE TO OPERATOR ACTION	Go To Step 3.
2.	Reset SPDS And Return To Procedure And Step In Effect	
* 3.	Determine If Secondary Heat Sink Is Required As Follows:	
	a. Check RCS pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE	a. Reset SPDS and Go To PATH-1, Entry Point C.
	b. Check RCS temperature - GREATER THAN 350°F [310°F]	b. Perform the following: 1) Place RHR System in service using Supplement I. 2) WHEN adequate cooling with RHR is established, THEN reset SPDS and return to procedure and step in effect.
* 4.	Check Any Two S/G Wide Range Levels - LESS THAN 27% [34%]	IF any two S/G Wide Range Levels decrease to less than 27% [34%], THEN Go To Step 5. Go To Step 6.
5.	Perform The Following: a. Stop all RCPs b. Observe CAUTION prior to Step 30 and Go To Step 30	

Question: 60

Given the following conditions:

- A unit trip and safety injection have occurred due to a SGTR on 'A' SG.
- EPP-012, "Post-SGTR Cooldown using Backfill," is being implemented.
- RCS pressure is 940 psig.
- It has been determined that the accumulators should be isolated.
- The breakers for the accumulator discharge valves (SI-865A, B, C) have been closed.
- The 'A' accumulator discharge valve (SI-865A) loses light indication after it is given a closed signal.
- 'B' and 'C' accumulator valves stroke closed as expected.

Which ONE (1) of the following actions should be taken regarding 'A' accumulator?

a. Slow the rate at which the RCS is being depressurized to allow a controlled injection of the accumulator

b. Drain the accumulator to the Reactor Coolant Drain Tank

c. Vent the accumulator to Containment atmosphere

d. Maintain RCS pressure above 800 psig until a Containment entry can be made to locally close the discharge valve

Answer:

c. Vent the accumulator to Containment atmosphere

QUESTION NUMBER: 60
TIER/GROUP: ***RO*** 1/2 ***SRO***
K/A: 038EA1.30

Ability to operate and monitor the following as they apply to a SGTR: Safety injection and containment isolation systems

K/A IMPORTANCE: ***RO*** 4.0 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: EPP-012-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to the Post-SGTR Cooldown using Backfill as directed in EPP-12.

REFERENCES: EPP-12

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EPP-012-08 001

JUSTIFICATION:

a. Plausible since the accumulators are designed to inject into the RCS during an accident, but vented to prevent nitrogen gas injection into the RCS.

b. Plausible since this appears to be a method of lowering pressure in accumulator and it does drain to the RCDT, but should be vented, not drained since some pressure will still remain in the accumulator due to the nitrogen gas.

c. **CORRECT** Vented to prevent nitrogen gas injection into the RCS when the RCS depressurization continues.

d. Plausible since manual isolation would prevent the accumulator from injecting, but would delay the continued cooldown and depressurization. Procedure directs venting.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of actions regarding SI accumulators during EPP implementation

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Determine If SI Accumulators Should Be Isolated:	
	a. Check both of the following conditions exist: • RCS subcooling - GREATER THAN 35°F [55°F] AND • PZR level - GREATER THAN 10% [32%]	a. Go To EPP-17, SGTR With Loss Of Reactor Coolant: Subcooled Recovery.
7.	Isolate SI Accumulators As Follows:	
	a. Locally close the breakers for the following valves: • SI-865C, ACCUMULATOR C DISCHARGE (MCC-5, CMPT 9F) • SI-865A, ACCUMULATOR A DISCHARGE (MCC-5, CMPT 14F) • SI-865B, ACCUMULATOR B DISCHARGE (MCC-6, CMPT 10J)	
	b. Verify all ACCUM DISCHs - CLOSED • SI-865A • SI-865B • SI-865C	b. Vent any unisolated accumulator as follows: 1) Open the appropriate ACCUM VENT Valves: • SI-853A • SI-853B • SI-853C 2) Open HIC-936, ACC VENT HDR FLOW.

EPP-012-08 001

Given the following plant conditions:

- Plant trip and SI have occurred due to a SGTR on "A" SG
- EPP-012, "Post-SGTR Cooldown using Backfill" is in progress
- It has been determined that the accumulators should be isolated. The breakers for the accumulator discharge valves (SI-865A, B, C) have been closed.
- The "A" accumulator discharge valve (SI-865A) loses light indication after it is given a closed signal. "B" and "C" accumulator valves stroke closed as expected.

Which ONE (1) of the following would be the next action?

A. Continue the RCS Cooldown/Depressurization

✓B. Vent the "A" accumulator

C. Vent all accumulators

D. Contact Chemistry to obtain periodic born samples

Question: 76

Given the following plant conditions:

- Following a refueling outage, the unit is being raised to 100% power.
- Reactor Engineering has **NOT** implemented any power ramp rate limitations other than those stated in GP-005, "Power Operation."

Which ONE (1) of the following power changes would violate the power ramp rate limitations identified in GP-005?

a. Raising power from 7% to 14% over a 3-minute period

b. Raising power from 31% to 36.6% over a 1-hour period

c. Raising power from 62% to 65.8% over a 1-hour period

d. Raising power from 93% to 96.2% over a 1-hour period

Answer:

c. Raising power from 62% to 65.8% over a 1-hour period

QUESTION NUMBER: 76
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 001A1.06

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including: Reactor power

K/A IMPORTANCE: ***RO*** 4.1 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: GP-005-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in GP-005 by explaining the basis of each.

REFERENCES: GP-005

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number GP-005-07 003

JUSTIFICATION:

a. Plausible since this is the largest change, but changes of this magnitude are expected and acceptable while synchronizing the generator to the grid.

b. Plausible since this is the largest non-step change (5.6%), but limitations only apply above 50% power.

c. **CORRECT** Power ramp rate limitations are 3.5% per hour between 50% and 100%. This would be a 3.8% change over a 1-hour period.

d. Plausible since this exceeds the previous limitation of 3% per hour (3.2%) and is at the highest given power level, but the limit is 3.5% per hour.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the power ramp rate limitations

REFERENCES SUPPLIED:

5.8 During start-up and loading of the Turbine, S/G water level is very unstable and has a tendency to swell. S/G levels should be maintained from 40% to 50% on narrow range level indication for better control. The wide range and narrow range tend to disagree slightly when there is a transient level condition. Wide range level indication should be used to observe which direction the level is moving. If the narrow range level approaches the High or LO-LO Level trip point, Turbine loading should be stopped until S/G level recovers. Wide swings in Feedwater Regulating Valve positions, in the open or closed direction, should be avoided, as this can cause water level to shrink or swell out of control. Sustained Turbine operation at less than 5% of rated load should be avoided.

5.9 The Feedwater Regulating Valves FCV-478, FCV-488, FCV-498, and Rod Control should be placed in MANUAL when switching Turbine first stage pressure channels. The Feedwater Regulating Valves FCV-478, FCV-488, FCV-498, should be placed in MANUAL when switching steam flow channels, or feedwater flow channels.

5.10 For Turbine startups and scheduled load changes the heatup and loading rates specified in Curves 7.8, 7.9, and 7.10 should be adhered to.

5.11 Power Ramp Rate Limits are restricted after core fuel movement to 3.5%/hr from 50% to 100% power. During subsequent power increases, this ramp limit may apply depending on the maximum power level achieved and length of operation at that power level. (ESR 98-00395)

5.12 The RCS Design Basis Document states that the PZR Spray Valves are designed to prevent PZR pressure from reaching the lift setpoint of the PZR PORVs following a step reduction of 10% of full power under automatic Reactor control during normal plant operations. Normal loading and unloading is 5% of full power per minute. Operability Determination 95-015 Rev 2 identifies that when one PZR Spray Valve is out of service, step changes should be limited to 5% of full power to reduce the potential for challenging the PZR PORVs. (CAPS Project CR 95-02365)

5.13 Exhaust hood temperature should not be allowed to exceed 175 °F with exhaust hood spray out of service. If the temperature cannot be reduced to less than 175°F, the unit should be shutdown and the trouble corrected. The maximum exhaust hood temperature permitted for short periods of time is 250 °F. A Generator Lockout will occur if the exhaust hood temperature is 225 °F for greater for 5 minutes.

GP-005-07 003

Given the following plant conditions:

- Following a refueling outage, the reactor was operated at full power for 118 days
- Then shutdown for 35 days for required maintenance
- A plant startup has been performed to 20% power

What RATE LIMITS, if any, apply to the REACTOR POWER INCREASE from 20% to full power?

A. 5% per minute

✓B. 3% per hour

C. 3% per minute

D. 10% per hour

Question: 77

Given the following conditions:

- The reactor has tripped from 100% power due to a feed line break.
- SI has been actuated.
- AFW pumps are supplying feed to the SGs.
- Immediate operator actions are complete.
- Foldout A has been implemented.
- The Outside AO reports a large leak at the CST.

Which ONE (1) of the following describes the available backup sources to the AFW Pump Suction?

	PREFERRED BACKUP	**ALTERNATE BACKUP**
a.	Service Water	Deepwell Water
b.	Service Water	Fire Water
c.	Deepwell Water	Service Water
d.	Fire Water	Service Water

Answer:

a.	Service Water	Deepwell Water

QUESTION NUMBER: 77
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 061K1.07

Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Emergency water source

K/A IMPORTANCE: ***RO*** 3.6 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: AFW-05

DESCRIBE the performance and design attributes of the major AFW System components.

REFERENCES: EPP-Foldout A
OP-402

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AFW-02 002

JUSTIFICATION:

a. **CORRECT** Both Service Water and Deepwell water can be used with Service Water being the preferred backup source.

b. Plausible since Service Water is the preferred backup source, but Fire Water is not a backup source.

c. Plausible since both Service Water and Deepwell water can be used, but Service Water is the preferred backup source.

d. Plausible since Service Water is a backup source, but Fire Water is not a backup source.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of backup water supplies to AFW

REFERENCES SUPPLIED:

CONTINUOUS USE

FOLDOUT A

(Page 1 of 6)

1. RCP TRIP CRITERIA

   IF BOTH conditions below are met, THEN stop all RCPs:

   - SI Pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW TO THE CORE
   - RCS Subcooling - LESS THAN 35°F [55°F]

2. SI ACTUATION CRITERIA

   IF EITHER condition below occurs, THEN Actuate SI and Go To PATH-1, Entry Point A:

   - RCS Subcooling - LESS THAN 35°F [55°F]
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [32%]

3. SPRAY ACTUATION CRITERIA

   IF a valid CV Spray Signal occurs, THEN dispatch an Operator to the Safeguards Racks to block CV Spray as follows: (A screwdriver is available locally for opening the panels)

   a. At the front of Safeguards Relay Rack 51, rotate Test Switch Number 5 (PC-951A) to the PUSH TO TEST position.

   b. At the front of Safeguards Relay Rack 63, rotate Test Switch Number 5 (PC-951A) to the PUSH TO TEST position.

4. AFW SUPPLY SWITCHOVER CRITERIA

   IF CST level decreases to less than 10%, THEN switch to backup water supply using OP-402, Auxiliary Feedwater System.

5.5 Starting an AFW pump to fill S/Gs when the S/Gs are in Wet Layup on Recirculation could result in overpressurization of the S/G Wet Layup System.

5.6 **IF** HVH-7A is inoperable, **THEN** MDAFW B will be inoperable. **IF** HVH-7B is inoperable, **THEN** MDAFW A will be inoperable.

5.7 Backup water supply valves from both the Service **AND** Deepwell Water are to be normally closed with the telltale valve between each of the two block valves open to insure that no untreated water enters the plant cycle during normal operation. **IF** backup water is required, **THEN** Service Water is the primary backup and Deepwell water a secondary backup.

5.8 When using the deepwell pumps as a backup supply to the AFW Pumps, maximum **TOTAL** allowable feed rates for various combinations of AFW Pumps vs. deepwell pumps in operation have been established. The flow rates are based on 200 gpm per deepwell pump **AND** assumes 90 gpm seal leakoff flow, 165 gpm recirc flow for the SDAFW Pump, **AND** 60 gpm recirc flow for each MDAFW Pump. These flow rates are to prevent runout **AND** possible damage to the deepwell pumps.

5.9 **IF** the CST level decreases to 10% during AFW operation, **THEN** a backup water supply should be placed in service. Service Water should be used as first backup supply to AFW Pumps. **IF** Service Water is not available, **THEN** Deepwell Water should be used as backup to AFW Pumps.

5.10 The proper sequence to follow when securing a MDAFW Pump is, first, stop the pump, allow it to stop rotating, then close the motor operated discharge valves (V2-16A, V2-16B, V2-16C). This sequence will allow proper seating of the check valves **AND** allow the discharge valves to fully seat which prevents back leakage through all these valves.

5.11 A possible consequence of check valve or discharge valve backleakage is steam binding of the AFW Pumps. Steam binding of the MDAFW Pumps may be indicated by warm discharge piping between the discharge check valve(s) **AND** the V2-16(s). Steam binding of the SDAFW Pump may be indicated by a warm pump casing. **IF** steam binding of any of the AFW Pumps is suspected, **THEN** refer to the Infrequent Operation section of this procedure.

5.12 The Condensate Storage Tank should be maintained full to provide a maximum available water supply.

5.13 **IF** the CST level is decreasing , **THEN** prior to reaching 34%, the reliability of the MDAFW Pumps **AND** power supplies shall be evaluated. Starting the SDAFW Pump to ensure its availability should be considered. The SDAFW Pump should be used to feed the S/Gs but can be operated on recirculation.

AFW-02 002

Given the following plant conditions:

- The reactor has tripped from 100% power due to a feed line break
- SI has been actuated
- AFW pumps are supplying feed to the S/G's
- Immediate operator actions are complete
- Foldout A has been implemented
- The Outside AO reports a large leak at the CST

Which ONE (1) of the following describes the available supply sources to the AFW Pump Suction?

A. CST, Hotwell

B. Circulating Water, Service Water

C. Deepwell Water, Fire Water

✓D. Deepwell Water, Service Water

Question: 78

Given the following conditions:

- A turbine runback has occurred from 100% to 70% power.
- RCS Tavg is 567 °F.
- PZR Pressure is 2265 psig.
- PZR Level is 51%.

Which ONE (1) of the following describes the expected condition of the proportional heaters and pressurizer spray valves?

	PROPORTIONAL HEATERS	**SPRAY VALVES**
a.	On	Open
b.	On	Closed
c.	Off	Open
d.	Off	Closed

Answer:

a.	On	Open

QUESTION NUMBER: 78

TIER/GROUP: *RO* 1/1 *SRO*

K/A: 027AK2.03

Knowledge of the interrelations between the Pressurizer Pressure Control Controllers and positioners

K/A IMPORTANCE: *RO* 2.6 *SRO*

10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: PZR-09

EXPLAIN the normal operation of the PZR and PRT control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-059

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number AOP-019-08 002

JUSTIFICATION:

a. **CORRECT** Heaters are on due to level being more than 5% above program and sprays are open due to a high deviation signal of more than 25 psid.

b. Plausible since heaters are on due to being more than 5% above program, but sprays should be open due to a high deviation signal of more than 25 psid.

c. Plausible since sprays are open due to a high deviation signal of more than 25 psid, but heaters should be on due to being more than 5% above program.

d. Plausible since heaters would normally be expected to be off due to the high pressure condition, but should be on due to level deviation.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of effect of conflicting conditions for heater and spray operation

REFERENCES SUPPLIED:

5.1.6 PZR Level Control Setpoints

1.	Level program as function of T_{avg} (TM-459)	
	for T_{avg} 547°F	22.2% of level span
	for T_{avg} 575.4°F	53.3% of level span
	(Program is linear from 547°F to 575.4°F)	
	Low limit	22.2% of level span
	High limit	53.3% of level span
2.	Low-Low Level Heater Cutout (LC-459C, LC-460C)	14.4.% of level span
3.	Level Controller (LC-459F)	
	Proportional gain	10% charging pump speed/% level deviation
	Reset time constant	430 seconds
4.	Letdown Valve Isolation	14.4% of level span
5.	Back-up Heaters on	+5% of programmed level

6.0 SYSTEM OPERATION

6.1 Normal Operation

Insurge of RCS Coolant - produced by increase in T_{avg}. An insurge of coolant will reduce volume of the steam bubble causing an increase in the temperature and pressure of the steam. The steam space or bubble becomes superheated and some minor condensation occurs at surface and on walls.

The increased pressure causes the spray valve to open which cools and condenses a part of the steam bubble, thereby reducing pressure.

The increase in level will energize backup heaters if the level increases to 5% above program.

Outsurge of RCS Coolant

An outsurge of RCS coolant will increase the volume of the steam bubble, which will cause water to flash to steam, limiting the pressure decrease.

Proportional heaters will be full on to limit pressure decrease. If pressure decrease is

$$\frac{2235 - 1700}{800} * 10 = 6.69 \text{ on the 10 turn pot.}$$

The output of PC-444J (setpoint signal) is sent to PC-444A to be compared to the actual pressure. PC-444A has a gain of 2 which effectively cuts in half the range of control of PZR pressure to 400 psi around the setpoint determined by PC-444J. The controller output is then directed to the proportional heaters, spray valves via controllers PC-444C and PC-444D, backup heaters, PZR PORV 456 and PI-458 and is displayed on the meter on PC-444J

The components operated by PC-444A operate at fixed deviation from setpoint or controller output as observed on the meter on PC-444J, no matter what setpoint is dialed in on PC-444J. For example the backup heaters are set to turn on 25 psi below set pressure. If set pressure is 2235 psig, their setpoint would be 2210 psig and the control output when they came on would be as follows:

$$\frac{2210\text{-}2035}{400} = .4375 \text{ or } 43.75\% \text{ demand}$$

If the pot on PC-444J were then set at 6.25 this would give a set pressure of 2200 psig. When the output of PC-444A was at 43.75% the backup heaters would come on, pressure would be 2175 psig; 25 psi below set pressure. The setpoints normally listed for heater, spray, and PCV-456 setpoints are based on a set pressure of 2235 psig where PC-444J is normally set.

As stated before, PC-444A is a Proportional + Integral controller, therefore controller output may not correspond exactly to the pressure monitored by operator. If pressure is away from setpoint for an extended period of time the controller output may saturate while increasing its output trying to return pressure to setpoint.

5.1.2 PZR Pressure Control Setpoint (PZR-Figure 10)

1.	PZR Pressure Controller (PC-444A)	
	Proportional gain	2
	Reset time constant	12 sec
	Rate time constant	off
	Pressure set point, Pref	2235 psig
2.	Spray Valve Controllers (PC-444C, PC-444D)	
	Proportional gain in % spray valve Lift per psi	2%/psi

	Set point where spray is initiated on compensated pressure signal from PC-444A	+25 psi (2260 psig)
	Setpoint where spray is full open	+75 (2310 psig)
3.	Variable Heater Controller	
	Proportional gain in % heating power	-3.33%/psi per psi
	Set point where proportional heating is full on, on signal from PC-444A	-15 psi (2220 psig)
	Setpoint where proportional heating is full off	+15 psi (2250 psig)
4.	Power Relief Valve, PCV-455C operating on compensated pressure signal from PC-444A to PC-444B	+100 (2335 psig)
5.	Back-up heater turned on, on compensated pressure signal from PC-444A to PC-444F	-25 psi (2210 psig)
	Back-up heaters turned off	-15 psi (2220 psig)
6.	Power Relief Valve (PCV-456) operated on actual pressure (PC-445A)	2335 psig

5.1.3 PZR PORV Control (PZR-Figure 8 & PZR-Figure 13)

The PZR PORVs have two modes of control, Normal and Low Temperature Overpressure Protection (LTOPP). In normal mode the PORVs have a permissive of 2000 psig to open in Automatic. This "permissive" is supplied by the protection channels meeting a 2/3 logic . As stated before PCV-456 receives its signal from PT-445 set at 2335 psig and PCV-455C receives its signal from PC-444A at +100 psi which is nominally 2335 psig also. When the key switch for OVERPRESSURE PROTECTION on the RTGB is place in the LOW PRESSURE position (one switch for each PORV) the input to each PORV is shifted to the LTOPP controller.

5.1.4 Low Temperature Overpressure Protection Control (LTOPP) (PZR-Figure 13)

LTOPP control is required to be activated when the RCS is cooled down below 360°F to minimize Pressurized Thermal Shock (PTS) concerns. The LTOPP controller uses the lowest of TE-410, TE-420 and TE-430 to determine RCS temperature and pressure as sensed by PT-500 and PT-501. The lift setpoint is variable based upon auctioneered low RCS temperature. At an RCS temperature of 350°F, the pressure setpoint is 400 psig. The setpoint of the Comparators PC502 and PC503 are increased as RCS temperature is increased. The setpoint will not decrease below 400 psig.

AOP-019-08 002

The unit is at power and a transient occurred. RCS pressure is below normal, the crew has implemented AOP-019, Malfunction Of RCS Pressure Control. They are at the point in the procedure that asks if the master pressure controller PC-444J is operating properly in AUTO. What is the correct response for the master pressure controller PC-444J based on current plant conditions?

A. heaters off and sprays open

B. heaters off and sprays shut

C. heaters on and sprays open

✓D. heaters on and sprays shut

Question: 79

Following an accident, FRP-C.2, "Response to Degraded Core Cooling," is being implemented.

After the performance of several steps in FRP-C.2, the following Critical Safety Function Status Tree (CSFST) conditions are noted:

- Integrity - RED
- Core Cooling - RED
- Containment - ORANGE
- Heat Sink - YELLOW
- Subcriticality - YELLOW
- Inventory - YELLOW

Which ONE (1) of the following describes which action should be taken by the CRSS?

a. Remain in FRP-C.2, "Response to Degraded Core Cooling," until completion and then recheck the CSFSTs

b. Transition to FRP-C.1, "Response to Inadequate Core Cooling" due to the RED condition on Core Cooling

c. Transition to FRP-P.1, "Response to Imminent Pressurized Thermal Shock," due to the RED condition on Integrity

d. Transition to FRP-J.1, "Response to High Containment Pressure," due to the ORANGE condition on Containment

Answer:

b. Transition to FRP-C.1, "Response to Inadequate Core Cooling" due to the RED condition on Core Cooling

QUESTION NUMBER: 79
TIER/GROUP: ***RO*** 3 ***SRO***
K/A: 2.4.22

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

K/A IMPORTANCE:	***RO***	3.0	***SRO***	
10CFR55 CONTENT:	***55.41(b) RO***	10	***55.43(b) SRO***	

OBJECTIVE: OMM-022-09

DETERMINE the different flowpaths generated by OMM-022.

REFERENCES: OMM-022

SOURCE: ***New*** ☒ ***Significantly Modified*** ☐ ***Direct*** ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible since FRP-C.2 is being performed in response to an ORANGE path on Core Cooling, but FRP-C.1 has additional actions which will need to be performed in response to the worsening condition.

b. CORRECT The highest RED path should be addressed first and Core Cooling has a higher priority than Integrity.

c. Plausible since Integrity is a RED path, but Core Cooling has a higher priority.

d. Plausible if a misconception exists that ORANGE paths are a higher priority, but the highest priority are RED paths.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Evaluation of CSFST to determine highest priority

REFERENCES SUPPLIED:

8.2.6 (Continued)

3. Monitoring of the Critical Safety Function Status Trees takes place in accordance with its own rules of usage, in parallel with the recovery actions being performed by the Operator. The monitoring may be done directly by one of the Operators in the control room or by some other member of the shift assigned to the control room (normally the STA).

4. Monitoring is accomplished by use of the Safety Parameter Display System (SPDS) feature of the ERFIS computer or by use of the manual status boards. The CRSS is the designated primary SPDS user while the Shift Technical Advisor is available to assist the CRSS as the secondary SPDS user.

5. Status Trees ask a series of questions about plant conditions, and in general, each question asked depends on the answer to the previous question. This dependency results in a branching pattern, which is referred to as a "tree."

6. There are six different trees, each one evaluating a separate safety aspect (Critical Safety Function) of the plant. At any given time, a Critical Safety Function status is represented by a single Path through its tree. Since each Path is unique, it is uniquely labeled at its end point, or terminus. This labeling consists of color-coding of the terminus and possible transition to an appropriate FRP, if required by that safety status. If the status is normal for a particular Critical Safety Function, no transition is specified, and the condition is clarified by the words CSF-SAT.

7. Color-coding can be either RED, ORANGE, YELLOW, or GREEN, with GREEN representing a "SAT" safety status. Each non-green color represents an action level that should be addressed according to the Rules of Priority for Status Tree Use.

8. Several special conditions also affect the CSFSTs indicated by ERFIS:

   - All CSFSTs are forced to a GREEN-condition when the plant mode is Cold Shutdown.

   - The Heat Sink Tree is forced to a GREEN-condition when the plant is less than 350°F.

8.2.6.8 (Continued)

- The Subcriticality Tree is forced to a GREEN-condition when the plant mode is Power Operation or Hot Shutdown except:
  - When in the Power Operation mode, the actual Critical Safety Function Status will be displayed if the Reactor Trip and Bypass Breakers are open.
  - When in any mode, the actual Critical Safety Function Status will be displayed if a trip condition exists (as determined from Reactor Protection System inputs).

9. The six Status Trees are always evaluated in the following sequence (order of priority):

   1) Subcriticality (S)

   2) Core Cooling (C)

   3) Heat Sink (H)

   4) Integrity (P)

   5) Containment (J)

   6) Inventory (I)

10. If identical color priorities are found on different trees during monitoring, the required action priority is determined by the above sequence. For example, a RED-condition on the Subcriticality Tree takes priority over a RED-condition on Core Cooling Tree.

11. The user begins monitoring with the Subcriticality Tree. Questions are answered based on plant conditions at the time, and the appropriate branch line followed to the next question. An individual Status Tree evaluation is complete when the user arrives at a color-coded terminus. With the exceptions noted below, the color and instructions of the terminus are noted and the user continues to the next tree in sequence.

    a. If any RED terminus is encountered, the operator is required to immediately stop any Path or EPP in progress, and to perform the Function Restoration Procedure (FRP) required by the terminus.

8.2.6 (Continued)

15. Following FRP implementation, a YELLOW-condition might indicate a residual off-normal condition. The Operator is allowed to decide whether or not to implement any YELLOW-condition FRP.

16. When using the SPDS to monitor the CSFSTs, the "SPDS Reset" feature must be used prior to initiating CSFST monitoring. SPDS software "locks in" the highest priority condition occurring during the transient, regardless of whether or not the condition is still present.

17. The only requirement of the monitoring function is that the CRSS in charge of recovery actions be immediately informed of RED or ORANGE-conditions, and regularly advised of YELLOW or GREEN-conditions.

18. The Path or EPP actions in progress are suspended if either a RED or ORANGE-condition is detected on a Status Tree. Path or EPP actions are not to be performed while a Critical Safety Function is being restored from a RED or ORANGE-condition, unless required by the FRP in effect. Conversely, in a few cases, the FRPs are not performed while certain EPPs are in effect. These cases will be explicitly noted in the EPP.

19. After restoration of any Critical Safety Function from a RED or ORANGE-condition, recovery actions may continue when the FRP is complete. Most often, the FRP will return the Operator to the suspended Path or EPP. At times, an FRP will require a transition to a different Path or EPP because of conditions created within the FRP.

20. Upon continuation of recovery actions, some judgement is required by the Operator to avoid inadvertent reinstatement of a RED or ORANGE- condition by undoing some critical step in a Function Restoration Procedure. The plant recovery procedures are optimal in assuming that equipment is available when required. The appearance of a RED or ORANGE-condition in most cases implies that some equipment or function required for safety is not available, and some adjustment may be required in the recovery procedures.

Question: 80

Given the following conditions:

- A reactor trip has occurred from 100% power.
- All SGs levels indicate 6%.

Upon initiation of AFW, which ONE (1) of the following correctly describes the automatic response of the AFW system to these conditions?

a. The normally closed MDAFW pump discharge flow control valves (FCV 1424 and 1425) fully open

b. The normally open SDAFW pump discharge flow control valve (FCV 6416) throttles closed

c. The normally closed SDAFW pump discharge flow control valve (FCV 6416) throttles open

d. The normally open MDAFW pump discharge flow control valves (FCV 1424 and 1425) throttle closed

Answer:

b. The normally open SDAFW pump discharge flow control valve (FCV 6416) throttles closed

QUESTION NUMBER: 80
TIER/GROUP: ***RO*** 2/1 ***SRO***
K/A: 061A3.03

Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start

K/A IMPORTANCE: ***RO*** 3.9 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: AFW-09

EXPLAIN the normal operation of the AFW control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-042

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AFW-10 023

JUSTIFICATION:

a. Plausible since FCV-1424 & 1425 are normally closed, but the valves do not fully open. When the pumps are started, the discharge flow control loops are energized and the valves throttle to maintain flow rate at the setpoint.

b. **CORRECT** The normally open valve FCV-6416 will throttle to maintain desired flow on a pump start.

c. Plausible since FCV-6416 does throttle to maintain desired flow, but it is normally open.

d. Plausible since FCV-1424 and FCV-1425 do throttle to maintain desired flow, but the valves are normally closed.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of AFW valve operation on startup

REFERENCES SUPPLIED:

3.3 Control Valves

The AFW system contains an automatic electrohydraulic flow control valve for each MDAFW pump and the SDAFW pump. Each valve's controller is located in the control room and all other components are located in proximity to their respective pump. These valves and their associated controls are used to set AFW pump discharge flowrate and automatically maintain the rate as S/G pressure varies. These valve provide flow control for the AFW system and NPSH (anti-cavitation) protection for the pumps.

These control valves have electrohydraulic actuators which can be automatically positioned based upon the respective AFW pump discharge flow.

A local manual operator is provided for operating the control valve in the event that the control system fails. The control system can also be operated in manual from the RTGB. If RCS temperature is greater than 350°F, then using manual places the plant in a Tech Specs LCO action statement (3.7.4).

MDAFW pump discharge flow control valves (1424 and 1425) control the flow from each MDAFW pump to the S/Gs. These normally closed valves begin to open when the MDAFW pumps are started. The valves "fail-closed" on loss of electric power or loss of control signal. FCV-1424 is powered from IB#2 and FCV-1425 from IB#3. In modes 1, 2 and 3 each control valve is normally in AUTO and set at 325 gpm. When the RCS temperature is ≤ 350°F, these controllers shall be in Auto and set to a flowrate of 100 gpm.

SDAFW pump discharge flow control valve (6416) controls the flow from the SDAFW pump to the S/Gs. This normally open valve begins to adjust when the SDAFW pump is started. This valve will "fail-open" on a loss of electrical power or loss of the control signal. FCV-6416 is powered from LP-26. In modes 1, 2 and 3, FCV-6416 is normally in AUTO and set at 500 gpm.

4.0 INSTRUMENTATION

4.1 AFW (AFW) Flow Indication System

There are three dual flow edge meters - 0-500 gpm, one per S/G for the MDAFW pumps and SDAFW pump, located on RTGB.

S/G 1 Aux. Feedwater Flow (Motor Driven)	FI-1425A
S/G 1 Aux. Feedwater Flow (Steam Driven)	FI-1426A
S/G 2 Aux. Feedwater Flow (Motor Driven)	FI-1425B
S/G 2 Aux. Feedwater Flow (Steam Driven)	FI-1426B
S/G 3 Aux. Feedwater Flow (Motor Driven)	FI-1425C
S/G 3 Aux. Feedwater Flow (Steam Driven)	FI-1426C

Question: 96

Given the following conditions:

- The unit is operating at 30% power.
- A dropped control rod has just been re-aligned.
- While attempting to reset the Rod Control Urgent Failure alarm, the operator inadvertently pushes the Rod Control STARTUP button.

Which ONE (1) of the following describes the effect of operating the incorrect button?

a. All Control Bank control rods drop into the core, causing an automatic reactor trip

b. All rods, including Control Bank and Shutdown Bank rods, drop into the core, causing an automatic reactor trip

c. All rods remain in their current position and there is **NO** effect on the Rod Control System circuitry

d. All rods remain in their current position, but the Rod Control System circuitry senses all rods are fully inserted

Answer:

d. All rods remain in their current position, but the Rod Control System circuitry senses all rods are fully inserted

QUESTION NUMBER: 96
TIER/GROUP: *RO* 2/1 *SRO*
K/A: 001K6.11

Knowledge of the effect of a loss or malfunction on the Location and operation of CRDS fault detection (trouble alarms) and reset system, including rod control annunciator

K/A IMPORTANCE: *RO* 2.9 *SRO*
10CFR55 CONTENT: *55.41(b) RO* 6 *55.43(b) SRO*

OBJECTIVE: RDCNT-07

EXPLAIN the purpose and location of the Rod Control System controls and indications.

REFERENCES: SD-007

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number HNP-RO-2000 24

JUSTIFICATION:

a. Plausible since improper operation of correct button could result in rods dropping into core, but operated button only resets starting points for rod control circuitry.

b. Plausible since improper operation of correct button could result in rods dropping into core, but operated button only resets starting points for rod control circuitry.

c. Plausible if misconception that effect is nothing if performed at power since button is normally only operated prior to withdrawing any rods, but operated button resets starting points for rod control circuitry.

d. **CORRECT** Operating button at power does not affect actual rod position, but resets rod control such that circuitry senses rods are at "full inserted" position.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of the function of rod control system controls

REFERENCES SUPPLIED:

selected in the manual mode.

CONTROL BANK B
Control Bank B rods can be moved manually using the IN-HOLD-OUT lever. The bank overlap program is overridden. The rod speed signal is the same as that selected in the manual mode.

CONTROL BANK C
Control Bank C rods can be moved manually using the IN-HOLD-OUT lever. The bank overlap program is overridden. The rod speed signal is the same as that selected in the manual mode.

CONTROL BANK D
Control Bank D rods can be moved manually using the IN-HOLD-OUT lever. The bank overlap program is overridden. The rod speed signal is the same as that selected in the manual mode.

5.1.3 Start-Up Pushbutton

This pushbutton, mounted on the control board, is used to reset the following equipment prior to plant start-up:

- All step counters on the RTGB.
- The master cycler reversible counter.
- All slave cycler counters.
- The bank overlap counter.
- All internal memory and alarm circuits.
- The pulse to analog converter in the IRPI System.

5.1.4 Alarm Reset Pushbutton

This pushbutton, mounted on the RTGB, is used to reset Rod Control System urgent alarms. Rod control alarms displayed on the plant annunciator are cleared by the annunciator system reset pushbutton.

5.1.5 Auto Rod Defeat Pushbutton

Prevents auto rod movement when moving ROD BANK SELECTOR SWITCH through the AUTO position.

5.1.6 Lift Coil Disconnect Switches

A lift coil disconnect switch is furnished for each control and shutdown CRDM. These switches, located on a panel at the rear of the RTGB, are used in retrieving a dropped or

Question: 97

Service Water Pump "D" is capable of being powered from which ONE (1) of the following power sources?

a. **ONLY** 480 VAC Bus E-1

b. **ONLY** 480 VAC Bus E-2

c. Either 480 VAC Bus E-1 OR 480 VAC DS Bus

d. Either 480 VAC Bus E-2 OR 480 VAC DS Bus

Answer:

d. Either 480 VAC Bus E-2 OR 480 VAC DS Bus

QUESTION NUMBER: 97
TIER/GROUP: ***RO*** 2/3 ***SRO***
K/A: 076K2.01

Knowledge of bus power supplies to the following: Service water

K/A IMPORTANCE: ***RO*** 2.7 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: SW-06

LIST power supplies for the major SERVICE WATER System components as listed in the EDPs.

REFERENCES: EDP-002

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number SW-06 005

JUSTIFICATION:

a. Plausible if misconception regarding power supply, but normal supply is E-2 and alternate supply is DS Bus.

b. Plausible since normal supply is E-2, but can also be powered by alternate supply of DS Bus.

c. Plausible since alternate supply is DS Bus, but normal supply is E-2.

d. **CORRECT** Normal supply to SW Pump D is E-2 and alternate supply is DS Bus.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of power supplies to SW Pump

REFERENCES SUPPLIED:

9.0 480V-E2

480V-E2 POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: E-1/E-2 ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
23A	**CHARGING PUMP C** CHG-PMP-C	163B	52/23A
23B	**SAFETY INJECTION PUMP C** SI-PMP-C	239	52/23B
23C	**FEED TO MCC-6** MCC-6	1188	52/23C
24A	**SERVICE WATER PUMP C** SW-PMP-C	833	52/24A
24B	**CV RECIRC FAN, HVH-4** HVH-4	514	52/24B
24C	**AUX FEEDWATER PUMP B** AFW-PMP-B	655	52/24C
25A	**CV RECIRC FAN, HVH-3** HVH-3	513	52/25A
25B	**SERVICE WATER PUMP D (NORMAL SUPPLY)** SW-PMP-D	834B	52/25B
25C	**CV SPRAY PUMP B** CV-SPRAY-PMP-B	290	52/25C
26A	**FEED TO MCC-18** MCC-18	1189	52/26A
26B	**RESIDUAL HEAT REMOVAL PUMP B** RHR-PMP-B	216	52/26B
26C	**COMPONENT COOLING WATER PUMP C** CCW-PMP-C	209	52/26C
27A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
27B	**EMERGENCY DIESEL GENERATOR B TO 480V BUS E-2** 480V-E2	895	52/27B

7.0 480V-DS

480V-DS POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: 4160V SWITCHGEAR ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
32A	**FEED TO 480V BUS DS** 480V-DS	1015	52/32A
32B	**DEDICATED SHUTDOWN DIESEL GENERATOR TO 480V BUS DS (ALT POWER)** 480V-DS	1016	52/32B
33A	**CONTROL POWER TRANSFORMER (*)** CPT/480V-DS	N/A	N/A
33B	**SERVICE WATER PUMP D (ALT POWER)** SW-PMP-D	834C	52/33B
33C	**COMPONENT COOLING WATER PUMP A** CCW-PMP-A	201	52/33C
33D	**RESIDUAL HEAT REMOVAL PUMPS (ALT POWER)** RHR-PMP-A, B	1752	52/33D
34A	**POTENTIAL TRANSFORMER** PT/480V-DS	N/A	N/A
34B	**CHARGING PUMP A** CHG-PMP-A	161	52/34B
34C	**FEED TO MCC-5 (ALT POWER)** MCC-5	N/A	52/34C
34D	**FEED TO PP-51** PP-51	N/A	52/34D

* Compartment 33A also contains the Charging Pump A total run time meter and the Component Cooling Water Pump A total run time meter.

Question: 98

Given the following conditions:

- A plant cooldown is in progress in accordance with GP-007, "Plant Cooldown From Hot Shutdown to Cold Shutdown."
- RCS Pressure is 1500 psig.
- RCS Tavg is 515°F.
- A RCS leak is identified inside containment.

Which ONE (1) of the following identifies the valid signals that could result in a Containment Ventilation Isolation under these conditions?

a.
- Hi Steamline ΔP
- Alarm on R-12, Containment Noble Gas Monitor

b.
- Low Pressurizer Pressure Safety Injection
- Alarm on R-14C, Plant Effluent Noble Gas Monitor

c.
- Manual actuation of Containment Isolation Phase A
- Alarm on R-12, Containment Noble Gas Monitor

d.
- Manual actuation of Containment Isolation Phase A
- Alarm on R-14C, Plant Effluent Noble Gas Monitor

Answer:

c.
- Manual actuation of Containment Isolation Phase A
- Alarm on R-12, Containment Noble Gas Monitor

QUESTION NUMBER: 98
TIER/GROUP: ***RO*** 2/2 ***SRO***
K/A: 029K4.03

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Automatic purge isolation

K/A IMPORTANCE: ***RO*** 3.2 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 9 ***55.43(b) SRO***

OBJECTIVE: ESF-09

EXPLAIN the normal operation of the ESFAS control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES:
TS Table 3.3.2-1
TS Table 3.3.6-1
SD-006

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number ESF-04 006

JUSTIFICATION:

a. Plausible since these would both cause a CVI if the SI signals were not blocked, but under these conditions the steamline differential pressure will not cause a CVI.

b. Plausible since the low pressure would cause a CVI if the SI signals were not blocked, but under these conditions the the low pressure will not cause a CVI.

c. **CORRECT** CVI is caused by manual actuation (same actuation as Phase A), containment radiation (gaseous and particulate), or safety injection. The SI blocks initiated by GP-007 include all signals except manual and high containment pressure.

d. Plausible since manual actuation will cause a CVI, but R-14C only isolates any waste gas release.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of when ESF signals causing a CVI are blocked

REFERENCES SUPPLIED:

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1,2,3,4, (a),(b),(c)	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	1,2,3,4, (a),(b),(c)	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Radiation				
a. Gaseous	(a),(b),(c)	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	(d)
b. Particulate	(a),(b),(c)	1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	(d)
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Functions 1.a-f, for all initiation functions and requirements.			

(a) During CORE ALTERATIONS.
(b) During movement of irradiated fuel assemblies within containment.
(c) During Purging.
(d) Trip Setpoint shall be in accordance with the methodology in the Offsite Dose Calculation Manual.

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure – High	1,2,3,4	3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 4.45 psig	4 psig
d. Pressurizer Pressure – Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1709.89 psig	1715 psig
e. Steam Line High Differential Pressure Between Steam Header and Steam Lines	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 108.95 psig	100 psig
f. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} – Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	543°F
g. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure – Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	614 psig

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.
(a) Above the Pressurizer Pressure interlock.
(b) Above the T_{avg}-Low interlock.
(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.
(d) A function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.

NOTE: "A" Component Cooling Pump will start anytime on low pressure if power is available. "A" pump is on the DS Bus.

6.4 Actions That Can Be Initiated By Other Signals6.4 Actions That Can Be Initiated By Other Signals

6.4.1 Steam Line Isolation6.4.1 Steam Line Isolation

As previously noted, a spray actuation (P-signal) will close all three main steam isolation valves. This action will also occur if there is a high steam line flow coincident with low steam line pressure or low Tavg. (Does not occur on manual spray actuation.)

The main steam isolation valves can be shut individually from the RTGB by their control switch or by the steam line isolation pushbuttons.

Additionally an automatic main steam line isolation actuation will provide a signal to the Safety Injection initiation logic and a safety injection will occur unless it has been blocked.

6.4.2 Feedwater Isolation6.4.2 Feedwater Isolation

As previously noted, an SI actuation will cause a complete feedwater isolation. A reactor trip with median Tavg (TC-408K) less than 554°F will shut the main feedwater regulating valves. A high-high steam generator level (2/3 @ 75%) will shut its respective main feedwater regulating valve, bypass valve, trip both main feedwater pumps, and trip the turbine.

The Feedwater isolation signal must be reset manually if it was caused by an SI signal or steam generator high-high level before normal operation can resume. There is one key operated override/reset switch on the RTGB for each feed line.

6.4.3 Phase "A" Containment isolation and Isolation Valve Seal Water System actuation and Containment Ventilation isolation.6.4.3 Phase A Containment isolation and Isolation Valve Seal Water System actuation and Containment Ventilation isolation

In addition to being actuated by an SI signal, it can be actuated by depressing one of the two manual pushbuttons.

6.4.4 Containment Ventilation Isolation

As previously noted a Containment ventilation isolation will occur upon receipt of an

S-signal or P-signal. It will also be initiated if radiation monitor R-11 or R-12 alarms.

6.4.5 Control Room Ventilation Emergency Pressurization Mode6.4.5 Control Room Ventilation Emergency Pressurization Mode

As previously noted this action will occur upon receipt of an SI signal. This action will also be initiated automatically on an alarm of the area monitor for the control room (R-1). The operator has the ability to shift control room ventilation to the emergency pressurization mode with the normal control switch.

6.5 Blocks6.5 Blocks

Some of the SI signals can be blocked manually from the RTGB when the plant is being intentionally cooled down. There are also instances specified in OMM-022 that specify other times when blocking SI is acceptable.

6.5.1 Low pressurizer pressure and high steam line differential pressure6.5.1 Low pressurizer pressure and high steam line differential pressure

Low pressurizer pressure and high steam line differential pressure can be blocked (provided pressurizer pressure is <2000 psig on 2/3 channels) and unblocked using a three position (BLOCK, unmarked (mid position), UNBLOCK) switch located on the RTGB. These SI initiation signals are normally blocked during a plant cooldown when pressurizer pressure is less than 1950 psig. These signals will be automatically unblocked when pressurizer pressure is increased to 2000 psig. These signals can also be unblocked with a switch on the RTGB. Before these signals are manually or automatically unblocked, the operator should check to see if the bistables for these signals are cleared.

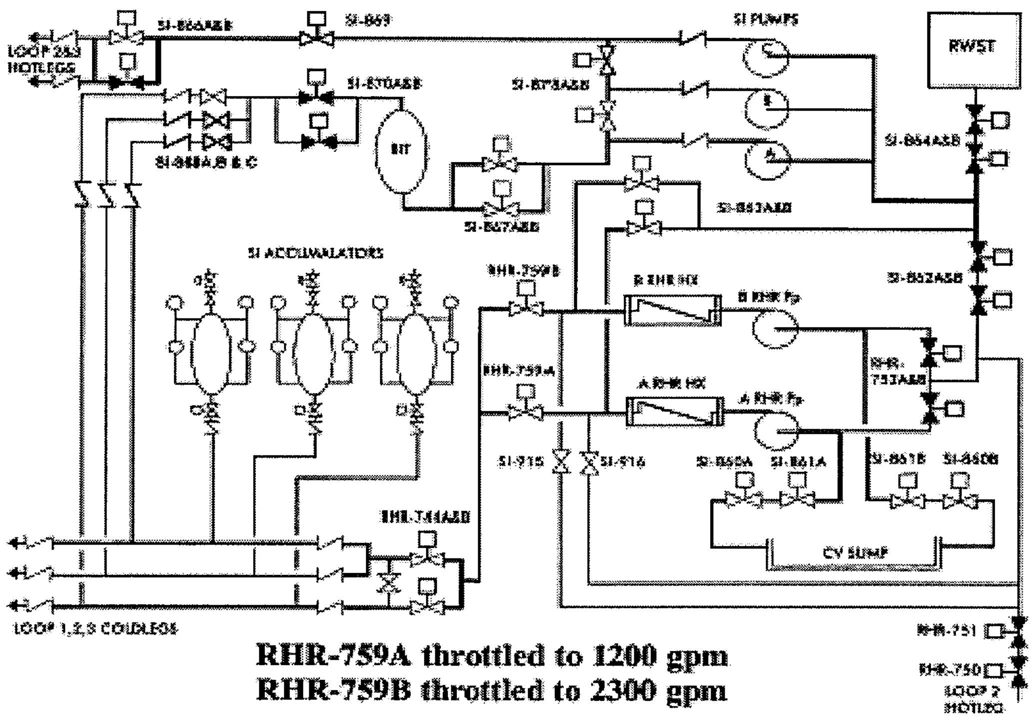
6.5.2 High steam line flow coincident with low steam line pressure or low T_{avg}6.5.2 High steam line flow coincident with low steam line pressure or low Taverage

High steam line flow coincident with low steam line pressure or low T_{avg} and the Hi-Hi CV pressure SI signal can be blocked (provided that Tavg is <543°F on 2/3 channels) and unblocked using a three position (BLOCK, unmarked (mid position), UNBLOCK) switch on the RTGB. This signal is automatically unblocked when T_{avg} reaches 543°F or can be manually unblocked with the switch on the RTGB. Before these signals are manually or automatically unblocked, the operator should check to see if the bistables for these signals are cleared.

7.0 TECHNICAL SPECIFICATIONS7.0 TECHNICAL SPECIFICATIONS

Question: 99

Given the following drawing containing an ECCS alignment:



RHR-759A throttled to 1200 gpm
RHR-759B throttled to 2300 gpm

Which ONE (1) of the following describes the ECCS alignment?

a. Cold leg injection

b. Cold leg recirculation

c. Hot leg injection

d. Long term recirculation

Answer:

d. Long term recirculation

QUESTION NUMBER: 99
TIER/GROUP: ***RO*** 2/2 ***SRO***
K/A: 006A3.06

Ability to monitor automatic operation of the Valve lineups

K/A IMPORTANCE: ***RO*** 3.9 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: RHR-03

DESCRIBE the major flow path through the RHR Systems.

REFERENCES: SD-002
EPP-010

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number RHR-03 007

JUSTIFICATION:

a. Plausible since flow is going to the cold legs, but suction is from sumps instead of RWST.

b. Plausible since flow is going to the cold legs, but additional flow from SI pumps is going to hot legs.

c. Plausible since flow is going to the hot legs, but suction is from sumps instead of RWST.

d. **CORRECT** RHR pumps are taking a suction from sump, providing flow to the cold legs, and providing a suction source to the SI pumps which are providing flow to the hot legs.

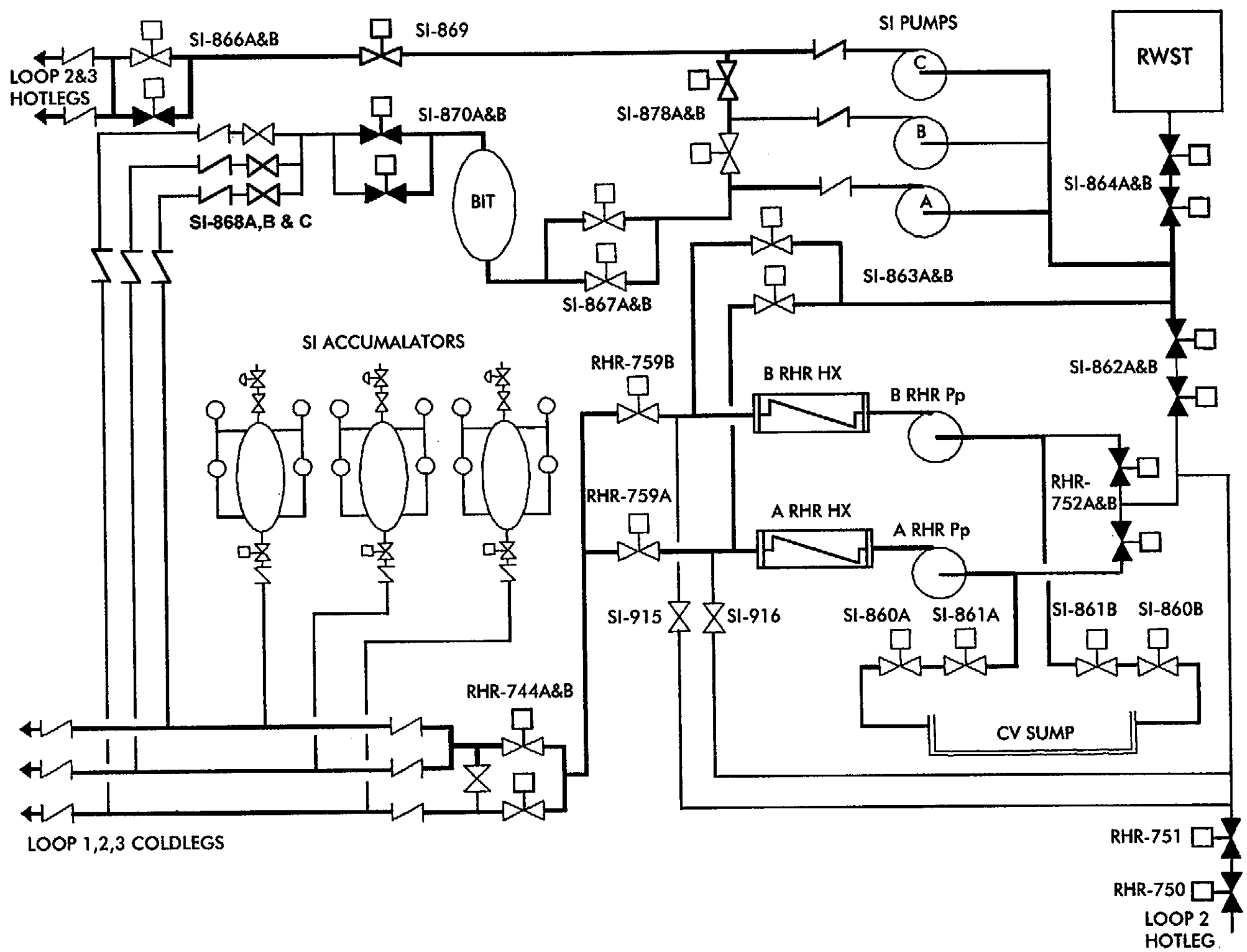
DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of ECCS flowpath to determine core cooling method

REFERENCES SUPPLIED:

LONG TERM RECIRCULATION WITH RCS PRESSURE <125 PSIG

SI-FIGURE-4 (Rev. 0)



RHR-759A throttled to 1200 gpm
RHR-759B throttled to 2300 gpm

INFORMATION USE ONLY

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Opening SI-866A AND SI-866B, HOT LEG INJs, with only one SI Pump running will cause pump runout.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
3.	Align For Hot Leg Recirculation As Follows:	
	a. Check SI Pump Status - TWO RUNNING	a. Stop the running SI Pump AND RHR Pump.
	b. Verify SI-866A, LOOP 3 HOT LEG INJ - OPEN	b. Open SI-866B, LOOP 2 HOT LEG INJ.
	c. Verify BIT OUTLET Valves - CLOSED • SI-870A • SI-870B	
	d. Check SI Valve Status • SI-866 - ONE OPEN • SI-870A & B - BOTH CLOSED	d. WHEN the valves have repositioned, THEN Go To Step 3.e.
	e. Check Pump Status • One RHR Pump - RUNNING • Two SI Pumps - RUNNING	e. Start One RHR Pump AND One SI Pump on each available Emergency Bus.
	f. Go To Step 8.	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Steps 4 through 7 must be performed without delay to minimize the time without flow through the core.

****

4. Perform The Following:

   a. Verify the RHR PUMPs - ALL STOPPED:

   b. Verify RHR HX DISCH Valves - CLOSED

      - RHR-759A
      - RHR-759B

   c. Verify RHR LOOP RECIRC Valves - OPEN

      - SI-863A
      - SI-863B

****

CAUTION

Opening SI-866A AND SI-866B, HOT LEG INJs, with only one SI Pump running will cause pump runout.

****

5. Verify The Following Valves Aligned For Hot Leg Recirculation:

INSTRUCTIONS	RESPONSE NOT OBTAINED
a. SI-866A, LOOP 3 HOT LEG INJ - OPEN	a. Open SI-866B, LOOP 2 HOT LEG INJ.

   b. BIT OUTLET Valves - CLOSED

      - SI-870A
      - SI-870B

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Valves RHR-759A and RHR-759B, RHR HX DISCHs, are closed. The RHR Pumps will run dead-headed and are subject to damage until the SI Pumps are started.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Establish Hot Leg Recirculation Flow As Follows:	
	a. Check RHR-759A - CLOSED	a. Perform the following: 1) Verify CLOSED RHR-759B. 2) Verify RHR PUMP A is stopped. 3) Open SI-863B, RHR LOOP RECIRC. 4) Start RHR PUMP B. 5) Go To Step 7.
	b. Open SI-863A, RHR LOOP RECIRC.	b. Perform the following: 1) Verify RHR-759B CLOSED. 2) Open SI-863B, RHR LOOP RECIRC. 3) Close SI-863A. 4) Start RHR PUMP B 5) Go To Step 7.
	c. Start RHR PUMP A	c. Perform the following: 1) Verify RHR-759B CLOSED. 2) Open SI-863B, RHR LOOP RECIRC. 3) Start RHR PUMP B

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Start One SI Pump On Each Available Emergency Bus	
8.	Check Indicated Flow On The Appropriate Flow Meters:	

PATH	FLOW METERS
SI-866B	FI-940, SI HOT LEG HEADER FLOW FI-933, SI LOOP 2 HOT LEG FLOW
SI-866A	FI-940, SI HOT LEG HEADER FLOW FI-932, SI LOOP 3 HOT LEG FLOW

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Determine If Flow Should Be Established To Cold Legs As Follows:	
	a. Check RCS pressure - LESS THAN 125 PSIG	a. Go To Step 11.
	b. Check <u>ALL</u> of the below components - OPERABLE • FI-605, RHR TOTAL FLOW • RHR-759A & B, RHR HEAT EXCHANGER OUTLETs • SI-863A & B, RHR LOOP RECIRCs. • RHR Pumps A & B	b. Go To Step 11.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
10.	Align For Cold Leg Injection As Follows:	
	a. Establish communications with operators stationed at the breakers for RHR HEAT EXCHANGER OUTLETs: • RHR-759A (MCC-5, CMPT 14C) • RHR-759B (MCC-6, CMPT 13C)	
	b. Start the second RHR PUMP	b. Go To Step 11.
	c. Verify BOTH RHR LOOP RECIRC Valves - OPEN • SI-863A • SI-863B	
	d. Open RHR-759A, RHR HX A DISCH <u>AND</u> locally open RHR-759A Breaker when RHR flow on FI-605 indicates 1200 gpm	
	e. Open RHR-759B, RHR HX B DISCH <u>AND</u> locally open RHR-759B breaker when RHR flow on FI-605 indicates 2300 gpm	
	f. Go To Step 15	
11.	Check Time Since Hot Leg Flow Established - 16 HOURS	<u>WHEN</u> 16 hours has elapsed, <u>THEN</u> Go To Step 12.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
12.	Establish Cold Leg Injection As Follows:	
	a. Check SI Pump Status - TWO RUNNING	a. Stop the running SI Pump <u>AND</u> RHR Pump.
	b. Verify at least one BIT OUTLET Valve - OPEN • SI-870A <u>OR</u> • SI-870B	
	c. Verify SI-869, SI HOT LEG HDR - CLOSED	c. Verify BOTH SI-866A <u>AND</u> SI-866B are CLOSED.
	d. Check SI Valve Status • SI-869 - CLOSED • SI-870A <u>OR</u> B - OPEN	d. <u>WHEN</u> the valves have repositioned, <u>THEN</u> Go To Step 12.e.
	e. Check Pump Status • One RHR Pump - RUNNING • Two SI Pumps - RUNNING	e. Start One RHR Pump <u>AND</u> One SI Pump on each available Emergency Bus.
13.	Check Time Since Cold Leg Flow Established - 16 HOURS	Contact Plant Operations Staff to evaluate long term plant status. When 16 hours has elapsed, <u>THEN</u> Go To Step 14.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
14.	Establish Hot Leg Injection As Follows:	
	a. Check SI Pump Status - TWO RUNNING	a. Stop the running SI Pump <u>AND</u> RHR Pump.
	b. Verify SI-869, SI HOT LEG HDR - OPEN	
	c. Verify one HOT LEG INJ Valve - OPEN • SI-866A <u>OR</u> • SI-866B	
	d. Verify BIT OUTLETs - CLOSED • SI-870A • SI-870B	
	e. Check SI Valve Status • SI-866 - ONE OPEN • SI-870A & B - BOTH CLOSED	e. <u>WHEN</u> the valves have repositioned, <u>THEN</u> Go To Step 14.f.
	f. Check Pump Status • One RHR Pump - RUNNING • Two SI Pumps - RUNNING	f. Start One RHR Pump <u>AND</u> One SI Pump on each available Emergency Bus.
	g. Go To Step 11	
15.	Contact Plant Operations Staff To Evaluate Long Term Plant Status	

- END -

RHR-03 007

The following valve lineup exists on the RHR system:

* SI-860A and B closed (CV SUMP TO RHR)
* SI-861A and B closed (CV SUMP TO RHR)
* SI-862A and B open (RWST TO RHR)
* RHR-744A and B open (RHR COLD LEG INJ)
* RHR-750 and RHR-751 closed (RHR LOOP SUPPLY)
* RHR-759A and B open (RHR HX DISC)

Which ONE (1) of the following flow paths is the RHR system aligned?

✓A. Injection from the RWST.

B. Cold leg recirculation from the containment sump.

C. Long term hot leg recirculation from the containment sump.

D. Cooldown lineups for normal plant cooldowns.

Question: 100

Given the following conditions:

- A Large Break LOCA has occurred.
- PATH-1 is being implemented.
- The CRSS directs you to "Verify Supplement D components capable of recirc."

Which ONE (1) of the following describes the actions permitted during performance of Supplement D, "Emergency Recirculation Equipment"?

a. Restoring flowpath from containment sump to RHR

b. Aligning flowpath from RHR pumps to the SI pumps

c. Restoring control power to SI valves controlled from the RTGB

d. Aligning flowpath from SI pumps to the hot legs

Answer:

c. Restoring control power to SI valves controlled from the RTGB

5.2.5 (Continued)

5. Supplement C - This supplement contains instructions that align the plant for cold leg recirculation. It is entered from EPP-3, Loss of All AC Power Recovery with SI Required.

6. Supplement D - This supplement is a listing of valves and components which must be available for Cold Leg Recirculation. Path-1 has a step which asks if Supplement D components are available. This means that Supplement D is to be reviewed to ensure that the valves or components listed are capable of being repositioned when the transition has been made to EPP-9, Transfer to Cold Leg Recirculation. When referenced by Path-1, Supplement D should **NOT** be used as permission to realign the valves included on that Supplement. It is acceptable, however, to restore control power to SI valves on the RTGB. It should be noted that all Supplement D components are not required to be capable of being repositioned. As a minimum the following are required:

   - One flowpath from the CV sump to the RHR Pumps.
   - One flowpath from the RHR Pumps to the SI Pumps.
   - One flowpath from the required pumps to the core.
   - Pumps as specified in the Supplement.

7. Supplement E - This supplement contains parameters to be monitored to verify that natural circulation flow exists. This allows Operations the option of performing a natural circulation cooldown in accordance with EPP-5 or maintaining current plant conditions while on natural circulation using Supplement E.

QUESTION NUMBER: 100
TIER/GROUP: ***RO*** 1/2 ***SRO***
K/A: 011 2.4.17

Knowledge of EOP terms and definitions (LBLOCA).

K/A IMPORTANCE: ***RO*** 3.1 ***SRO***
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: OMM-022-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of early action steps related to OMM-022.

REFERENCES: OMM-022

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number OMM-022-14 004

JUSTIFICATION:

a. Plausilbe since this is a flowpath that will be required for recirculation, but no valves are to be repositioned using Supplement D.

b. Plausible since this is a flowpath that will be required for recirculation, but no valves are to be repositioned using Supplement D.

c. **CORRECT** Supplement D is not used as permission to realign valves. It is acceptable, however, to restore control power to SI valves on the RTGB.

d. Plausible since this is a flowpath that will be required for long term recirculation, but no valves are to be repositioned using Supplement D.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of procedural requirements for EPP Supplements

REFERENCES SUPPLIED:

INITIAL SUBMITTAL

ROBINSON EXAM 2001-301

MARCH 26 - APRIL 2, 2001

INITIAL SUBMITTAL - SRO ONLY
WRITTEN EXAMINATION QUESTIONS

Question: 16

Given the following conditions:

- While performing a surveillance on LT-460, I&C personnel discovered at 1200 that the high level trip setpoint for the channel was 87.5%, which is outside the calibration tolerance band.
- The I&C personnel adjusted the LT-460 high level trip setpoint back to 91.0% at 1215 and completed the surveillance satisfactorily.
- They report the "as found" information to the I&C Supervisor who determines that the channel was inoperable in the "as found" condition.
- The I&C Supervisor notifies the SSO at 1230 of the inoperability of the channel in the "as found" condition.

Which ONE (1) of the following statements is correct concerning the operability of the channel in accordance with Technical Specifications?

a. An operability determination should be conducted to determine the total time the channel was inoperable.

b. An operability determination is **NOT** required since the channel is now operable

c. The channel is **NOT** operable and the bistables associated with LT-460 must be placed in a tripped condition no later than 1800.

d. The channel is **NOT** operable and he bistables associated with LT-460 must be placed in a tripped condition no later than 1830.

Answer:

b. An operability determination is **NOT** required since the channel is now operable

QUESTION NUMBER: 16
TIER/GROUP: *RO* *SRO* 3
K/A: 2.1.33

Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

K/A IMPORTANCE: *RO* *SRO* 4.0
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 2

OBJECTIVE: PZR-13

Given a plant condition and a copy of Technical Specifications, DETERMINE the applicable Technical Specifications requirements for the PZR and PRT System IAW H. B. Robinson Technical Specifications and Technical Specification Interpretations.

REFERENCES: TS Table 3.3.1-1

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number PZR-13 002

JUSTIFICATION:

a. Plausible since the channel is operable and an operability determination would be required if the as found setpoint was less conservative than allowed value, but the as found setpoint was more conservative.

b. **CORRECT** The channel is operable and an operability determination is not required since the as found setpoint was more conservative than allowed value.

c. Plausible since it would be inoperable if the trip setpoint was not conservative with respect to the allowable value, but the trip setpoint as found was conservative with respect to the allowable value and was adjusted to within calibration tolerance.

d. Plausible since it would be inoperable if the trip setpoint was not conservative with respect to the allowable value, but the trip setpoint as found was conservative with respect to the allowable value and was adjusted to within calibration tolerance.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 4

Application of operability determination for an out-of-tolerance instrument

REFERENCES SUPPLIED:

Table 3.3.1-1 (page 1 of 7)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11	≤ 110.93% RTP	108% RTP (2)
b. Low	$1^{(b)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 26.93% RTP	24% RTP
3. Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 37.02% RTP	25% RTP
	$2^{(d)}$	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 37.02% RTP	25% RTP
4. Source Range Neutron Flux	$2^{(d)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.28 E5 cps	1.0 E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.28 E5 cps	1.0 E5 cps
	$3^{(e)}$, $4^{(e)}$, $5^{(e)}$	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.
(2) The Nominal Trip Setpoint is as stated unless reduced as required by one or more of the following requirements: LCO 3.2.1 Required Action A.2.2; LCO 3.2.2 Required Action A.1.2.2; or LCO 3.7.1 Required Action B.2.
(a) With Rod Control System capable of rod withdrawal, or one or more rods not fully inserted.
(b) Below the P-10 (Power Range Neutron Flux) interlock.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication and alarm.

Question: 17

Given the following conditions:

- The unit has been shutdown for 30 days for refueling.
- Refueling cavity level is 18" below the flange.
- Initial water temperature is 106 °F.
- RHR cooling is lost.

Given the supplied references, which ONE (1) of the following indicates approximately how much time exists before Containment Closure is required?

a. 30 minutes

b. 35 minutes

c. 12.9 hours

d. 14.0 hours

Answer:

a. 30 minutes

QUESTION NUMBER: 17
TIER/GROUP: *RO* *SRO* 1/2
K/A: 025 2.1.25

Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data (Loss of RHR).

K/A IMPORTANCE: *RO* *SRO* 3.1
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 7

OBJECTIVE: AOP-020-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of RHR events as directed in AOP-020.

REFERENCES: Curve 3.5
OMM-033

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number GP-008-09 001

JUSTIFICATION:

a. **CORRECT** Using Curve 3.5, intersection of 30 days and "-10 to -36 Below Flange" curve results in a TIF of 0.32 minutes per degree (going to left from curve). Using formula, T = (200-106) x 0.32 = 30 minutes.

b. Plausible since this value is determined using the incorrect curve (Refueling Cavity Full) resulting in a TIF of 0.37 minutes per degree. T = (200-106) x 0.37 = 35 minutes.

c. Plausible since this value is determined using the correct curve (-10 to -36 Below Flange), but uses scale to right, instead of left, resulting in a TIF of 8.2 minutes per degree. T = (200-106) x 8.2 = 711 minutes = 12.9 hours.

d. Plausible since this value is determined using the incorrect curve (Refueling Cavity Full), but uses scale to right, instead of left, resulting in a TIF of 8.9 minutes per degree. T = (200-106) x 8.9 = 837 minutes = 14.0 hours.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating***

Application of plant curves to determine time to CV closure

REFERENCES SUPPLIED: Curve 3.5

8.1.2 **IF** the estimated time to close the open penetration exceeds 30 minutes, **THEN** an evaluation should be performed for the open penetration **AND** Operations Manager approval on Attachment 10.1 will be required.

> **NOTE:** Do Not use the curve for "Refueling Cavity Full" on Curve 3.5, Time to CV Closure, Unless Upper Internals are removed.
>
> **NOTE:** The curve for RCS "Water Level 0 inches to -10 inches Below Flange" will be utilized for RCS water level greater than 0 inches and Refueling Cavity Full with Upper Internals installed.

8.1.3 **IF** ALL the conditions in Section 8.1.1 above do not exist, **THEN** CV Closure Time shall be determined from Plant Curve 3.5, Time to CV Closure, in the Plant Curve Book.

> **NOTE:** When fuel is in the Containment, the crane used for removal and installation of the CV Equipment Hatch will remain available on-site to obtain CV Closure when implementation of Attachment 10.1 is required for the CV Equipment Hatch.

8.1.4 CV Equipment Hatch:

1. **IF** the following conditions are satisfied, **THEN** CV Closure is required in 5 hours.(DA 93-0046)
(ESR 95-00315)

   a. Fuel is in the Containment Vessel.

   b. RCS is intact (Excluding the Pressurizer PORV's).

   c. RCS temperature is less than 140°F.

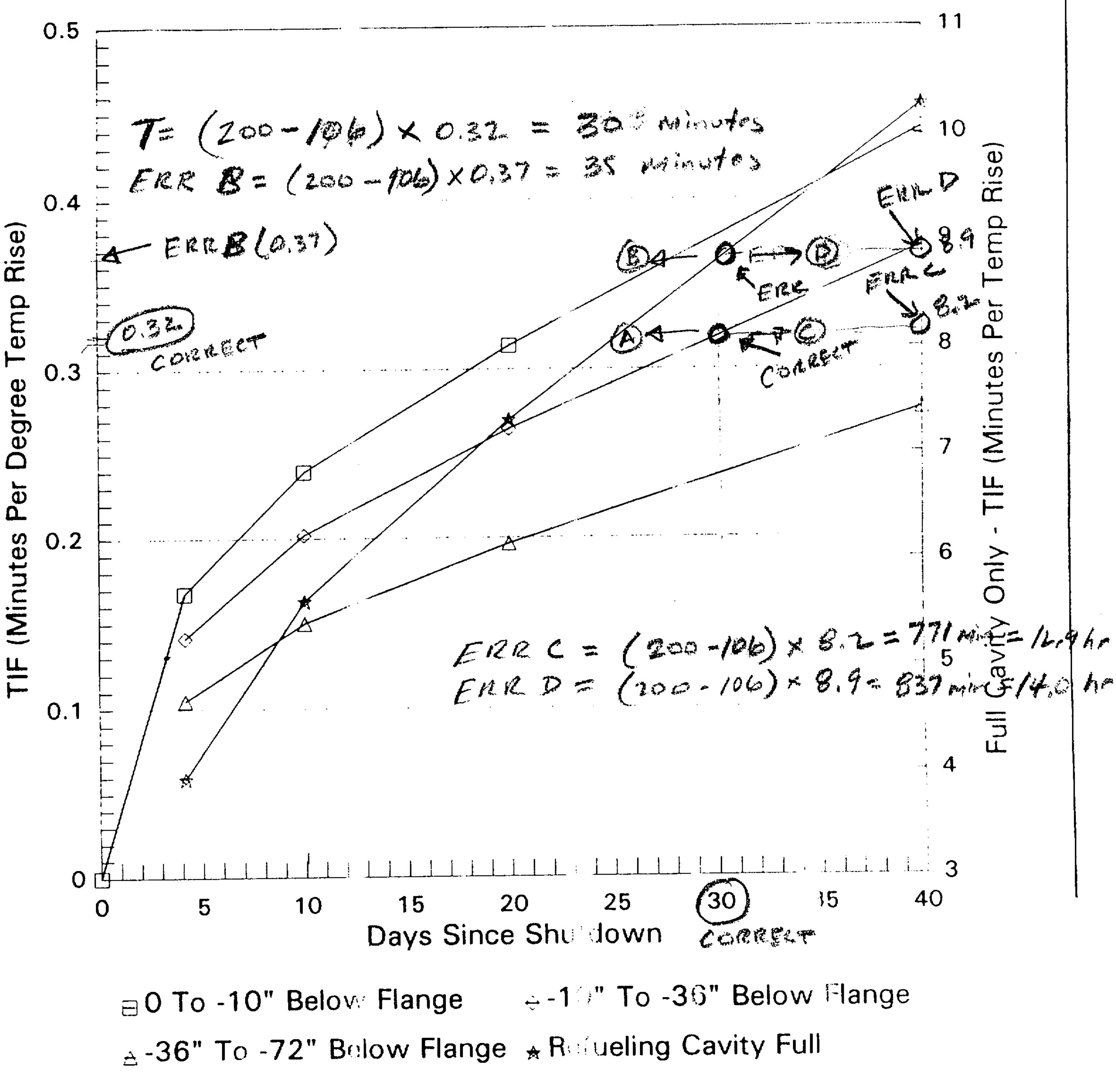
   d. RCS level is greater than -36 inches.

   e. One SI Pump with flowpath to three RCS Cold Legs is available.

   f. One Charging Pump with flowpath through CVC-310B, LOOP 2 COLD LEG CHG available.

Curve 3.5 - Time To CV Closure

Time = (200 - Initial Water Temp.) x Thermal Inertia Factor (TIF)



Based on Calculation RNP-M/MECH-1590

Use Thermal Inertia Factor = 0.00167 x t(hrs) prior to 100 Hours After Shutdown

INFORMATION USE

8.0 **INSTRUCTIONS**

8.1 Determining Penetration Closure Times

> **NOTE:** CV Closure time is not applicable with the core fully off loaded to the Spent Fuel Building.
>
> **NOTE:** AOP-020, Loss of Residual Heat Removal (Shutdown Cooling), will be utilized to provide core cooling if all RHR is lost.
>
> **NOTE:** When opening the CV Personnel Hatch, at least one of the doors will be capable of being closed. Any equipment impeding the closing of one of the doors shall be located in such a way that it can be immediately removed from the opening through the use of quick disconnects, clamps, etc.

8.1.1 **IF** the following conditions are satisfied, **THEN** allowed CV Closure Time is 30 minutes for all penetrations except the CV Equipment Hatch:

1. Two Trains of RHR are OPERABLE **AND**
   - Reactor Coolant System average temperature is less than or equal to 200°F
   - Reactor Coolant System level is above -36 inches

OR

2. One Train of RHR OPERABLE with refueling cavity level between 16 and 29 inches as indicated on the Refueling Cavity Level Indicator **AND**
   - Reactor Coolant System average temperature is less than or equal to 200°F
   - Reactor Vessel Upper Internals removed

GP-008-09 001

Given the following plant conditions:

- Plant has been shutdown for 20 days for refueling
- Refueling cavity is full
- Initial water temperature is 130 degrees F
- RHR cooling is lost

How much time exists before CV closure is required?

A. Approximately 8.75 minutes

B. Approximately 52 minutes

✓C. Approximately 8.75 hours

D. Approximately 52 hours

Question: 18

Given the following conditions:

- SG Tube Leakage in excess of Technical Specification limits was detected with the unit at power.
- The leaking SG has been identified.
- AOP-035, "SG Tube Leak," is being implemented.
- The leaking SG has been isolated.
- The RCS has been cooled down to 480 °F by core exit thermocouple readings.
- The RCS has been depressurized to less than leaking SG pressure and stabilized.
- All RCPs are running .
- Pressurizer level is 85%.

Which ONE (1) of the following describes the actions the operators should take if the affected SG level begins to decrease?

a. Increase charging flow

b. Turn on pressurizer heaters

c. Depressurize using normal sprays

d. Depressurize using auxiliary spray

Answer:

b. Turn on pressurizer heaters

QUESTION NUMBER: 18
TIER/GROUP: *RO* *SRO* 1/2
K/A: 037AA2.16

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Pressure at which to maintain RCS during S/G cooldown

K/A IMPORTANCE: *RO* *SRO* 4.3
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 4

OBJECTIVE: PATH-2-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to SGTR as directed in PATH-2.

REFERENCES: AOP-035

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number AOP-035-08 003

JUSTIFICATION:

a. Plausible since this will cause SG level to increase, but increased charging flow is not used due to the already high pressurizer level.

b. CORRECT RCS pressure is less than SG pressure for SG level to decrease. Raising RCS pressure will cause backflow to stop. Charging is not used to raise pressure since level is already high in the pressurizer.

c. Plausible since this action is used when SG level is changing, but depressurization is used when SG level is rising to create backflow from the SG to the RCS.

d. Plausible since this action is used when SG level is changing, but depressurization is used when SG level is rising to create backflow from the SG to the RCS.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of plant conditions during SG tube leak to determine proper actions

REFERENCES SUPPLIED:

CONTINUOUS USE

ATTACHMENT 3

S/G LEVEL CONTROL

(Page 1 of 1)

1. Monitor affected S/G level during cooldown.

2. WHEN required by the table below, THEN use the following in order of priority to depressurize the RCS:

   a. Normal spray

   b. Auxiliary spray with letdown in service

3. Take the action specified in the table below to maintain stable level in the affected S/G.

PZR LEVEL	**AFFECTED S/G LEVEL**		
	INCREASING	DECREASING	OFFSCALE HIGH
LESS THAN 24%	INCREASE CHARGING FLOW DEPRESSURIZE RCS	INCREASE CHARGING FLOW	INCREASE CHARGING FLOW MAINTAIN RCS AND RUPTURED S/G PRESSURES EQUAL
BETWEEN 24% AND 50%	DEPRESSURIZE RCS	TURN ON PZR HEATERS	MAINTAIN RCS AND RUPTURED S/G PRESSURES EQUAL
BETWEEN 50% AND 71%	DECREASE CHARGING FLOW DEPRESSURIZE RCS	TURN ON PZR HEATERS	MAINTAIN RCS AND RUPTURED S/G PRESSURES EQUAL
GREATER THAN 71%	DECREASE CHARGING FLOW	TURN ON PZR HEATERS	MAINTAIN RCS AND RUPTURED S/G PRESSURES EQUAL

- END -

AOP-035-08 003

Given the following plant conditions:

- Leakage in excess of tech spec limits was detected with the plant at power
- The leaking S/G has been identified
- AOP-035, S/G Tube Leakage was entered and is still in effect
- The leaking S/G has been isolated
- The RCS was cooled down to 480 F by core exit thermocouple readings
- The RCS was depressurized to less than leaking S/G pressure and stabilized
- All reactor coolant pumps are running
- Pressurizer level is 85%

Which one of the following describes the actions the operators should take if the affected S/G level begins to increase?

A. Depressurize RCS using normal spray

B. Decrease charging flow and depressurize RCS using normal spray

C. Decrease charging flow and turn on heaters

✓D. Decrease charging flow

Question: 19

Given the following conditions:

- The unit is operating at 40% power.
- An instrument air header break has occurred.
- Instrument air pressure at the receiver is 79 psig.
- Charging Pump 'A' speed has increased to maximum.
- HIC-121, Charging Flow, has failed open.
- VCT level has decreased to 11".

Which ONE (1) of the following actions should be directed to be taken?

a. Align the Charging Pump suction to the RWST and perform a plant shutdown per GP-006, "Normal Plant Shutdown From Power Operation to Hot Shutdown"

b. Align the Charging Pump suction to the RWST, trip the reactor, and go to PATH-1

c. Isolate charging and perform a plant shutdown per GP-006, "Normal Plant Shutdown From Power Operation to Hot Shutdown"

d. Isolate charging, trip the reactor, and go to PATH-1

Answer:

b. Align the Charging Pump suction to the RWST, trip the reactor, and go to PATH-1

Replacement

QUESTION NUMBER: 19
TIER/GROUP: ***RO*** ***SRO*** 1/2
K/A: 065AA2.06

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing

K/A IMPORTANCE: ***RO*** ***SRO*** 4.2
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: AOP-017-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of a loss of instrument air as directed by steps in AOP-017.

REFERENCES: AOP-017

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since the charging pump suction is to be aligned to the RWST, but a reactor trip is required instead of a normal shutdown.

b. CORRECT With VCT level low, the charging pump suction is to be aligned to the RWST, and since the RCS boron concentration will be rapidly increased a reactor trip is required.

c. Plausible since isolating charging would stop the level decrease in the VCT, but charging pump suction is aligned to the RWST and the plant is tripped.

d. Plausible since a reactor trip is required, but the charging pump suction is aligned to the RWST since charging cannot be isolated.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Analysis of plant conditions to determine reactor trip requirements in response to a loss of IA

REFERENCES SUPPLIED:

Question: 19

Given the following conditions:

- The unit is operating at 100 % power.
- AOP-017, "Loss of Instrument Air," is being implemented.

Which ONE (1) of the following would require a reactor trip during performance of this procedure?

a. Instrument Air header pressure at 58 psig

b. SA Compressor tripped while cross-connected to IA

c. Trip of **BOTH** Air Compressor 'D' and the Primary Air Compressor

d. CVCS Letdown isolated due to loss of air

Answer:

a. Instrument Air header pressure at 58 psig

QUESTION NUMBER: 19
TIER/GROUP: *RO* *SRO* 1/2
K/A: 065AA2.06

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing

K/A IMPORTANCE: *RO* *SRO* 4.2
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: AOP-017-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of a loss of instrument air as directed by steps in AOP-017.

REFERENCES: AOP-017

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AIR-09 006

JUSTIFICATION:

a. **CORRECT** IA pressure below 60 psig requires a reactor trip.

b. Plausible since attempts will be made to supply IA with SA in the event pressure continues to decrease, but a trip of the SA compressor would require isolating the cross-connect, not tripping.

c. Plausible since these 2 compressors will normally supply all IA requirements, but continued actions are taken to use IA compressors 'A' and 'B' if 'D' and the primary compressor are not available.

d. Plausible since this is an indication that control functions are beginning to be lost, but a reactor trip is not required until pressure is below 60 psig.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of reactor trip requirements in response to a loss of IA

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Plant Status - AT POWER	Go To Step 4.
* 2.	Check IA Header Pressure - LESS THAN 60 PSIG	IF IA pressure decreases to less than 60 psig, THEN Go To Step 3. Go To Step 4.
3.	Perform The Following: a. Trip the Reactor b. Go To PATH-1, while continuing with this procedure	
4.	Verify Instrument Air Compressor D - RUNNING	
5.	Verify The Primary Air Compressor - RUNNING	
* 6.	Check IA Header Pressure - LESS THAN 80 PSIG	IF IA pressure decreases to less than 80 psig, THEN observe NOTE prior to Steps 7 and 8 and perform Steps 7 and 8. Observe the NOTE Prior To Step 9 and Go To Step 9.

Question: 20

Given the following conditions:

- The unit is operating at 100% power.
- All plant systems are available.
- Maintenance is being planned on the following system trains that will make them each unavailable for between 42 and 48 hours:
  - PZR PORV 456
  - MDAFW Pump 'A'
  - SG 'C' PORV
  - RHR Pump 'A'

Given the supplied references, which ONE (1) of the following combinations are permitted to be taken out at the same time based on these planned maintenance times?

a.
- PZR PORV 456
- RHR Pump 'A'

b.
- PZR PORV 456
- MDAFW Pump 'A'

c.
- RHR Pump 'A'
- SG 'C' PORV

d.
- MDAFW Pump 'A'
- SG 'C' PORV

Answer:

d.
- MDAFW Pump 'A'
- SG 'C' PORV

QUESTION NUMBER: 20
TIER/GROUP: *RO* *SRO* 3
K/A: 2.2.18

Knowledge of the process for managing maintenance activities during shutdown operations.

K/A IMPORTANCE: *RO* *SRO* 3.6
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 1

OBJECTIVE: OMM-048-09

DEMONSTRATE the ability to evaluate a sample work schedule using Table 2 of the Matrix to determine plant configurations that are not recommended.

REFERENCES: OMM-048

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number PLP-056-09 005

JUSTIFICATION:

a. Plausible since this combination of components are supplied by same train, but referencing matrix indicates that this combination can be removed for up to 11 hours only which is less than the planned maintenance outage time.

b. Plausible since this combination of components are supplied by same train, but referencing matrix indicates that this combination can be removed for up to 37 hours only which is less than the planned maintenance outage time.

c. Plausible since this combination of components are supplied by same train, but referencing matrix indicates that this combination can be removed for up to 22 hours only which is less than the planned maintenance outage time.

d. CORRECT Matrix indicates that this combination can be removed from service simultaneously for up to 59 hours, which is greater than the expected maintenance period.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Application of PSA assessment of maintenance activities

REFERENCES SUPPLIED: OMM-048, Attachment 10.2

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 2. Matrix Showing Allowable Hours for Plant Configurations To Remain Non-Risk Significant

(DELTA CDP<1E-06)

Train A Matrix

Exceeding these allowed hours require PGM approval, review of non-quantifiable factors, contingency planning and PSA insights.
X - Safety Significant Exceeds Maximum Instantaneous CDF of 1E-3 and SHOULD BE AVOIDED

		RPS CHANNEL A	RCS PZR PORV 456	RHR PUMP A	CVCS CHGP B	SI PUMP A	S/G A PORV RV-1	S/G B PORV RV-2	S/G C PORV RV-3	MFWP A	AFW MDP A	AFW SDP	SW PUMP A	SW PUMP B	CCW PUMP A	CCW PUMP B	EDG A	EMERGENCY BUS E1	DC BAT CHG A/A1	AIR COMP A	AIR COMP PRIM	FIRE PUMP DIESEL	DEEPWELL PUMP B
		1080	2005	2045	2060	2080	3020	3020	3020	3050	3065	3065	4060	4060	4080	4080	5095	5175	5235	6135	6135	6175	6270
RPS CHANNEL A	1080	804	56	136	461	326	117	117	117	296	78	71	584	584	466	471	122	718	639	804	775	617	303
RCS PZR PORV 456	2005	56	93	11	85	26	54	54	54	39	37	26	86	85	83	76	24	91	90	91	87	92	79
RHR PUMP A	2045	136	11	174	149	169	22	22	22	116	65	60	161	161	154	155	106	169	165	173	172	164	143
CVCS CHGP B	2060	461	85	149	1081	147	124	124	124	363	95	96	718	718	506	279	165	932	804	1068	1043	789	461
SI PUMP A	2080	326	26	169	147	576	83	83	83	278	78	86	456	456	404	407	188	531	489	576	569	484	337
S/G A PORV RV-1	3020	117	54	22	124	83	136	52	52	109	59	59	128	128	124	124	80	134	131	136	136	131	117
S/G B PORV RV-2	3020	117	54	22	124	83	52	136	52	109	59	59	128	128	124	124	80	134	131	136	136	131	117
S/G C PORV RV-3	3020	117	54	22	124	83	52	52	136	109	59	59	128	128	124	124	80	134	131	136	136	131	117
MFWP A	3050	296	39	116	363	278	109	109	109	548	14	45	438	438	389	393	144	506	468	548	541	463	311
AFW MDP A	3065	78	37	65	95	78	59	59	59	14	104	9	98	98	97	97	73	102	101	104	102	100	93
AFW SDP	3065	71	26	60	96	86	59	59	59	45	9	105	92	90	98	98	14	100	92	105	102	100	93
SW PUMP A	4060	584	86	161	718	456	128	128	128	438	98	92	2190	73	782	850	184	1718	1307	2190	2086	903	551
SW PUMP B	4060	584	85	161	718	456	128	128	128	438	98	90	73	2190	834	850	184	1718	1307	2190	2086	913	551
CCW PUMP A	4080	466	83	154	506	404	124	124	124	389	97	98	782	834	1348	X	161	995	834	1348	1307	932	506
CCW PUMP B	4080	471	76	155	279	407	124	124	124	393	97	98	850	850	X	1390	172	1153	963	1369	1327	942	509
EDG A	5095	122	24	106	165	188	80	80	80	144	73	14	184	184	161	172	196	190	184	196	194	131	128
EMERGENCY BUS E1	5175	718	91	169	932	531	134	134	134	506	102	100	1718	1718	995	1153	190	6738	2137	6257	5475	2037	706
DC BAT CHG A/A1	5235	639	90	165	804	489	131	131	131	468	101	92	1307	1307	834	963	184	2137	3129	3129	2920	1537	93
AIR COMP A	6135	804	91	173	1068	576	136	136	136	548	104	105	2190	2190	1348	1369	196	6257	3129	8760	8760	2920	804
AIR COMP PRIM	6135	775	87	172	1043	569	136	136	136	541	102	102	2086	2086	1307	1327	194	5475	2920	8760	8760	2738	789
FIRE PUMP DIESEL	6175	617	92	164	789	484	131	131	131	463	100	100	903	913	932	942	131	2037	1537	2920	2738	2920	635
DEEPWELL PUMP B	6270	303	79	143	461	337	117	117	117	311	93	93	551	551	506	509	128	706	93	804	789	635	804

456 – RHR A 11 hr
456 – AFW A 37 hr
RHR A – SG PORV C 22 hr
AFW A – SG PORV C 59 hr

PLP-056-09 005

Given the following plant conditions:

- The unit is at 100% power operations
- All plant systems available
- You need to plan maintenance on the following system trains that will make them unavailable for less that 72 hours: SW Pump "C", SI Pump "C", RHR Pump "B', and the Diesel Firewater Pump.

Which ONE (1) of the following combinations are allowed by Table 2 assuming planned maintenance exceeds no Tech. Spec. limits?

A. Diesel Firewater pump and SW pump "C".

B. Diesel Firewater pump, RHR pump "B", and SI pump "C".

✓C. SW Pump "C" and SI pump "C".

D. RHR pump "B" and SW pump "C".

Question: 36

Given the following conditions:

- A reactor trip and safety injection have occurred due to a SGTR.
- A transition was made from PATH-1 to PATH-2.
- During the performance of PATH-2, an improper communication results in the CRSS incorrectly transitioning to EPP-17, "SGTR With Loss of Reactor Coolant: Subcooled Recovery."
- The first four (4) steps of EPP-17 either verify actions previously completed in PATH-1 or check plant indications only (**NO** ACTIONS ARE ACTUALLY PERFORMED).
- After completion of the first four (4) steps of EPP-17, the CRSS recognizes that the wrong procedure is being implemented.

Which ONE (1) of the following describes the actions that the CRSS should take to most quickly mitigate the consequences of the SGTR **WITHOUT** violating any procedures?

a. Continue on in EPP-17, transitioning to PATH-2, Entry Point J, when directed

b. Transition back to PATH-1, Entry Point A

c. Transition back to PATH-2, Entry Point J

d. Transition back to the point in PATH-2 where the incorrect transition was made

Answer:

d. Transition back to the point in PATH-2 where the incorrect transition was made

QUESTION NUMBER: 36

TIER/GROUP: *RO* *SRO* 1/1

K/A: WE01EA2.2

Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A IMPORTANCE: *RO* *SRO* 3.9

10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 5

OBJECTIVE: OMM-022-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in OMM-022 by explaining the basis of each.

REFERENCES: OMM-022

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number OMM-022-03 010

JUSTIFICATION:

a. Plausible since EPP-17 will provide a transition point back to PATH-2, Entry Point J, but this will delay the mitigation of the event.

b. Plausible since this is a permissible method of recovering from an incorrect transition, but this will delay the mitigation of the event.

c. Plausible since this would mitigate the event sooner than transitioning back to PATH-1, but this is not an acceptable alternative.

d. **CORRECT** If the incorrect transition is immediately recognizable AND no alterations of the mitigation strategy have occurred, he may move back to the point in the Network where the incorrect transition has occurred.

DIFFICULTY:

Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of administrative requirements for incorrect procedure transitions

REFERENCES SUPPLIED:

8.3.10 Incorrect EOP Transition

1. Should the Operator determine that he is in an incorrect Path or EPP, he has two options:

   - If the incorrect transition is immediately recognizable **AND** no alterations of the WOG mitigative strategy have occurred, he may move back to the point in the Network where the incorrect transition has occurred.

   - If the incorrect transition is not immediately recognizable **OR** alterations in the mitigative strategy have occurred, the Operator should move to Path-1, Entry Point A, and start over.

2. During the rediagnosis described above, complete reactuation of the Engineered Safety Features is allowed, but not required. Reactuation of necessary safety features during rediagnosis is guided by the requirements of the applicable Foldout and Operator judgement based on the symptoms present.

8.3.11 Adverse Containment Conditions Usage

1. When adverse containment conditions develop, the use of adverse containment condition setpoints shall be initiated.

2. The use of adverse containment condition setpoints shall be maintained from that point forward, even when adverse containment conditions no longer exist.

3. An adverse containment condition setpoint may or may not be provided. The operator shall use a setpoint with no brackets if no setpoint within brackets is provided, even if adverse containment conditions exist.

8.3.12 Special EPP Priority

1. Certain contingency EPPs take precedence over FRPs because of their treatment of specific initiating events. In all such cases, this precedence is identified in a CAUTION or NOTE at the beginning of the EPP.

OMM-022-03 010

In regard to incorrect EOP transition, should the Operator determine that he is in an incorrect PATH or EPP, AND IF the incorrect transition is not immediately recognizable OR alterations in the mitigative strategy have occurred, the Operator should do which ONE (1) of the following?

A. Immediately monitor the CSFSTs and proceed to the highest priority Functional Restoration Procedure

✓B. Move to PATH-1, Entry Point A, and start over

C. Move back to the point in the network where the incorrect transition was made, hold a crew brief, and proceed on in the applicable PATH or EPP.

D. Hold a crew brief including the SSO, perform an evaluation of plant status, continue on in the procedure in effect.

Question: 37

Given the following plant conditions:

- During a plant transient, Control Bank 'D' rods are moved inward.
- After the plant stabilizes, the Reactor Operator recognizes that two (2) Control Bank 'D' rods are misaligned by greater than allowed by Technical Specification limits.

Which ONE (1) of the following actions are to be taken?

a.
- Verify Shutdown Margin within 1 hour
- Realign the misaligned rods or be in Mode 3 within 2 hours

b.
- Verify Shutdown Margin within 1 hour
- Realign the misaligned rods or reduce power to < 70% within 2 hours

c.
- Verify Shutdown Margin within 1 hour
- Shutdown to Mode 3 within 6 hours

d.
- Trip the reactor
- Go to PATH-1

Answer:

c.
- Verify Shutdown Margin within 1 hour
- Shutdown to Mode 3 within 6 hours

QUESTION NUMBER: 37
TIER/GROUP: *RO* *SRO* 1/1
K/A: 005AA2.03

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable

K/A IMPORTANCE:	*RO*	*SRO*	4.4
10CFR55 CONTENT:	*55.41(b) RO*	*55.43(b) SRO*	6

OBJECTIVE: AOP-001-07

DETERMINE the action(s) required by Technical Specifications associated with AOP-001

REFERENCES: TS 3.1.4
AOP-001

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number AOP-001-07 003

JUSTIFICATION:

a. Plausible since these actions similar to these would be taken if only one rod was misaligned or stuck, but more conservative actions are required with 2 or more rods.

b. Plausible since these actions similar to these would be taken if only one rod was misaligned or stuck, but more conservative actions are required with 2 or more rods.

c. **CORRECT** With more than one rod misaligned or stuck, SDM must be verified within 1 hour and the plant must be placed in Mode 3 within 6 hours.

d. Plausible since a reactor trip will be initiated if more than one rod is dropped, but with more than 1 rod misaligned / stuck a controlled shutdown is performed.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of TS and procedural requirements for more than one misaligned / stuck rod

REFERENCES SUPPLIED:

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits provided in the COLR. <u>OR</u> D.1.2 Initiate boration to restore required SDM to within limit. <u>AND</u> D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual rod positions within alignment limit.	12 hours <u>AND</u> Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable

(continued)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

IMMOVABLE/MISALIGNED RODS

NOTE

- TECH SPEC limits for bank positions greater than OR equal to 200 steps are 15 inches (24 steps) alignment with associated Group Step Counter position.
- TECH SPEC limits for bank positions less than 200 steps are 7.5 inches alignment with average IRPI position of associated Bank.
- Use IRPI and/or Incore Flux Map for determination of Control Rod misalignment.
- ERFIS display GD ROD LOG may be used for additional information.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*11.	Check IRPI Rod Misalignment - GREATER THAN TECH SPEC LIMIT	Perform the following: a. IF the ROD BANK SELECTOR Switch was in Individual Bank Select WHEN the Urgent Failure condition occurred, THEN contact Engineering for recovery actions to restore normal rod sequencing. b. WHEN the urgent failure condition is corrected, THEN Depress ROD ALARM RESET Button on RTGB AND verify APP-005-E2 clears. c. WHEN the urgent failure Alarm has been cleared, THEN observe the CAUTION prior to Step 23 and Go To Step 23.
12.	Check Number Of Rods Indicating Misalignment - GREATER THAN ONE	Observe the NOTE prior to Step 14 and Go To Step 14.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

IMMOVABLE/MISALIGNED RODS

(Page 5 of 31)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
13.	Perform The Following:	
	a. Check SDM - WITHIN THE LIMITS SPECIFIED IN THE COLR	a. Initiate boration to restore SDM within 1 hour.
	b. Within 6 hours Place the unit in Mode 3 using GP-006, Normal Plant Shutdown From Power Operation To Hot Shutdown.	
	c. Go To Step 61	

AOP-001-07 003

During a power escalation it is discovered that 2 Control Bank "D" rods are out of alignment. An unsuccessful attempt was made to realign the rods (rods would not move). The Reactor is currently at 80% power.

Select the appropriate operational restriction that applies for this condition:

✓A. Place the unit in Hot Shutdown within 8 hours.

B. Maintain power < 90% (or 0.9 APL) and determine Hot Channel Factors.

C. Borate the RCS an amount equat to the worth of the stuck rods.

D. No action required if Bank "D" is < 200 steps.

Question: 38

Using the supplied references, which ONE (1) of the following conditions would require a One-Hour Notification in accordance with AP-030, "NRC Reporting Requirements"?

a. A manual reactor trip is actuated from 20% power due to a break in the Main Turbine Electro Hydraulic Control system piping

b. An automatic safety injection is actuated at 100% power due to an I&C Technician lifting an incorrect lead

c. You receive a report that a previously reported employee's positive FFD test was erroneous and is due to an administrative error at the laboratory

d. While on your tour, you note that the WCC SRO's speech is slurred and you smell alcohol on his breath

Answer:

c. You receive a report that a previously reported employee's positive FFD test was erroneous and is due to an administrative error at the laboratory

QUESTION NUMBER: 38

TIER/GROUP: *RO* *SRO* 3

K/A: 2.4.30

Knowledge of which events related to system operations/status should be reported to outside agencies.

K/A IMPORTANCE: *RO* *SRO* 3.6

10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 1

OBJECTIVE: AP-030-03

Given a reportable event, DETERMINE the reporting requirements of AP-030.

REFERENCES: AP-030

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number HNP-SRO-2000 76

JUSTIFICATION:

a. Plausible since a TS required shutdown from Mode 1 is a 1-hour notification, but trips from power are 4-hour notifications.

b. Plausible since safety injection actuations due to valid signals are 4-hour notifications, but a signal as a result of human error are not considered valid actuations.

c. CORRECT Per Attachment 11.1, false positives of an employee's FFD are 1-hour notifications.

d. Plausible since this would be a 1-hour if this were an NRC employee.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Interpretation and application of conditions to determine reporting requirements

REFERENCES SUPPLIED: AP-030, Attachments 11.1 and 11.2

Question: 38

Given the supplied references, which ONE (1) of the following conditions would require a One-Hour Notification in accordance with AP-030, "NRC Reporting Requirements"?

a. A manual reactor trip is actuated from 20% power due to a break in the Main Turbine Electro Hydraulic Control system piping

b. An automatic safety injection is actuated at 100% power due to an I&C Technician lifting an incorrect lead

c. While at 400 °F during a plant cooldown, a fire destroys the FTS-2000 network in the Technical Support Center

d. While at 190 °F during a plant heatup, it is discovered that wire leads for **BOTH** Safety Injection pumps in the Sequencer cabinets were inadvertently left lifted

Answer:

c. While at 400 °F during a plant cooldown, a fire destroys the FTS-2000 network in the Technical Support Center

QUESTION NUMBER: 38
TIER/GROUP: *RO* *SRO* 3
K/A: 2.4.30

Knowledge of which events related to system operations/status should be reported to outside agencies.

K/A IMPORTANCE: *RO* *SRO* 3.6
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 1

OBJECTIVE: AP-030-03

Given a reportable event, DETERMINE the reporting requirements of AP-030.

REFERENCES: AP-030

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number HNP-SRO-2000 76

JUSTIFICATION:

a. Plausible since a TS required shutdown from Mode 1 is a 1-hour notification, but trips from power are 4-hour notifications.

b. Plausible since safety injection actuations due to valid signals are 1-hour notifications, but a signal as a result of human error are not considered valid actuations.

c. **CORRECT** Per Attachment 7.1, this would be addressed under loss of emergency assessment, off-site response, or communication capability and would require a 1-hour notification.

d. Plausible since this renders both trains of ECCS inoperable, but since the plant is shutdown this is a 4-hour notification.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Interpretation and application of conditions to determine reporting requirements

REFERENCES SUPPLIED: AP-030, Attachments 7.1 and 7.2

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

10 CFR 50.72 states that immediate reports shall be made to the NRC Operations Center of these Emergency Events via the FTS-2000 as specified in the Emergency Plan. 10 CFR 50.72 additionally identifies Non-Emergency Events which are to be reported within One-Hour or Four-Hours to the NRC. FTS -2000 Telephones, which are distinctly labeled, are tan in color and are located in the Control Room, the TSC, and the EOF.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
NOTE: 10 CFR 50.72 recognizes the Emergency Plan and its four Emergency Classes of Unusual Event, Alert, Site Area Emergency and General Emergency.			
EMERGENCIES 10 CFR 50.72(a)(i) 10 CFR 30.32(i)(3)(viii) 10 CFR 40.31(i)(3)(viii)	Emergency Unusual Event Alert Site Area Emergency General Emergency	HBRSEP shall notify the NRC of the declaration of any of the Emergency Classes specified in the Emergency Plan.	– Declaration of an Unusual Event, Alert, Site Area Emergency, or General Emergency – Discovery of an event that should have resulted in an Emergency Classification, but no emergency was declared – Discovery that a declared emergency exceeded the Emergency Action Levels for a higher emergency declaration, but the higher classification was not declared
ERDS ACTIVATION 10 CFR 50.72(a)(4)	ERDS Emergency	HBRSEP shall activate the ERDS as soon as possible but not later than one hour after declaring an Alert, Site Area Emergency, or General Emergency.	– An Alert, Site Area Emergency, or General Emergency is declared.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC			
If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ENS (FTS-2000) as soon as practical and in all cases within one hour of the occurrence of any of the following:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
SHUTDOWN REQUIRED BY TS 10 CFR 50.72(b)(1)(i)(A)	Shutdown TS Shutdown Power Reduction	The initiation of any shutdown required by the TS.	– Unplanned Shutdown initiated due to maximum specific activity of the Reactor Coolant Water (plant shutdown required by TS) – Reactor Coolant System Leakage in excess of 10 GPM for greater than 24 hours (plant shutdown required by TS) – Component Cooling Water Heat Exchanger inoperable (if not corrected prior to expiration of Required Action Completion Time)
DEVIATION FROM TS (10 CFR 50.54(X)) 10 CFR 50.72(b)(1)(i)(B)	Deviation Departure License Condition	Any deviation from the TS authorized pursuant to 10 CFR 50.54(x).	– Intentional deviation from an approved plant procedure in order to preserve plant safety 10 CFR 50.54(x) (See PRO-NGGC-0200)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRINCIPAL SAFETY BARRIERS SERIOUSLY DEGRADED 10 CFR 50.72(b)(1)(ii)	Degraded Safety Barriers Fission Product Barrier	Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;	– Fuel cladding failures in the reactor, or in the storage pool, that exceed expected values, or that are unique or widespread, or that are caused by unexpected factors, and would involve a release of significant quantities of fission products – Cracks and breaks in the piping or reactor vessel, or major components in the reactor coolant system, that have safety relevance (steam generators, reactor coolant pumps, valves, etc.) – Significant welding or material defects in the RCS – Serious temperature or pressure transients – Loss of relief and/or safety valve functions during operation – Loss of Containment function or integrity - Complete loss of containment integrity function including (1) containment leakage rate greater than allowed value per SR 3.6.1.1 (i.e., entry into LCO 3.6.1 Condition A), (2) loss of containment penetration isolation functional capability (i.e., both barriers), or loss of containment spray capability
UNANALYZED PLANT CONDITION 10 CFR 50.72(b)(1)(ii)(A)	Safety Function Unanalyzed Condition	[or that resulted in the nuclear power plant being:] In an unanalyzed condition that significantly compromises plant safety;	– OTΔT changes are declared inoperable due to summator module lag constants. The channel response time exceeded the value assumed in the accident analysis. – Accumulation of voids in systems designed to remove heat from the reactor, that could inhibit the ability to adequately remove heat from the core, particularly under natural circulation conditions

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ENS (FTS-2000) as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
CONDITION OUTSIDE DESIGN BASIS OF PLANT 10 CFR 50.72(b)(1)(ii)(B)	Design Bases Loss of Safety Function	[or that resulted in the nuclear power plant being:] In a condition that is outside the design basis of the plant;	– Discovery of design errors that renders a safety system inoperable – Discovery that a single train of a safety system has been incapable of performing its design function for an extended time (well beyond surveillance intervals or Required Action Completion Times) – Safety related piping found not to be seismically qualified in accordance with design bases requirements
CONDITION NOT COVERED BY OPERATING/EMERGENCY PROCEDURES 10 CFR 50.72(b)(1)(ii)(C)	OP AOP EOP PATH CSFST	[or that resulted in the nuclear power plant being:] In a condition not covered by the operating and emergency procedures.	– An event is occurring having significant implications for the health and safety of the public and no AOP or EOP is applicable to the condition.
NATURAL PHENOMENON OR CONDITION THREATENING PLANT SAFETY 10 CFR 50.72(b)(1)(iii)	Earthquake Hurricane Tornado Weather Explosion Railroad	Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.	– Natural phenomenon (ice storm that significantly hampers personnel in the conduct of activities necessary for safe operation of the plant). – External hazards (railroad tank car explosion that poses an actual threat to Plant safety)
ECCS DISCHARGE INTO RCS 10 CFR 50.72(b)(1)(iv)	ECCS Actuation Safety Injection	Any event that results or should have resulted in ECCS discharge into the reactor coolant system as a result of a valid signal.	– Manual or automatic Safety Injection System actuation in response to a valid signal (Section 4.5 of this procedure)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ENS (FTS-2000) as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OF EMERGENCY ASSESSMENT, OFF-SITE RESPONSE, OR COMMUNICATIONS CAPABILITY 10 CFR 50.72(b)(1)(v)	Selective Signaling System Sirens FTS-2000	Any event that results in a major loss of emergency assessment capability, off-site response capability, or communications capability (e.g., significant portion of control room indication, FTS-2000, or off-site notification system).	– Loss of 23 or more of 45 Public Warning Sirens (≥50%) as indicated on the siren activation system for a period of at least 30 minutes at any one time. – Loss of greater than 50% of communications capability (i.e., offsite communications systems which include the Selective Signaling System, the Essex System and the Local Government Radio System). – Loss of greater than 50% of the ability of the TSC or EOF to function. – Loss of instrumentation indication capability to the extent that an Emergency Action Level cannot be determined to exceed an emergency classification. – Loss of FTS-2000 System if identified by the plant (Not reportable if identified by NRC) – Loss of commercial telephone system to the extent that required communications could not be made to official offsite locations (e.g., EOCs, Warning Points)
INTERNAL THREAT TO PLANT SAFETY (FIRES, TOXIC GAS, RADIOLOGICAL RELEASE) 10 CFR 50.72(b)(1)(vi)	Fire Toxic Explosive Release Personnel Safety	Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.	– Fire confirmed inside Protected Area (if fire poses an actual threat to plant safety or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant). – Unplanned release of radioactive gases or toxic gas inside Protected Area (if release significantly hampered site personnel in the performance of duties necessary for safe operation of the plant).

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

HBRSEP shall immediately notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SAFETY LIMIT, LIMITING SAFETY SYSTEM SETTING EXCEEDED 10 CFR 50.36(c)(1)(i)(A)	Safety Limit Limiting Safety System Setting	If any safety limit is exceeded, shut down the reactor. HBRSEP shall notify the [NRC within 1 hour via FTS-2000 per 10 CFR 50.72(a)(1), See Emergency Plan Procedures]. Operation must not be resumed until authorized by the NRC.	– Reactor pressure exceeds 2735 psig while at power – The limits of TS Table 2.1.1-1 are exceeded – Limiting Safety System Settings in TS Table 3.3.1-1 are exceeded
SAFETY SYSTEM DOES NOT FUNCTION AS REQUIRED 10 CFR 50.36(c)(1)(ii)(A)	ESF RPS Limiting Safety System Setting	HBRSEP shall notify the NRC if the automatic safety system [to correct an abnormal situation before a safety limit is exceeded] has been determined not to function as required.	– A failure mechanism is discovered that indicates that the RPS will not function to trip the reactor under certain required conditions.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the FTS-2000 within one hour after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
THEFT/UNLAWFUL DIVERSION OF SNM OR SPENT FUEL SHIPMENT 10 CFR 73.71(a)(1)	SNM Spent Fuel Security Safeguards	Any discovery of the loss of any shipment of SNM or spent fuel, and within one hour after recovery of or accounting for such lost shipment	– Shipment Emergency Event (Reference 2.9)
THEFT/UNLAWFUL DIVERSION OF SNM 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(1)	Theft of SNM Diversion Security Safeguards	Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause: (1) A theft or unlawful diversion of SNM	– Shipment Emergency Event (Reference 2.9)
SABOTAGE OF PLANT EQUIPMENT 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(2)	Sabotage Damage to Plant SNM Spent Fuel Security Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (2) Significant physical damage to a power reactor...or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent fuel a facility or carrier possesses.	– Shipment Emergency Event (Reference 2.9) – Security Event (Reference 2.11)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the FTS-2000 within one hour after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
UNAUTHORIZED TAMPERING WITH PLANT EQUIPMENT 10 CFR 73, Appendix G, I(a)(3)	Unauthorized Use Tampering Security System Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (3) Interruption of normal operation of HBRSEP through the unauthorized use of or tampering with its machinery, components, or controls including the security system.	– Security Event (Reference 2.11)
ENTRY OF UNAUTHORIZED PERSON INTO PROTECTED OR VITAL AREA 10 CFR 73, Appendix G, I(b)	Unauthorized Entry Security Safeguards	An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.	– Security Event (Reference 2.11)
FAILURE, DEGRADATION, OR DISCOVERED VULNERABILITY OF SAFEGUARD SYSTEM 10 CFR 73, Appendix G, I(c) Procedure SEC-NGGC-2147	Degradation Vulnerability Safeguards Unauthorized Undetected Access Security	Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area or transport for which compensatory measures have not been employed.	
INTRODUCTION OF CONTRABAND INTO VITAL OR PROTECTED AREA 10 CFR 73, Appendix G, I(d)	Contraband Unauthorized Security Safeguards	The actual or attempted introduction of contraband into a protected area, material process area, vital area, or transport.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM HBRSEP shall immediately notify the NRC Operations Center via FTS-2000, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
LOSS OR THEFT OF LICENSED MATERIAL (>1000X 10 CFR 20 LIMITS) 10 CFR 20.2201(a)(i)	Loss Theft Missing Licensed Radioactive Material	Immediately notify the NRC, after its occurrence becomes known, any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in [10 CFR 20] Appendix C under such circumstances that it appears to HBRSEP that an exposure could result to persons in unrestricted areas.	– A radiography source is discovered missing. The source is licensed to the radiography contractor. If the contractor does not make the required notification, HBRSEP should notify the NRC Operations Center via FTS-2000.
EXTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, OR SNM (5X ANNUAL LIMIT) 10 CFR 20.2202(a)(1)	Byproduct Source SNM Exposure Dose Release Occupational	Notwithstanding any other requirements for notification, immediately notify the NRC of any event involving byproduct, source, or SNM possessed by HBRSEP that may have caused or threatens to cause any of the following conditions: 1. An individual to receive: (i) A total effective dose equivalent of 25 rems or more; or (ii) An eye dose equivalent of 75 rems or more; or (iii) A shallow dose equivalent to the skin or extremities of 250 rads or more; or 2. The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM			
HBRSEP shall immediately notify the NRC Operations Center via FTS-2000, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
INTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, SNM (>5X OCCUPATIONAL LIMIT) 10 CFR 20.2201(a)(i)	Intake Ingestion Release Source Byproduct SNM	The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - ISFSI HBRSEP shall immediately notify the NRC Operations Center via FTS-2000, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
ISFSI - ACCIDENTAL CRITICALITY OR LOSS OF SNM 10 CFR 72.74	ISFSI Criticality SNM Loss	The licensee shall notify the NRC Operations Center via FTS-2000 within one hour of discovery of accidental criticality or any loss of SNM.	– Unusually high radiation readings discovered in the vicinity of the ISFSI that could indicate possibility of a criticality event
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SNM SHIPMENTS HBRSEP shall notify the NRC Operations Center via the FTS-2000 within one hour of the following:			
LOST OR UNACCOUNTED SHIPMENT OF SNM 10 CFR 70.52(b) 10 CFR 73.71(a)(1)	Shipment Loss SNM Spent Fuel Theft Diversion Safeguards Security	HBRSEP shall notify the NRC Operations Center via the FTS-2000 within one hour after discovery of any loss of any shipment of SNM or spent fuel or any incident in which an attempt has been made, or is believed to have been made, to commit a theft or unlawful diversion of SNM.	– Shipment Emergency Event (Reference 2.9) – Security Event (Reference 2.11)
LOST OR UNACCOUNTED SHIPMENT OF SNM - RECOVERY 10 CFR 73.71(a)(1)	Recovery Accounting Shipment SNM Security Safeguards	HBRSEP shall notify the NRC Operations Center via the FTS-2000 within one hour after recovery of, or accounting for, any lost shipment of SNM.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FOLLOW-UP

With respect to the telephone notifications made under paragraphs (a) and (b) of 10 CFR 50.72, in addition to making the required initial notification, HBRSEP shall during the course of the event immediately report:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(1)	Degradation Emergency Class Change Update Termination	(i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.	– Refer to Reference 2.27
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(2)	Result Evaluation Effectiveness Unknown	(i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood.	
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(3)	Open Continuous Communication	Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.	– Refer to Reference 2.27

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS - NRC REGION II OFFICE

HBRSEP shall immediately notify the final delivery carrier and, by telephone and telegram, mailgram, or facsimile, the NRC Region II Office when:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
THEFT/UNLAWFUL DIVERSION OF TRITIUM 10 CFR 30.55(c)	Incident Theft Tritium Attempt Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft of more than 10 curies of tritium (outside of spent fuel) at any one time or more than 100 curies of tritium in one calendar year.	– 10 Curies of tritium discovered missing from the Chemistry Laboratory, and reason exists to suspect that the tritium was stolen
THEFT/UNLAWFUL DIVERSION OF SOURCE MATERIAL 10 CFR 40.64(c)	Incident Attempt Theft Diversion Source Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of more than 15 pounds of Source Material at any one time or 150 pounds of Source Material in any one calendar year.	– A source assembly is discovered missing from a new fuel shipment.
SHIPPING PACKAGE RADIOACTIVELY CONTAMINATED 10 CFR 20.1906(d)(1)	Contamination Shipment	Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87;	– New or Spent Fuel Shipment Cask arrives with surface contamination in excess of limits.
SHIPPING PACKAGE EXCEEDING EXTERNAL DOSE RATE LIMITS 10 CFR 20.1906(d)(2)	Radiation Dose Rate Shipment	External radiation levels exceeds of the limits of 10 CFR 71.47.	– New or Spent Fuel Shipment Cask arrives with external radiation levels in excess of limits.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FFD

The NRC Region II Administrator must be notified immediately by telephone of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
NRC EMPLOYEE NOT FIT FOR DUTY 10 CFR 26.27(d)	Alcohol Influence Substance NRC employee FFD Fitness for Duty	If HBRSEP has a reasonable belief that an NRC employee may be under the influence of any substance, or unfit for duty...the Region II Administrator must be notified immediately by telephone. During other than normal working hours, the NRC Operations Center via FTS-2000 must be notified.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FFD

The NRC Operations Center via FTS-2000 must be notified immediately by telephone of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
FALSE POSITIVE ERROR ON FFD SPECIMEN 10 CFR 26, Appendix A, Subpart B, 2.8(e)(5)	FFD Fitness for Duty False Positive Specimen Laboratory	Should a false positive error occur on a blind performance test specimen and the error is determined to be an administrative error, HBRSEP shall promptly notify the NRC.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - IAEA

The NRC Director, NRR or Director, NMSS must be notified immediately by telephone of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SURPRISE VISIT OF IAEA OFFICIAL 10 CFR 75.7	IAEA International Atomic Energy Agency Credential	HBRSEP shall immediately communicate by telephone, with respect to the credentials of any other person who claims to be an IAEA representative and shall accept telephone confirmation of such credentials by the Commission.	– Person arrives on site bearing IAEA credentials, who is not accompanied by an NRC employee, and has had no prior confirmation in writing of credentials.

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
DEGRADED SAFETY BARRIERS DISCOVERED WHILE SHUT DOWN 10 CFR 50.72(b)(2)(i)	Shutdown Safety Barrier Fission Product Barriers Degrade Unanalyzed	Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.	– Corrosion of Reactor Coolant System piping found while shutdown (indicative of a material problem that caused abnormal degradation of the RCS pressure boundary). – Significant degradation of Reactor Fuel Rod Cladding identified during testing of fuel assemblies (Reference 2.19).

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ESF OR RPS INITIATION (MANUAL/AUTOMATIC) 10 CFR 50.72(b)(2)(ii)	Manual Automatic Actuation Engineered Safety Feature ESF Valid Clearance Ventilation System Reactor Protection System RPS Reactor Trip	Any event or condition that results in a manual or automatic actuation of any ESF, including the RPS, except when: (A) The actuation results from and is part of a pre-planned sequence during testing or reactor operation; (B) The actuation is invalid and: (1) Occurs while the system is properly removed from service; (2) Occurs after the safety function has been already completed; or (3) Involves only the following specific ESFs or their equivalent systems: (i) Not Applicable (ii) Control Room emergency ventilation system; (iii) Reactor building ventilation system; (iv) Fuel building ventilation system; or (v) Auxiliary building ventilation system.	– Safety Injection System actuation (also see Emergency Plan Procedures) – Reactor Trip (Manual or Automatic). – EDG start due to a valid undervoltage trip signal on emergency bus E1 or E2 – A single train of Containment Isolation actuates. – A valid signal for Containment Ventilation Isolation occurs. All ESF actuations are reportable except the following three categories. 1) An invalid ESF or RPS actuation occurs when the system is already properly removed from service if all requirements of plant procedures for removing equipment from service have been met. This includes required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers. 2) An invalid ESF or RPS actuation occurs after the safety function has already been completed (e.g., an invalid containment isolation signal while the containment isolation valves are already closed, or an invalid actuation of the RPS when all rods are fully inserted). 3) ESF actuations that are caused by non-ESF systems may be excluded because these are not considered ESF actuations of safety significance. (Reference 2.19) 4) Invalid actuations of the listed ventilation systems.

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
CONDITION THAT COULD PREVENT FULFILLMENT OF SAFETY FUNCTIONS 10 CFR 50.72(b)(2)(iii)	Loss of Safety Function Residual Heat Mitigation Shutdown Generic Setpoint Drift Engineering Evaluation Operability Determination Common Mode Failure	Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition. (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident.	– Loss (inoperability) of both Trains, e.g., ECCS, Low Temperature Overpressure Protection System, or Lake Robinson water level below LCO 3.7.8 limit. 1) Overpressurization of the RCS (if Overpressure Protection System fails to perform its intended function) - Loss of one Train of required equipment, and the cause of the failure could fail the other train, and there is a reasonable expectation that the other train would not fulfill its safety function if required. 1) Contaminated lubrication fluid degrades SI Pump operation (a single condition could prevent fulfillment of a safety function if both trains could be reasonably expected to be inoperable). 2) EDG Air Start Solenoids (if it demonstrates a design, procedural, or equipment deficiency that could prevent the fulfillment of a safety function, i.e., if both diesels are susceptible to same problem) - Multiple equipment inoperability or unavailability. 1) Generic setpoint drift (if indicative of a generic and/or repetitive problem with switches used in safety systems) 2) Oversized breaker wiring lugs (incompatible pigtails and lugs could cause one or more safety systems to fail to perform their intended functions) – Control Rod failure (if failure prevented the fulfillment of a safety function) – Operator action to inhibit the RPS (actions would prevent fulfillment of a safety function)

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
AIRBORNE RELEASE TO UNRESTRICTED AREA (>20X 10 CFR 20 LIMITS) 10 CFR 50.72(b)(2)(iv)(A)	Airborne Release Unrestricted Public Radioactive Effluent	Any airborne radioactive release that, when averaged over a time period of 1 hour, results in concentrations in unrestricted area that exceeds 20 times the applicable concentration specified in Appendix B to 10 CFR 20, Table 2, Column 1.	– Unplanned gaseous release (if release exceeded 20 times the applicable concentrations specified in Appendix B, Table 2, Column 1 of 10 CFR 20 averaged over a time period of one hour)
LIQUID EFFLUENT RELEASE TO UNRESTRICTED AREA (>20X 10 CFR 20 LIMITS) 10 CFR 50.72(b)(2)(iv)(B)	Liquid Release Unrestricted Public Radioactive Effluent Concentration Discharge	Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentration specified in Appendix B to 10 CFR 20 , Table 2, Column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.	– Radioactive release exceeding TS (if release exceeds 20 times the applicable limit of Appendix B, Table 2, Column 2 of 10 CFR 20 when averaged over one hour)
TRANSPORT OF CONTAMINATED INJURED PATIENT 10 CFR 50.72(b)(2)(v)	Contaminate Injured Person Medical Transport Rescue Hospital	Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment.	– Any event requiring the transport of a radioactively contaminated or potentially contaminated (Reference 2.19) person to an off-site medical facility for treatment

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRESS RELEASES AND GOVERNMENT NOTIFICATIONS 10 CFR 50.72(b)(2)(vi)	News Release Press Radio Television Fatality Environment Public Health and Safety Release	Any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials.	– Any News release concerning - A fatality, - Inadvertent release of radioactively contaminated materials to public areas - unusual or abnormal releases of radioactive effluents, or - Information associated with an Emergency Event except when the ERO is activated (Reference 2.27) – Notification to other government agencies concerning - A fatality on site, - Health and safety of the public or site personnel, - Inadvertent release of radioactively contaminated materials to public areas, - Discovered endangered species kill.

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as possible but not later than 4 hours after the discovery of any of the following events or conditions involving spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - EXPOSURES TO RADIATION OR RADIOACTIVE MATERIALS IN EXCESS OF LIMITS, OR RELEASES IN EXCESS OF LIMITS 10 CFR 72.75(b)(1)	ISFSI Release Exposure Fire Explosion Toxic	Any event that prevents immediate actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits, or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases).	– Explosion or fire involves ISFSI resulting in radiological releases
ISFSI - DEFECT IMPORTANT TO SAFETY 10 CFR 50.72(b)(2)(vii)(A) 10 CFR 72.75(b)(2)	ISFSI Defect Safety	A defect in any spent fuel storage structure, system, or component which is important to safety.	– A defect discovered in the design or construction of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits
ISFSI - REDUCTION IN EFFECTIVENESS 10 CFR 50.72(b)(2)(vii)(B) 10 CFR 72.75(b)(3)	ISFSI Confinement Reduction Effectiveness	A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use.	– Wear or degradation of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits
ISFSI - DEPARTURE FROM LICENSE CONDITION 10 CFR 72.75(b)(4)	ISFSI Emergency Departure Deviation Health and Safety License Condition	An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under 10 CFR 72 when the action is immediately needed to protect the public health and safety and no action consistent with license conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.	– Action taken in an emergency that departs from procedure that is deemed necessary to prevent releases or radiation doses to the public in excess of 10 CFR 20 limits (See PRO-NGGC-0200)

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via FTS-2000 as soon as possible but not later than 4 hours after the discovery of any of the following events or conditions involving spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - TREATMENT OF CONTAMINATED PERSON AT OFFSITE MEDICAL FACILITY 10 CFR 72.75(b)(5)	ISFSI Contaminate Injured Person Medical Transport Rescue Hospital	An event that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination.	– An individual is injured requiring offsite medical treatment and receives contamination from ISFSI(s) that cannot be removed prior to transport
ISFSI - FIRE OR EXPLOSION 10 CFR 72.75(b)(6)	ISFSI Fire Explosion Damage Integrity	An unplanned fire or explosion damaging any spent fuel, or any device, container, or equipment containing spent fuel when the damage affects the integrity of the material or its container	– ISFSI unit is damaged by an external explosion and the integrity of the ISFSI unit is potentially affected

Question: 76

Which of the following conditions would require a One-Hour Notification in accordance with AP-617, Reportability Determination and Notification?

a. A manual reactor trip is actuated from 40% power due to a trip of the running Main Feedwater Pump

b. An automatic safety injection is actuated at 100% power due to an I&C Technician lifting an incorrect lead

c. While at 400°F during a plant cooldown, all warning sirens in Lee County are reported to be out-of-service due to severe weather.

d. While at 400°F during a plant heatup following a refueling outage, the plant is cooled down to Mode 4 to meet a Technical Specification action statement.

Answer:

c. While at 400°F during a plant cooldown, all warning sirens in Lee County are reported to be out-of-service due to severe weather.

SRO BACKUP REPLACEMENT

Question: 39

Given the following conditions:

- The RCS is at 190°F during a plant cooldown.
- A break in the CCW system has resulted in all CCW pumps being tripped.
- All RCPs have been secured.
- Charging Pump 'B' is running, with Charging Pump 'A' secured.
- Charging Pump 'C' is under clearance.
- AOP-~~017~~ 014, Attachment 1, "Emergency Cooling to Charging Pump," has just been started.

Which ONE (1) of the following describes how the Charging Pumps should be configured until emergency cooling is available?

a. All Charging Pumps should be stopped ~~(LOSS OF COOLING TO PUMPS)~~

b. Charging Pumps 'A' and 'B' should be alternately operated at minimum speed every 15 minutes ~~(MINIMIZE HEATING OF PUMPS AND ALLOW COOLDOWN)~~

c. Charging Pump 'B' should be operated at minimum speed ~~(SEAL INJECTION REQUIRED UNTIL RCS < 150°F)~~

d. Charging Pump 'B' should be operated at maximum speed ~~(SEAL INJECTION REQUIRED UNTIL RCS < 150°F, BUT HIGHER FLOW RESULTS IN LESS HEATING OF PUMP)~~

Answer:

d. Charging Pump 'B' should be operated at maximum speed

Replacement

Question: 39

Given the following conditions:

- The unit is in a refueling outage.
- GP-010, "Refueling", is being implemented.
- **NO** core alterations are in progress.
- At 0800, CCW was isolated to the operating RHR pump seal cooler per OP-201, "Residual Heat Removal System."

Which ONE (1) of the following describes when the RHR pump must be secured?

a. At 0900

b. When core alterations are resumed

c. When RCS temperature exceeds 140 °F.

d. When RHR pump discharge temperature exceeds 135 °F.

Answer:

d. When RHR pump discharge temperature exceeds 135 °F.

QUESTION NUMBER: 39

TIER/GROUP: ***RO*** ***SRO*** 1/1

K/A: 026AA2.04

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW

K/A IMPORTANCE: ***RO*** ***SRO*** 2.9

10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: RHR-05

DESCRIBE the performance and design attributes of the major RHR System components.

REFERENCES: OP-201

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number OP-201-06 001

JUSTIFICATION:

a. Plausible since TS allows RHR to be removed from operation for this period of time while refueling, but RHR pump seal cooling is not based on this limit.

b. Plausible since RHR operability is required during core alterations, but pump operability is not affected by CCW flow to the seal coolers until seal cooler temperature increases enough to cause RHR pump concerns.

c. Plausible since this is the upper administrative limit for RCS temperature during refueling, but RHR pump operation is not based on this limit.

d. **CORRECT** If CCW is not available to the RHR pump seal coolers, the RHR pumps shall not be operated with pump discharge temperature greater than 135 °F.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of precaution and limitations for RHR pump operation

REFERENCES SUPPLIED:

5.7 When both RHR-757C and RHR-757D are open, 3750 gpm total per running pump as read from FI-605, FI-608A and
FI-608B shall not be exceeded, except as allowed/required by approved test procedures for which total flowrates may be as high as 4200 gpm for one pump or 8400 gpm for two pumps.

5.8 When running RHR Pumps with SI-863A and/or SI-863B open, RHR-744A and RHR-744B should be closed to prevent excessive RHR pump runout.

5.9 If CCW is not available to the RHR pump seal coolers, the RHR pumps shall not be operated with pump discharge temperature greater than 135 °F. With CCW available to the RHR pump seal coolers there is no time limit for running a single pump with flow only through the heatup recirculation line. It will be necessary to rotate the RHR pumps to avoid exceeding the 50 °F ΔT limit between RHR loops as stated in GP-007.

5.10 RHR pump flowrates of less than 2,800 gpm have been shown to increase pressure and flow fluctuations and should be avoided when plant conditions permit. This does not apply during recirculation operation. (ACR 91-078)

5.11 With the exception of swapping running pumps, when RHR is aligned for core cooling, both RHR Pumps should not be run simultaneously on recirculation when forward flow is not established to prevent pump over heating from dead heading of the weaker pump. (CR 98-01791)

5.12 With no flow in the RHR system, an RHR Pump should not be started with FCV-605 in automatic. This could allow runout of the pump before FCV-605 could respond to control flow.

5.13 RHR-750 **AND** RHR-751 shall not be operated (electrically or manually) in a dry condition. Damage to the valve seat may result without water to provide lubrication.

5.14 The principles of **ALARA** shall be used in planning and performing work and operations in the Radiation Control Area.

5.15 This procedure has been screened IAW PLP-037 criteria and determined not applicable to PLP-037.

Question: 40

Given the following conditions:

- A reactor trip has occurred.
- A transition has been made from PATH-1 to EPP-4, "Post Trip Response."
- APP-004-B2, PZR LO PRESS TRIP, is flashing.
- RCS Pressure is 1825 psig and lowering slowly.
- Pressurizer level is 13% and decreasing at 2% per minute.
- RCS Temperature is 553 °F and lowering slowly.
- 'B' and 'C' Charging Pumps are running.

Which ONE (1) of the following describes the instructions the CRSS should give to the Reactor Operator?

a. Start **BOTH** Safety Injection Pumps

b. Verify Letdown isolated and start 'A' Charging Pump

c. Initiate Safety Injection

d. Stabilize RCS temperature

Answer:

c. Initiate Safety Injection

QUESTION NUMBER: 40
TIER/GROUP: ***RO*** ***SRO*** 2/2
K/A: 006A2.12

Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences: Conditions requiring actuations

K/A IMPORTANCE: ***RO*** ***SRO*** 4.8
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: FOLDOUT A-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of events as directed by steps in EPP-Foldouts.

REFERENCES: EPP-Foldouts

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number FOLDOUT A-03 001

JUSTIFICATION:

a. Plausible since this is a required action in Foldout B, but in Foldout A SI initiation is required if unable to maintain pressurizer level.

b. Plausible since starting an additional charging pump may help keep level above 10%, but PATH-1 directs starting only 2 charging pumps and leakage beyond this capacity requires SI initiation.

c. **CORRECT** In Foldout "A", required to initiate Safety Injection if unable to maintain Pressurizer level greater than 10%.

d. Plausible since lowering RCS temperature will result in lowering level, but RCS temperature is still above no-load temperature and level should be maintained after no-load temperature achieved.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of Foldout page criteria for initiating safety injection

REFERENCES SUPPLIED:

CONTINUOUS USE

FOLDOUT A

(Page 1 of 6)

1. RCP TRIP CRITERIA

   IF BOTH conditions below are met, THEN stop all RCPs:

   - SI Pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW TO THE CORE
   - RCS Subcooling - LESS THAN 35°F [55°F]

2. SI ACTUATION CRITERIA

   IF EITHER condition below occurs, THEN Actuate SI and Go To PATH-1, Entry Point A:

   - RCS Subcooling - LESS THAN 35°F [55°F]
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [32%]

3. SPRAY ACTUATION CRITERIA

   IF a valid CV Spray Signal occurs, THEN dispatch an Operator to the Safeguards Racks to block CV Spray as follows: (A screwdriver is available locally for opening the panels)

   a. At the front of Safeguards Relay Rack 51, rotate Test Switch Number 5 (PC-951A) to the PUSH TO TEST position.

   b. At the front of Safeguards Relay Rack 63, rotate Test Switch Number 5 (PC-951A) to the PUSH TO TEST position.

4. AFW SUPPLY SWITCHOVER CRITERIA

   IF CST level decreases to less than 10%, THEN switch to backup water supply using OP-402, Auxiliary Feedwater System.

Question: 56

Given the following conditions:

- GP-003, "Normal Plant Startup from Hot Shutdown to Critical," is being performed.
- The reactor is **NOT** critical.
- Two (2) doublings have been performed.
- The ECP extrapolated from the 1/M plot is 44 steps on CBD.
- The minimum calculated critical position for the startup is 62 steps on CBD and the maximum calculated critical position is 174 steps on CBD.

Which ONE (1) of the following choices describes the correct actions to be taken?

a. Add 250 gallons of boric acid to the RCS

b. Insert all Control Banks and Shutdown Bank B rods

c. Continue the reactor startup and perform an additional doubling

d. Perform a normal reactor shutdown per GP-006

Answer:

c. Continue the reactor startup and perform an additional doubling

QUESTION NUMBER: 56
TIER/GROUP: *RO* *SRO* 2/1
K/A: 001A2.12

Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erroneous ECP calculation

K/A IMPORTANCE: *RO* *SRO* 4.2
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 6

OBJECTIVE: GP-003-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to events as directed in GP-003.

REFERENCES: GP-003

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number GP-003-03 020

JUSTIFICATION:

a. Plausible since this action would be performed if the reactor actually achieved criticality below the minimum control rod insertion limit, but additional doublings should be performed.

b. Plausible since this action would be performed if the reactor actually achieved criticality below the minimum control rod insertion limit, but additional doublings should be performed.

c. **CORRECT** A minimum of three doublings are performed before actually achieving criticality unless the predicted position is outside the +/- 500 pcm position, when a fourth doubling would be performed before achieving criticality.

d. Plausible since this action would be performed if the reactor actually achieved criticality below the minimum rod position for criticality (-500 pcm position), but additional doublings should be performed.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of procedural requirements providing guidance for reactor startup activities

REFERENCES SUPPLIED:

5.8.3 Shutdown Bank "A" shall be at the fully withdrawn position whenever reactivity is being changed by Boron or Xenon changes, RCS temperature changes, or Control Rods, other than Shutdown Bank "A". The following exceptions to this rule may be applied:

> **NOTE:** The COLR identifies the required Shutdown Margin (SDM) based on plant conditions. The required Boron Concentration can be determined using Powertrax or the Plant Curve Book using the SDM identified in the COLR.

1. The RCS has been borated and confirmed by sampling, to be at least at the Boron Concentration needed to provide the required SDM for MODE 3 and is being maintained at MODE 3. Approval of the Manager - Operations, or his designated alternate, shall be given for Shutdown Bank "A" to be inserted.

2. The RCS has been borated and confirmed by sampling, to be at least at the Boron Concentration needed to provide the required SDM for MODE 5. Approval of the Manager - Operations, or his designated alternate, shall be given for Shutdown Bank "A" to be inserted.

5.8.4 **IF** Shutdown Bank "A" cannot be withdrawn, **THEN** the RCS shall be borated as required IAW Step 5.8.3.

5.9 Precautions During Approach to Critical:

5.9.1 Startup Rate shall **NOT** be permitted to exceed 1.0 decade/minute as read on the STARTUP RATE METER.

5.9.2 An Inverse Count Rate Ratio Plot (I/M) with a minimum of four data points (including baseline data point which is taken after Shutdown Banks "A" and "B" are fully withdrawn) shall accompany the Reactor startup.

5.9.3 The Reactor Operator may shutdown the Reactor if the predicted critical rod position from the 1/M plot falls outside the +/-500 pcm positions. (Project 97-00161)

5.10 Whenever possible, the Steam Dump Valves should be used for temperature control instead of Steam Line PORVs.

INIT

8.2.20 Withdraw Shutdown Bank "B" as follows:

1. To ensure sufficient time is available to achieve criticality using the current ECP and thereby satisfy ITS SR 3.1.6.1, check that Attachment 10.1 was completed less than 2 hours ago. ____

2. Select SBB on the Rod Bank Selector switch. ____

3. Withdraw Shutdown Bank "B" to 225 steps while performing the checks of Attachment 10.3. ____

4. **WHEN** Shutdown Bank "B" is greater than 20 steps, **AND** MODE 2 has not been declared, **THEN** perform the following:

   a. Make a plant announcement that MODE 2 has been entered. ____

   b. Use the PMODE function to change the ERFIS Mode indication to display MODE 2. ____

5. Verify the Source Range count rate stabilizes **AND** does **NOT** increase in an unexpected manner. ____

> **NOTE:** A minimum of four inverse count rate ratio (1/M) data points are required on the approach to criticality. The data points should be taken each time the count rate approaches a value that is double the previous stable data point. This is referred to as "doubling". The first data point, Reference Count Rate (CR_0), is obtained after Shutdown Bank "B" has been fully withdrawn.
>
> The Audio Count Rate VOLUME AND AUDIO MULTIPLIER should be adjusted as the count rate increases to maintain a distinguishable audible count rate.

8.2.21 **WHEN** Shutdown Bank "B" is fully withdrawn **AND** the count rate is stable, **THEN** record the time **AND** Reference Count Rate (CR_0) on Attachment 10.2. ____

8.2.23 (Continued) INIT

3. Transfer NR-45 from the selected Source Range channel to the highest reading Intermediate Range **AND** Power Range channels.

IR N-________

PR N-________ ____

> **NOTE:** The approach to criticality should take approximately four doublings of the indicated reference count rate (CR_0) under ideal conditions. The target count rate is intended to serve as a known stable reactivity state suitable for data taking and criticality predictions.
>
> It is **NOT** necessary, **AND** impractical, to attempt to stabilize at exactly double the previous count rate, therefore the use of a "target count rate" (as applied to each doubling of the count rate) is intended to allow the Operator to stabilize the core as close as is practical to the "doubling" count rate without excessive rod motion.
>
> APP-005-F2, ROD BOTTOM ROD DROP, will extinguish when Control Bank "A" is above 20 steps.

8.2.24 Withdraw control rods to achieve the target count rate determined in Attachment 10.2 as follows:

1. Select "M" on the Rod Bank Selector switch. ____
2. Withdraw Control Rods until count rate is approximately equal to the target count rate while performing the checks and verifications of Attachment 10.3. ____
3. Verify the count rate stabilizes **AND** does **NOT** increase in an unexpected manner. ____
4. **IF** criticality is indicated, **THEN** Go To Section 8.3. ____

> **NOTE:** Each successive reactivity addition will require less rod motion **AND** a longer time for the count rate to stabilize. The NR-45 trace should be closely monitored and cross-checked against available instrumentation to determine when count rate has stabilized following each successive rod pull to double counts.

8.2.25 **WHEN** rod motion has been stopped **AND** count rate is stable, **THEN** record the required information on Attachment 10.2. ____

INIT

8.3 Critical Operations

8.3.1 Check that Criticality was achieved above the Minimum Rod Position for Criticality **AND** below the Maximum Rod Position for Criticality. ____

8.3.2 **IF** criticality occurs **AND** the Control Rods are BELOW the Minimum Control Rod Insertion Limit, **THEN** perform the following:

1. Shutdown the Reactor as follows:
   - Add 250 gallons of Boric Acid to the RCS. ____
   - Insert **ALL** Control Banks **AND** Shutdown Bank "B". ____
2. Assign a Startup Number. ____
3. Notify the Reactor Engineer of the reactivity anomaly. ____
4. N/A the remainder of this GP-003 **AND DO NOT** continue until the situation is resolved. ____

8.3.3 **IF** the Reactor goes Critical below the Minimum Rod Position for Criticality, **THEN** perform the following:

- Perform a Reactor Shutdown IAW GP-006. ____
- Assign a Startup Number **AND** N/A the remainder of this procedure. ____
- Notify Reactor Engineer of the anomaly. ____

8.3.4 **IF** the Reactor does **NOT** go critical with control rods at the Maximum Rod Position for Criticality, **THEN** perform the following:

- Insert all Control Banks **AND** Shutdown Bank "B". ____
- N/A the remainder of this procedure. ____
- Notify Reactor Engineer of the anomaly. ____

GP-003-03 020

Given the following plant conditions:

- GP-003, Normal Plant Startup from Hot Shutdown to Critical, is in progress.
- The reactor is sub-critical.
- Three doublings have been performed.
- The ECP extrapolated from the 1/M plot is 182 steps on CBD.
- The minimum calculated critical position for the startup is 62 steps on CBD and the maximum calculated critical position is 174 steps on CBD.

Which ONE (1) of the following choices describes the correct actions to be taken?

A. Perform a normal reactor shutdown IAW GP-006, assign a startup number and N/A the remainder of GP-003, and notify Reactor Engineering.

B. Insert all control banks and Shutdown Bank B rods, N/A the remainder of GP-003, and notify Reactor Engineering.

✓C. Perform an additional doubling and see if the extrapolated critical position will fall within the minimum and maximum allowable critical positions

D. IAW GP-003, Maintain current plant conditions, notify Reactor Engineering for guidance.

Question: 57

Given the following conditions:

- A large steam line break occurred while the unit was operating at 100% power.
- After performing the actions of PATH-1, a transition was made to FRP-P.1, "Response to Pressurized Thermal Shock."
- An RCS soak has been initiated.
- RCS temperature has been stable at 360 °F for the past 25 minutes.
- RCS pressure is 450 psig.

Which ONE (1) of the following describes an action that would be permissible during the RCS soak period?

a. Increase SG level by adjusting the AFW flow controllers

b. Increase RHR flow by adjusting the RHR HX Bypass Flow controller

c. Increase subcooling margin by adjusting the Steam Dump controller

d. Increase subcooling margin by energizing pressurizer heaters

Answer:

b. Increase RHR flow by adjusting the RHR HX Bypass Flow controller

QUESTION NUMBER: 57
TIER/GROUP: ***RO*** ***SRO*** 1/1
K/A: WE08EA2.2

Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A IMPORTANCE: ***RO*** ***SRO*** 4.1
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: FRP-P.1-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in FRP-P.1 by explaining the basis of each.

REFERENCES: FRP-P.1
SD-003

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number FRP-P.1-03 011

JUSTIFICATION:

a. Plausible if misconception is that changes can be made within normal limits of control band, but this would result in an RCS cooldown which is not permitted.

b. **CORRECT** Increasing RHR bypass flow will not affect flow through the RHR HX so RCS temperature will remain stable or increase. During the soak period, operations that cause an increase in pressure or a decrease in temperature are not permitted.

c. Plausible if misconception is that changes can be made within normal limits of control band, but this would result in an RCS cooldown which is not permitted.

d. Plausible if misconception is that changes can be made within normal limits of control band, but this would result in an increase in RCS pressure which is not permitted.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of operations on RCS temperature and pressure to ensure compliance with PTS requirements

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
44.	Check Cooldown Rate In RCS Cold Legs - GREATER THAN 100°F IN ANY 60 MINUTE PERIOD	Go To Step 47.
45.	Check RCS Temperature - HAS BEEN STABLE FOR ONE HOUR	Perform the following: a. Do NOT cooldown the RCS. b. Do NOT increase RCS pressure. c. Perform actions of any other procedures in effect which do NOT cooldown the RCS OR increase RCS pressure. d. WHEN RCS temperature has been stable for one hour, THEN Go To Step 46.
46.	Observe The Following Restrictions: a. Maintain RCS pressure and Cold Leg temperature within the limits of Attachment 1, Post Soak Cooldown Limit Curve, during ALL subsequent actions b. Maintain cooldown rate in RCS Cold Legs less than 50°F/hr OR administrative limits of GP-007, Plant Cooldown From Hot Shutdown To Cold Shutdown, which ever is more restrictive in any 60 minute period	
47.	Reset SPDS AND Return To Procedure And Step In Effect	

- END -

5.3 RHR LOOP ISOL SI-862 A & B, RWST to RHR Pump Suction Isolation

Two motor operated valves are provided to isolate the RHR pump suction from the RWST. When lining up for the injection phase, they will be open. In the recirculation phase, they will be closed prior to taking a suction on the CV floor. They are also closed when the RCS is being cooled by the RHR System. To prevent over pressurization of the RWST and other related low pressure piping and to prevent depressurizing the RCS to RWST, these valves are interlocked so they can't be opened unless the RHR System is less than 210 psig (862A and 863A -PC-601A, 862B and 863B -PC-600B). Keyed switches located behind the RTGB remove the control power from these valves during normal operation.

5.4 RHR-FCV-605, RHR HX Bypass Flow

FCV-605 will automatically maintain a preset flowrate through the operating RHR loop (set by operator). It is an air operated, fail closed valve. If FCV-605 did fail closed, all the flow would be directed through the RHR heat exchanger. This may result in Cooldown rate being higher than desired until valve control was obtained. This problem is addressed in AOP-020, Loss of RHR Cooling.

FCV-605 works in conjunction with hand control valve RHR-HCV-758 and FT-605. HCV-758 is adjusted to increase or decrease flow through the RHR Heat Exchangers to change the Heat up or Cooldown rate. This causes total system flow to be effected and is sensed by FT-605. The flow loop circuitry provides a control signal to FCV-605 which maintains a constant total system flow.

At power, Instrument Air is isolated to FCV-605 (Required by Tech. Specs. when >1000 psig). A portable skid mounted controller is available for use during Post Fire Repairs if FCV-605 control circuits are damaged or inoperable. These procedures would also line up to use the Nitrogen system for motive force and for valve control.

5.5 RHR-HCV-758, RHR HX Discharge Flow

HCV-758 is throttled from RTGB to control Cooldown or Heat up rate by controlling RHR flow through the heat exchanger. It is an air operated valve that fails closed.

At power, Instrument Air is isolated to HCV-758 (Required by Tech. Specs. when >1000 psig). A portable skid mounted controller is available for use during Post Fire Repairs if HCV-758 control circuits are damaged or inoperable. These procedures also allow the use of Nitrogen as a backup for motive force and for valve control.

Question: 58

Given the following conditions:

- Following a loss of all AC, EPP-1, "Loss of All AC Power," is being performed.
- Attachment 5, "Removing Control Power From Safeguard Equipment," has been completed.
- The SGs are being depressurized which results in a Safety Injection signal being actuated.
- The Safety Injection signal is reset after being actuated.
- During the SG depressurization, the Dedicated Shutdown Diesel Generator is started.
- Several minutes later, Emergency Diesel Generator 'A' is started.
- SW Pump 'A' automatically starts.
- SG pressures are stabilized by local operator action.

Plant conditions are now:

- EDG 'A' is running.
- SW Pump 'A' is running.
- **NO** other pumps are running.
- All SI valves are aligned in their pre-trip position.
- RCS pressure is 1400 psig.
- RCS temperature is 492 °F.
- RCS subcooling is 96 °F.
- Pressurizer level is 6%.

Which ONE (1) of the following identifies the procedure to be used for recovery from this condition?

a. EPP-2, "Loss Of All AC Power Recovery Without SI Required"

b. EPP-3, "Loss Of All AC Power Recovery With SI Required"

c. EPP-22, "Energizing Plant Equipment Using Dedicated Shutdown Diesel Generator"

d. EPP-25, "Energizing Supplemental Plant Equipment Using the DSDG"

Answer:

b. EPP-3, "Loss Of All AC Power Recovery With SI Required"

QUESTION NUMBER: 58
TIER/GROUP: ***RO*** ***SRO*** 1/1
K/A: 055 2.4.16

Knowledge of EOP implementation hierarchy and coordination with other support procedures (Station Blackout).

K/A IMPORTANCE: ***RO*** ***SRO*** 4.0
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: EPP-001-08

Given plant conditions EVALUATE the appropriate actions to mitigate the consequences of a complete loss of all AC power as directed by the steps in EPP-1.

REFERENCES: EPP-001

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since no SI equipment has actuated and no valves have repositioned, but EPP-3 is used due to the low level in the pressurizer.

b. **CORRECT** Although no SI equipment has actuated and no valves have repositioned, EPP-3 is used due to the low level in the pressurizer.

c. Plausible since this procedure is performed during the performance of EPP-1, but is not used as a recovery procedure.

d. Plausible since this procedure may be performed as a supplemental procedure during the loss of all AC, but is not used as a recovery procedure.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Analysis of plant conditions following AC power restoration to determine recovery flowpath

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
57.	Select Recovery Procedure:	
	a. RCS subcooling - GREATER THAN 35°F [55°F]	a. Go To EPP-3, Loss Of All AC Power Recovery With SI Required.
	b. PZR level - GREATER THAN 10% [32%]	b. Go To EPP-3, Loss Of All AC Power Recovery With SI Required.
	c. Check SI Equipment - ANY ACTUATED ON AC POWER RECOVERY • Pumps started <u>OR</u> • Valves repositioned	c. Go To EPP-2, Loss Of All AC Power Recovery Without SI Required.
	d. Go To EPP-3, Loss Of All AC Power Recovery With SI Required	

- END -

Question: 59

Given the following conditions:

- The unit is in Mode 3.
- RCS temperature is at no-load Tavg.
- RCS pressure is 2235 psig.
- RCS gross activity is < 100/E-Bar μCi/gm.
- Dose Equivalent Iodine I-131 is 200 μCi/gm.
- These conditions have existed for the past 48 hours.

Given the supplied references, which ONE (1) of the following describes the requirements for these conditions?

a. Power may be increased, but **CANNOT** exceed 44%

b. No-load conditions may be maintained indefinitely, but the unit **CANNOT** be started up

c. RCS temperature must be reduced to < 500 °F within 6 hours

d. Mode 4 conditions must be established within 6 hours

Answer:

c. RCS temperature must be reduced to < 500 °F within 6 hours

QUESTION NUMBER: 59
TIER/GROUP: *RO* *SRO* 1/1
K/A: 076AA2.02

Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS

K/A IMPORTANCE:	*RO*	*SRO*	3.4
10CFR55 CONTENT:	*55.41(b) RO*	*55.43(b) SRO*	2

OBJECTIVE: RCS-12

Given a plant condition and a copy of Technical Specifications, DETERMINE the applicable Technical Specifications requirements for the Reactor Coolant System IAW H. B. Robinson Technical Specifications and Technical Specification Interpretations.

REFERENCES: TS 3.4.16

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number RCS-13 031

JUSTIFICATION:

a. Plausible since power limit for acceptable operation for this value of DEQ I-131 is 44%, but must have been restored below 1.0 uCi/gm within 48 hours.

b. Plausible since power operations would not be permitted since the time period for restoration within limits has expired, but must reduce RCS temperature within 6 hours.

c. CORRECT Although DE I-131 is within the limits of TS Figure 3.4.16-1, it is > 1.0 uCi/gm and must have been restored within 48 hours. Since this has not been completed, a cooldown to < 500 °F is required.

d. Plausible since Mode applicability is Modes 1-3, but applicability in Mode 3 is further defined as with Tavg > 500 °F.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 2

Application of conditions to Tech Specs to determine required actions

REFERENCES SUPPLIED: TS 3.4.16

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	------------Note------------ LCO 3.0.4 is not applicable. ----------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	-----------------NOTE----------------- Only required to be performed in MODE 1. -------------------------------------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-----------------NOTE----------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

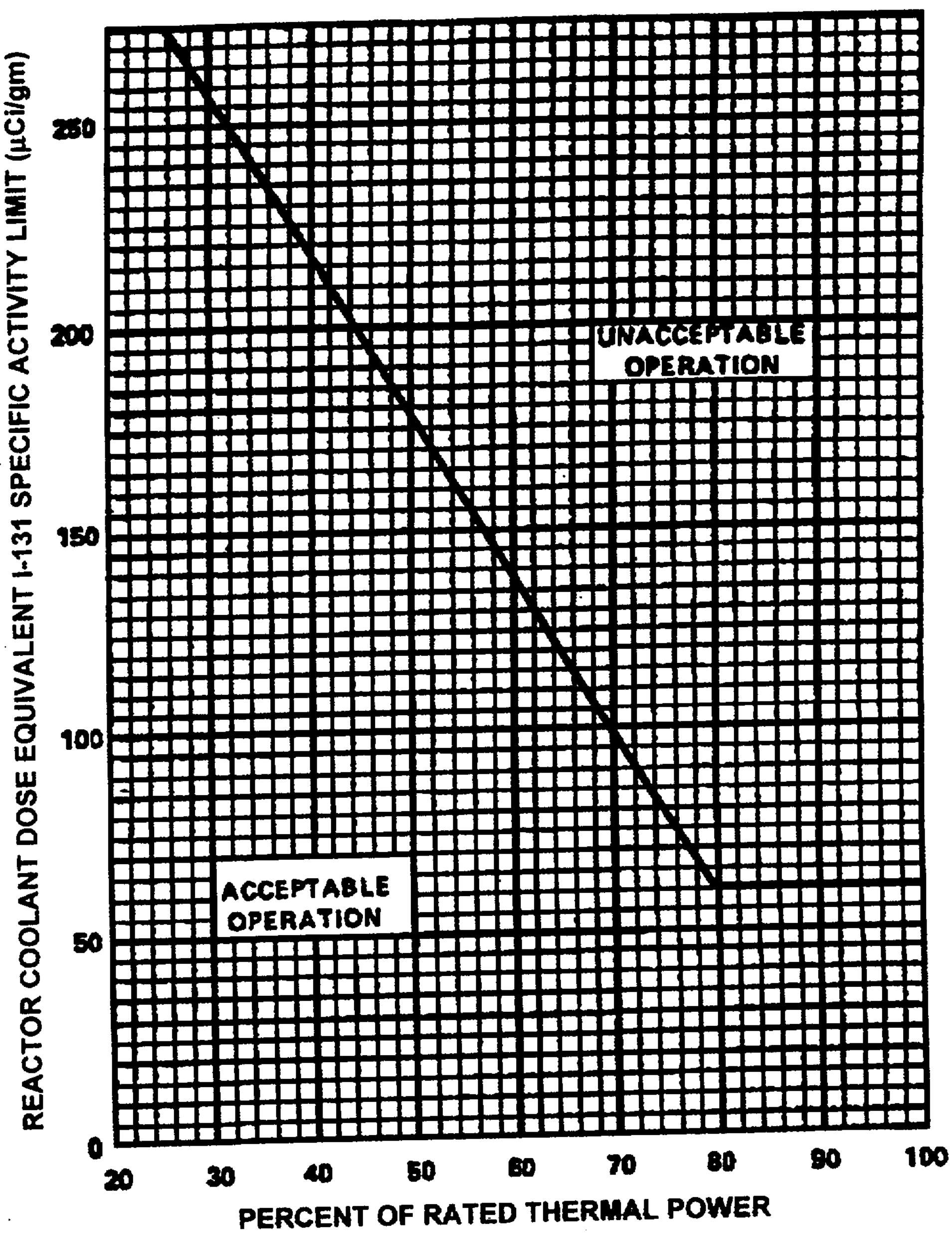


Figure 3.4.16-1
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

Question: 60

Given the following conditions:

- A SGTR has occurred.
- Following the performance of PATH-1 and PATH-2, a transition has been made to EPP-17, "SGTR with Loss of Reactor Coolant: Subcooled Recovery."
- Containment pressure is 0.2 psig.

Given the supplied references, which ONE (1) of the following describes conditions requiring a transition from EPP-17 to EPP-18, "SGTR with Loss of Reactor Coolant: Saturated Recovery"?

a.
- RWST level at 63%
- Containment water level at 6"

b.
- RWST level at 46%
- Containment water level at 124"

c.
- Ruptured SG level at 76%
- RCS Subcooling at 58 °F

d.
- Ruptured SG level at 63%
- RCS Subcooling at 41 °F

Answer:

b.
- RWST level at 46%
- Containment water level at 124"

QUESTION NUMBER: 60
TIER/GROUP: ***RO*** ***SRO*** 1/2
K/A: 038 2.4.4

Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (SGTR).

K/A IMPORTANCE: ***RO*** ***SRO*** 4.3
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: EPP-018-02

RECOGNIZE the selected entry level conditions of EPP-18.

REFERENCES: EPP-017

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EPP-018-02 003

JUSTIFICATION:

a. Plausible since containment sump level is very low, but a transition to EPP-18 is not required until RWST level is below 56% with no increase in sump level.

b. **CORRECT** With RWST level at 46%, minimum required containment water level to continue in EPP-17 is 168". A transition to EPP-18 is required.

c. Plausible since ruptured SG level is a condition for transitioning to EPP-18, but level must be above 84% for management to determine that EPP-18 should be implemented.

d. Plausible since ruptured SG level is a condition for transitioning to EPP-18, but level must be above 84% for management to determine that EPP-18 should be implemented.

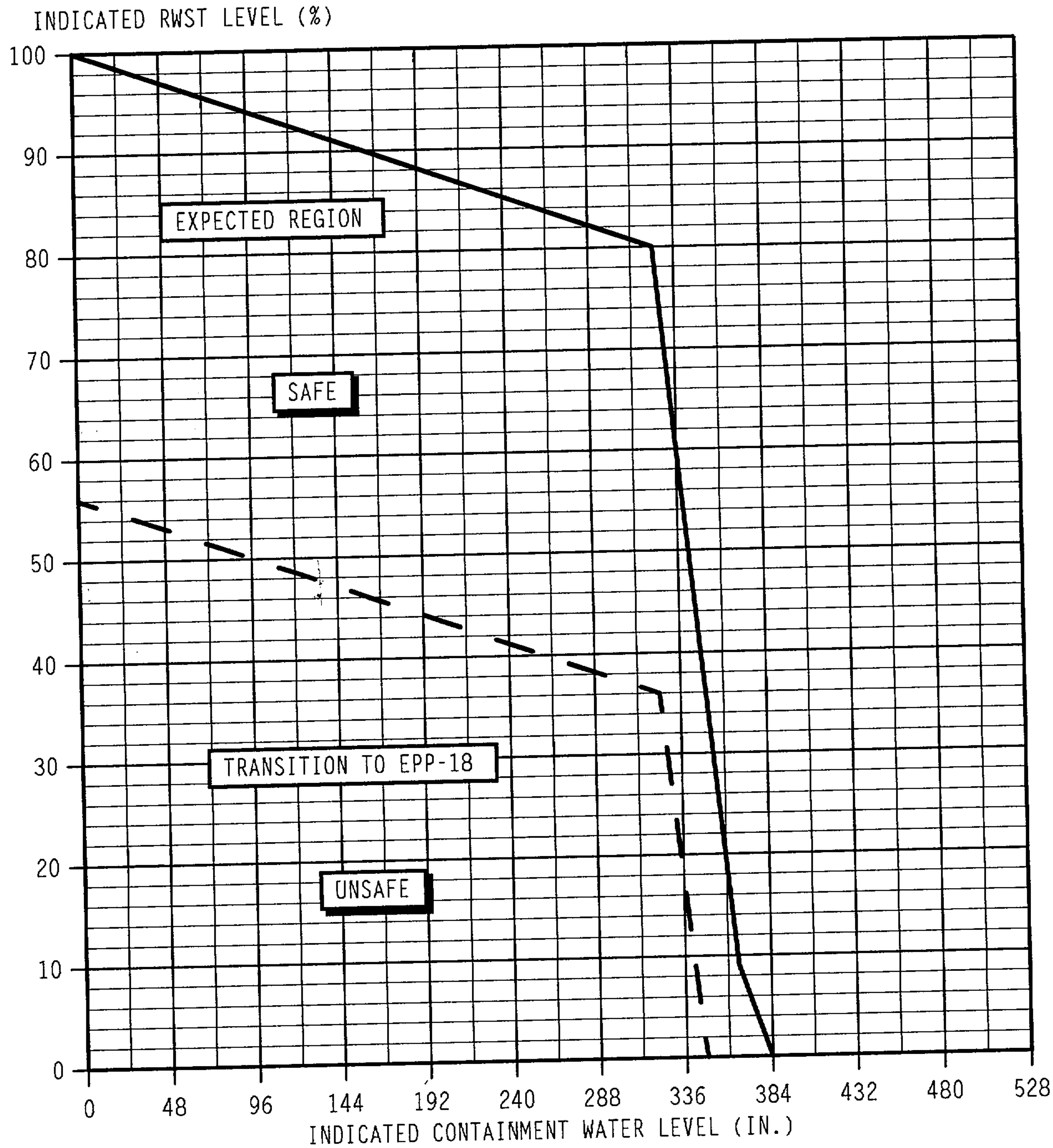
DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Application of given data to EPP curves to determine required action in response to SGTR

REFERENCES SUPPLIED: EPP-17, Attachment 1

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*25.	Determine If Subcooled Recovery Is Appropriate As Follows:	
	a. Check RWST level - GREATER THAN 56%	a. Determine expected CV sump level using Attachment 1, Containment Sump Level Vs. RWST Level. <u>IF</u> CV sump level less than expected, <u>THEN</u> Go To EPP-18, SGTR With Loss Of Reactor Coolant: Saturated Recovery.
	b. Check ruptured S/G level - LESS THAN 84% [82%]	b. Contact Plant Operations Staff to determine if recovery should be completed using EPP-18, SGTR With Loss Of Reactor Coolant: Saturated Recovery based upon the following: • Availability of RVLIS • Capability of steam lines to support the weight of water • Secondary liquid activity
26.	Check RCS Subcooling - GREATER THAN 35°F [55°F]	Go To Step 40.
27.	Check SI And RHR Pump Status: • SI PUMPS - ANY RUNNING <u>OR</u> • RHR PUMPS - ANY RUNNING IN LO HEAD INJECTION MODE	Control charging flow to maintain PZR level. Go To Step 31.
*28.	Check PZR Level - GREATER THAN 71% [60%]	Place all PZR Heaters in OFF. <u>IF</u> PZR level increases above 71% [60%], <u>THEN</u> energize PZR heaters to maintain steam bubble. Observe <u>CAUTION</u> prior to Step 30 and Go To Step 30.

ATTACHMENT 1
CONTAINMENT SUMP LEVEL VS. RWST LEVEL
Page 1 of 1



EPP-018-02 003

Which ONE (1) of the following correctly describes the conditions requiring a transition from EPP-017, SGTR with Loss of Reactor Coolant: Subcooled Recovery to EPP-018, SGTR with Loss of Reactor Coolant: Saturated Recovery?

✓A. RWST level is low and there is no corresponding increase in containment sump level.

B. The ruptured SG level is high and a Station Blackout has occurred.

C. The ruptured SG pressure is high and approaching the safety setpoint.

D. RCS level is low and there is no corresponding increase in the ruptured SG level.

Question: 76

Given the following conditions:

- A reactor trip and safety injection have occurred due to a large break LOCA.
- A transition has been made from PATH-1 to EPP-15, "Loss of Emergency Coolant Recirculation."
- The minimum required Safety Injection flow has been established in accordance with EPP-15.
- RVLIS is now indicating 78% Full Range and increasing slowly.
- Core Exit Thermocouples (CETs) are now indicating 568 °F and decreasing slowly.

Which ONE (1) of the following actions should be taken regarding Safety Injection flow?

a. Maintain flow at its current value

b. Decrease flow until either RVLIS stops increasing OR CETs stop decreasing

c. Increase flow to increase RVLIS level to $\geq$ 90% Full Range

d. Increase flow to decrease CETs to $\leq$ 547 °F

Answer:

a. Maintain flow at its current value

QUESTION NUMBER: 76
TIER/GROUP: *RO* *SRO* 1/2
K/A: WE11EA2.2

Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A IMPORTANCE: *RO* *SRO* 4.2
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 5

OBJECTIVE: EPP-015-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to EPP-15.

REFERENCES: EPP-015

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** Although the minimum flow is more than that required to restore RVLIS level or remove heat from the core and it would conserve RWST inventory to reduce flow, this minimum flow must be maintained.

b. Plausible since reducing flow would conserve RWST inventory, but the minimum flow established must be maintained.

c. Plausible since it would be desirable to raise RCS level, but the minimum required level is 69% to ensure a minimum level above the fuel to remove heat.

d. Plausible since it is desirable to achieve no-load conditions, but as long as temperature is decreasing heat is being removed.

DIFFICULTY:
Comprehensive/Analysis [X] *Knowledge/Recall* [ ] *Rating* 3

Analysis of plant conditions during a loss of recirculation flow to determine required actions

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*38.	Evaluate Effectiveness Of RCS Makeup Flow As Follows:	
	a. Check RVLIS indication - GREATER THAN REQUIRED FROM TABLE	a. Increase RCS makeup flow to maintain RVLIS indication as necessary.

RCP STATUS	**REQUIRED RVLIS INDICATION**
ONE RUNNING	40% DYNAMIC RANGE
NONE RUNNING	69% FULL RANGE

	b. Check Core Exit T/Cs - STABLE OR DECREASING	b. Increase RCS makeup flow to establish T/Cs stable or decreasing.

Question: 77

Given the following conditions:

- The unit is operating at 60% power.
- Chemistry reports that SG 'A' has exceeded Secondary Action Level (SAL) -2 limits for pH and Conductivity.

Which ONE (1) of the describes the actions that must be taken in response to exceeding the SAL-2 limits?

a. Return the parameters to within SAL-1 limits within 100 hours of initiating SAL-2 OR initiate a power reduction to less than 30%

b. Take immediate actions to reduce power to approximately 30% within 8 hours

c. Return the parameters to within its normal value within 100 hours of initiating SAL-2 OR commence a shutdown and cooldown to less than 250 °F

d. Take immediate actions to shutdown and cooldown to less than 250 °F as rapidly as plant constraints permit

Answer:

b. Take immediate actions to reduce power to approximately 30% within 8 hours

QUESTION NUMBER: 77
TIER/GROUP: *RO* *SRO* 3
K/A: 2.1.34

Ability to maintain primary and secondary plant chemistry within allowable limits.

K/A IMPORTANCE:	*RO*		*SRO*	2.9
10CFR55 CONTENT:	*55.41(b) RO*		*55.43(b) SRO*	5

OBJECTIVE: OMM-001-13-008

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to plant chemistry as directed in OMM-001-13.

REFERENCES: OMM-001-13

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number OMM-001-13-03 004

JUSTIFICATION:

a. Plausible since the parameters must be returned within SAL-1 limits within 100 hours, but if they are not SAL-3 entry must be made. A reduction in power to < 30% within 8 hours would still be required.

b. CORRECT Immediate actions are required to reduce power to < 30% while attempting to lower chemistry within limits.

c. Plausible since the parameters must be returned within SAL-1 limits within 100 hours, but if they are not SAL-3 entry must be made. A reduction in power to < 30% within 8 hours would still be required.

d. Plausible since this is the action required for entry into a Primary Action Level (PAL) -3 condition, but a SAL-2 entry requires a power reduction to < 30% within 8 hours.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of required actions for entry into Chemistry Action Levels

REFERENCES SUPPLIED:

8.5 Secondary Action Level (SAL) Responses

8.5.1 The chemical Control Parameters and their limits for SALs are listed in CP-001.

8.5.2 Refer to CP-005, Secondary Chemistry Corrective Action Program, for a detailed description of the requirements for the following three SALs:

1. SAL-1 Response (Mode 1)
   a. Return the parameter to below the SAL-1 Limit within one week of entering SAL-1; **OR**
   b. **IF** the parameter is **NOT** below the SAL-1 Limit within one week of entering SAL-1, **THEN GO TO** SAL-2 for those parameters having SAL-2 Limits; **OR**
   c. **IF** the parameter will **NOT** be below the SAL-1 Limit within one week of entering SAL-1, and SAL-2 is **NOT** entered, **THEN** obtain approval for this deviation in advance from the Robinson Plant, General Manager
2. SAL-2 Response (Mode 1)
   a. Take immediate actions to reduce power **AND** achieve approximately 30% within eight hours of entering SAL-2.
   b. **IF** the control parameter values can be brought below SAL-2, **THEN** the Power Reduction can be terminated. Full power operation can resume when the control parameter value is below the SAL-1 Limit.
   c. Return the control parameter to below the SAL-1 Limit within 100 hours of entering SAL-2; **OR**
   d. **IF** the parameter is **NOT** below the SAL-1 Limit within 100 hours of entering SAL-2, **THEN GO TO** SAL-3 for those parameters having SAL-3 Limits (even if the SAL-3 Limit is not exceeded);**OR**
   e. **IF** the parameter will **NOT** be below its SAL-1 Limit within 100 hours of entering SAL-2, and SAL-3 is **NOT** entered, **THEN** obtain approval for this deviation in advance from the Robinson Plant General Manager.

OMM-001-13-03 004

Given the following plant conditions:

- The plant is at 60% power
- Chemistry reports that S/G "A" has exceeded Action Level (AL) -1 limits for pH and Conductivity

Which ONE (1) of the following statements describes the correct actions concerning a secondary chemistry parameter which exceeds its AL-1 specification with the unit on line at 60% power?

A. Power operations are not restricted until greater than 70% power for secondary AL-1 parameters.

B. Return the parameter to within its normal value within 12 hours of initiating AL-1 OR initiate a power reduction to less than 30%.

C. Return the parameter to within its normal value within 12 hours of initiating AL-1 OR commence a shutdown and cooldown to less than 350EF.

✓D. Return the parameter to within its normal value within one week of initiating AL-1 OR initiate AL-2 for those parameters having AL-2 values.

Question: 78

Given the following plant conditions:

- The unit is operating at 100% power.
- A plant transient occurs.
- Pressurizer pressure stabilizes at 1950 psig.

Technical Specification 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," must be entered and pressurizer pressure must be restored above 2205 psig within 2 hours if the transient lowers power to ...

a. 73% over a 5 minute period.

b. 88% over a 5 second period.

c. 90% over a 3 minute period.

d. 77% over a 3 second period.

Answer:

c. 90% over a 3 minute period.

QUESTION NUMBER: 78
TIER/GROUP: ***RO*** ***SRO*** 1/2
K/A: 027AA2.04

Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Tech-Spec limits for RCS pressure

K/A IMPORTANCE: ***RO*** ***SRO*** 4.3
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 2

OBJECTIVE: PZR-13

Given a plant condition and a copy of Technical Specifications, DETERMINE the applicable Technical Specifications requirements for the PZR and PRT System IAW H. B. Robinson Technical Specifications and Technical Specification Interpretations.

REFERENCES: TS 3.4.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number PZR-13 008

JUSTIFICATION:

a. Plausible since this power ramp exceeds 5% per minute so the TS is not applicable, but a common misconception is that the TS does not apply during any transient.

b. Plausible since this step change exceeds 10% so the TS is not applicable, but a common misconception is that the TS does not apply during any transient.

c. **CORRECT** The TS is applicable and pressure must be restored within 2 hours if the ramp does not exceed 5% per minute or a step change of greater than 10% does not occur.

d. Plausible since this step change exceeds 10% so the TS is not applicable, but a common misconception is that the TS does not apply during any transient.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of TS requirements for pressurizer pressure

REFERENCES SUPPLIED:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2205 psig;

b. RCS average temperature ≤ 579.4°F; and

c. RCS total flow rate ≥ 97.3 x 10^6 lbm/hr.

APPLICABILITY: MODE 1.

-----NOTE-----
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

Pressurizer Pressure Transmitter Failure

(Page 1 of 1)

NOTE

Steps 1 through 4 are Immediate Action Steps.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Either PZR PORV - OPEN	Go To Step 3.
2.	Close The Open PORV • PCV-456 • PCV-455C	Close the associated PORV BLOCK Valve: • PCV-456 - RC-535 • PCV-455C - RC-536
3.	Check PT-444 - FAILED TRANSMITTER	Go To Step 6.
4.	Control PZR Pressure Controller PC-444J As Follows: a. Place PC-444J in MAN b. Restore PZR Pressure to the desired control band	
5.	Verify PCV-455C in AUTO	
6.	Verify Selector Switch PM-444 - SELECTED TO THE OPERABLE CHANNEL • REC 444 • REC 445	
7.	Go To Procedure Main Body, Step 2	

- END -

PZR-13 008

Given the following plant conditions:

- Unit is initially in a normal 100% power lineup
- Pressurizer PI-444 fails high.
- The operators respond per AOP-025 and stabilize the plant pressure at 1950 psig.
- Both PORVs indicate closed

Which ONE (1) of the following describes the appropriate actions to comply with Technical Specifications?

A. Close and remove power from associated block valve within one hour, restore RCS pressure to > 2000 psig within 1 hours.

B. Close and maintain power to associated block valve within one hour, restore RCS pressure to
> 2000 psig within 2 hours.

C. Close and remove power from associated block valve within one hour, restore RCS pressure to > 2205 psig within 1 hours.

✓D. Close and maintain power to associated block valve within one hour, restore RCS pressure to
> 2205 psig within 2 hours.

Question: 79

Given the following conditions:

- A seismic event has occurred.
- A reactor trip and safety injection have occurred following a SGTR.
- A transition is being made from PATH-1 to PATH-2 and the CRSS is conducting a shift brief.
- The following have occurred as a result of the seismic event:
  - A service water header break has occurred.
  - All instrument air compressors have tripped.
  - A fire header break has occurred inside containment.

Which ONE (1) of the following procedures should the CRSS direct an extra operator to perform while PATH-2 is being performed?

a. AOP-017, "Loss of Instrument Air"

b. AOP-021, "Seismic Disturbances"

c. AOP-022, "Loss of Service Water"

d. AOP-032, "Response to Flooding from the Fire Protection System"

Answer:

a. AOP-017, "Loss of Instrument Air"

Replacement

QUESTION NUMBER: 79
TIER/GROUP: *RO* *SRO* 3
K/A: 2.4.16

Knowledge of EOP implementation hierarchy and coordination with other support procedures.

K/A IMPORTANCE: *RO* *SRO* 4.0
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 5

OBJECTIVE: OMM-022-08

DEMONSTRATE an understanding of selected steps, cautions, and notes in OMM-022 by explaining the basis of each.

REFERENCES: OMM-022

SOURCE: *New* ☒ *Significantly Modified* ☐ *Direct* ☐

Bank Number NEW

JUSTIFICATION:

a. CORRECT Concurrent AOPs for implementation while in the EOP network include AOP-005, AOP-014, AOP-017, and AOP-018.

b. Plausible since it would be desirable to respond to the event which caused all of the failures, but AOP-021 is not considered a concurrent AOP.

c. Plausible since it would be desirable to respond to the loss of service water to ensure availability, but AOP-022 is not considered a concurrent AOP.

d. Plausible since it would be desirable to prevent flooding inside containment due to potential for LOCA dilution, but AOP-032 is not considered a concurrent AOP.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of plant conditions and knowledge of which AOPs are acceptable to perform concurrently with emergency procedures

REFERENCES SUPPLIED:

Question: 79

Given the following conditions:

- A reactor trip and safety injection have occurred following a SGTR.
- A transition is being made from PATH-1 to PATH-2 and the CRSS is conducting a shift brief.

Which ONE (1) of the following procedures would be appropriate to perform concurrently with PATH-2 following the crew brief?

a. AOP-003, "Malfunction of Reactor Makeup Control," due to an inability to establish normal boration flow

b. AOP-017, "Loss of Instrument Air," due a loss of all instrument air compressors

c. AOP-031, "Operation with High Switchyard Voltage," due to WEST 115 KV BUS VOLTAGE indicating 120.2 KV

d. AOP-032, "Response to Flooding from the Fire Protection System," due to a fire header break inside containment.

Answer:

b. AOP-017, "Loss of Instrument Air," due a loss of all instrument air compressors

QUESTION NUMBER: 79
TIER/GROUP: *RO* *SRO* 3
K/A: 2.4.16

Knowledge of EOP implementation hierarchy and coordination with other support procedures.

K/A IMPORTANCE: *RO* *SRO* 4.0
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 5

OBJECTIVE: OMM-022-08

DEMONSTRATE an understanding of selected steps, cautions, and notes in OMM-022 by explaining the basis of each.

REFERENCES: OMM-022

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since it would be desirable to establish normal boration control as part of the SGTR recovery actions, but AOP-003 is not considered a concurrent AOP.

b. **CORRECT** Concurrent AOPs for implementation while in the EOP network include AOP-005, AOP-014, AOP-017, and AOP-018.

c. Plausible since it would be desirable to establish normal switchyard voltage to protect equipment from abnormal voltages, but AOP-031 is not considered a concurrent AOP.

d. Plausible since it would be desirable to prevent flooding inside containment due to potential for LOCA dilution, but AOP-032 is not considered a concurrent AOP.

DIFFICULTY:
Comprehensive/Analysis [ ] ***Knowledge/Recall*** [X] ***Rating*** 3

Knowledge of which AOPs are acceptable to perform concurrently with emergency procedures

REFERENCES SUPPLIED:

3. Non SPDS functions of ERFIS do not meet all of the same qualifications as SPDS, therefore these functions should **NOT** be relied on for sole indication during use of the EOP Network.

8.3.14 Interface Between EOP Network and AOPs/Concurrent AOPs

1. Events which result in utilization of AOPs may later deteriorate to the point of implementing the procedures of the EOP Network. When this occurs, the potential exists for equipment to be improperly utilized and for resources to be unnecessarily diluted by continuing the subsequent actions of AOPs in effect or implementing AOPs which may become applicable while trying to concurrently proceed through the EOP Network.

2. With the exception of concurrent AOPs, the immediate and subsequent actions of AOPs need not be continued while within the EOP Network since the procedures of the EOP Network have been constructed to address critical safety functions without these AOPs.

3. The following AOPs are considered concurrent AOPs and should be performed while in the EOP Network:

   - AOP-005
   - AOP-014
   - AOP-017
   - AOP-018

4. In the case of the above referenced AOPs, it is expected that the CRSS will continue with the EOPs while another licensed operator implements the AOP after any applicable immediate actions of the EOPs have been completed. The operator performing the AOP will notify the CRSS and RTGB operator of all RTGB controls to be manipulated and/or local actions to be taken which could impact the performance of the EOPs.

Question: 80

Given the following conditions:

- A Component Cooling Water train was declared inoperable on March 1st, at 0530.
- At 0330 on March 4th, a Technical Specifications required shutdown was commenced.
- It is currently 0400 on March 4th.
- The unit is currently at 62% power.
- System Engineering has just notified the Control Room that a generic issue requires declaring ALL AFW pumps inoperable.
- They estimate that it will be approximately 12 hours before any AFW pump will be capable of being declared operable.

In accordance with Technical Specifications, which ONE (1) of the following describes the actions required?

a. Be in MODE 3 by 0930

b. Be in MODE 3 by 1100

c. Be in MODE 3 by 1130

d. Maintain MODE 1 until at least one AFW pump is declared operable

Answer:

d. Maintain MODE 1 until at least one AFW pump is declared operable

QUESTION NUMBER: 80
TIER/GROUP: ***RO*** ***SRO*** 2/1
K/A: 061 2.1.12

Ability to apply technical specifications for a system (AFW).

K/A IMPORTANCE: ***RO*** ***SRO*** 4.0
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 2

OBJECTIVE: AFW-13

Given a plant condition and a copy of Technical Specifications, DETERMINE the applicable Technical Specifications requirements for the AFW System IAW H. B. Robinson Technical Specifications and Technical Specification Interpretations.

REFERENCES: TS 3.7.4
TS 3.7.6

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AFW-13 009

JUSTIFICATION:

a. Plausible since CCW TS actions require the plant be placed in Mode 3 within 6 hours if the inoperable train cannot be restored to operable within 72 hours. This time is 6 hours after the shutdown started, but the AFW condition suspends the action.

b. Plausible since TS 3.0.3 actions require the plant be placed in Mode 3 within 7 hours if an LCO and its actions cannot be met. This time is 7 hours after receiving the AFW report, but an action does exist for this condition.

c. plausible since CCW TS actions require the plant be placed in Mode 3 within 6 hours if the inoperable train cannot be restored to operable within 72 hours. This time is 6 hours after the 72 hours are completed, but the AFW condition suspends the action.

d. **CORRECT** The unit is in a seriously degraded condition with no safety related means for conducting a cooldown. LCO 3.0.3 and all other LCO required actions requiring Mode changes are suspended until one AFW pump and flow path are restored to operable status.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of Technical Specification requirements when all AFW pumps are inoperable

REFERENCES SUPPLIED: TS 3.7.4 and TS 3.7.6

3.7 PLANT SYSTEMS

3.7.6 Component Cooling Water (CCW) System

LCO 3.7.6 Two CCW trains powered from emergency power supplies shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required CCW train inoperable.	A.1	--------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4," for residual heat removal loops made inoperable by CCW. -------------------- Restore required CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

3.7 PLANT SYSTEMS

3.7.4 Auxiliary Feedwater (AFW) System

LCO 3.7.4 Four AFW flow paths and three AFW pumps shall be OPERABLE.

---------------------------NOTE---------------------------
Only one AFW flow path with one motor driven pump is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being used for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AFW pump inoperable in MODE 1, 2, or 3. OR One or two AFW flow paths inoperable in MODE 1, 2, or 3.	A.1 Restore AFW pump or flow path(s) to OPERABLE status.	7 days AND 8 days from discovery of failure to meet the LCO
B. Two motor driven AFW pumps inoperable in MODE 1, 2, or 3. OR Three motor driven AFW flow paths inoperable in MODE 1, 2, or 3.	B.1 Restore one motor driven AFW pump or one flow path to OPERABLE status.	24 hours AND 8 days from discovery of failure to meet the LCO

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	18 hours
D. Steam driven AFW pump or flow path inoperable in MODE 1, 2, or 3. <u>AND</u> One motor driven AFW pump or flow path inoperable in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	18 hours
E. Four AFW flow paths inoperable in MODE 1, 2, or 3. <u>OR</u> Three AFW pumps inoperable in MODE 1, 2, or 3.	E.1	---------NOTE------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW pump and flow path are restored to OPERABLE status. -------------------- Initiate action to restore one AFW pump and flow path to OPERABLE status.	Immediately
F. Required AFW pump and flow path inoperable in MODE 4.	F.1	Initiate action to restore AFW pump and flow path to OPERABLE status.	Immediately

AFW-13 009

Given the following plant conditions AND a copy of Tech Specs:

- Reactor Power at 100%
- SDAFW Pump is out of service for maintenance
- V2-16A,B and C (MDAFW to S/Gs) have just been declared inoperable due to MOV issues

Which ONE (1) of the following best describes what ACTION(S) is required?

✓A. LCO 3.0.3 is not applicable. Initiate action to restore one AFW pump and flowpath to OPERABLE status Immediately.

B. Initiate power reduction to MODE 2 immediately. Restore AFW pump or flow path(s) to OPERABLE status within 7 days AND 8 days from discovery of failure to meet LCO.

C. Take action immediately to place unit in MODE 4. Restore AFW pump or flow path(s) to OPERABLE status within 24 hours AND 8 days from discovery of failure to meet LCO.

D. Enter LCO 3.0.3. Action shall be initiated within 1 hour to place the unit, as applicable, in MODE 3 within 7 hours; MODE 4 within 13 hours.

Question: 96

Given the following conditions:

- The unit is in Mode 2 with a reactor startup being performed.
- Shutdown Bank (SDB) 'B' is at 125 steps, being withdrawn.
- APP-005-A1, SR DET LOSS OF DC, alarms.
- N31 indications - before alarm 1200 cps; after alarm 1300 cps
- N32 indications - before alarm 1300 cps; after alarm 700 cps

Which ONE (1) of the following describes the required action to be taken?

a. Commence a reactor shutdown

b. Trip the Reactor and go to PATH-1

c. Stop rod motion

d. Drive SDB "B" rods in to <20 steps

Answer:

c. Stop rod motion

QUESTION NUMBER: 96
TIER/GROUP: ***RO*** ***SRO*** 1/2
K/A: 032AA2.01

Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Normal/abnormal power supply operation

K/A IMPORTANCE: ***RO*** ***SRO*** 2.9
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 2

OBJECTIVE: NIS-012

STATE the Technical Specification Limitations for the Nuclear Instrumentation System. Include the bases.

REFERENCES: GP-003
TS 3.3.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number ITS 001

JUSTIFICATION:

a. Plausible since this is a conservative action, but only required to stop any positive reactivity additions in progress for one SR inoperable <P6.

b. Plausible since this is a conservative action and is required if both SR channels are inoperable, but only required to stop any positive reactivity additions in progress for one SR inoperable <P6.

c. **CORRECT** Only required to stop any positive reactivity additions in progress for one SR inoperable <P6.

d. Plausible since this is a conservative action, but only required to stop any positive reactivity additions in progress for one SR inoperable <P6.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of Tech Spec requirements for SR channels during a startup

REFERENCES SUPPLIED:

5.0 PRECAUTIONS AND LIMITATIONS

5.1 Before withdrawing any rod bank from the fully inserted position, the group step counters shall be at zero steps for that bank and each individual rod position indicator should indicate within 7.5 inches of the average of its bank position.

5.2 Criticality shall be anticipated at any time when the Shutdown Banks or Control Banks are being withdrawn, or when boron dilution operations are in progress.

5.3 If the count rate on either Source Range channel increases by a factor of two or more during any step involving a boron concentration change, the operation shall be stopped immediately and suspended until a satisfactory evaluation of the situation has been made.

5.4 When the Reactor is subcritical, positive reactivity shall not be added by more than one method at a time. (Exception: Due to the slow insertion rate contributed by the decay of Xenon, positive reactivity addition by the Operator may be performed during periods of Xenon decay.)

5.5 The Reactor will not be made critical until the Hydrogen concentration in the RCS is at least 15 cc/kg of water. (Westinghouse Recommendation, Standard Information Package on Chemistry, Criteria & Specification SIP 5-1, Table 1.5 Note B)

5.6 The following requirements apply to the Source Range Nuclear Instruments when in MODE 2 below P-6: (ITS Table 3.3.1-1 item 4)

- **IF** one Source Range channel becomes inoperable, **THEN** immediately suspend operations involving positive reactivity additions.

- **IF** two Source Range channels become inoperable, **THEN** immediately trip the Reactor and Go To PATH-1.

Table 3.3.1-1 (page 1 of 7)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11	≤ 110.93% RTP	108% RTP (2)
b. Low	$1^{(b)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 26.93% RTP	24% RTP
3. Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 37.02% RTP	25% RTP
	$2^{(d)}$	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 37.02% RTP	25% RTP
4. Source Range Neutron Flux	$2^{(d)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.28 E5 cps	1.0 E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.28 E5 cps	1.0 E5 cps
	$3^{(e)}$, $4^{(e)}$, $5^{(e)}$	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.
(2) The Nominal Trip Setpoint is as stated unless reduced as required by one or more of the following requirements: LCO 3.2.1 Required Action A.2.2; LCO 3.2.2 Required Action A.1.2.2; or LCO 3.7.1 Required Action B.2.
(a) With Rod Control System capable of rod withdrawal, or one or more rods not fully inserted.
(b) Below the P-10 (Power Range Neutron Flux) interlock.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication and alarm.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One channel inoperable.	E.1	Place channel in trip.	6 hours
	OR		
	E.2	Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	2 hours
	OR		
	F.2	Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1	Suspend operations involving positive reactivity additions.	Immediately
	AND		
	G.2	Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1	Restore channel(s) to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6
I. One Source Range Neutron Flux channel inoperable.	I.1	Suspend operations involving positive reactivity additions.	Immediately

(continued)

Question: 97

Given the following conditions:

- An accident has occurred which has resulted in activation of the Emergency Plan.
- A repair team is preparing to enter an area to effect repairs that will protect a piece of valuable company property.
- The dose rate in the area is 15 Rem/hour.

Which ONE (1) of the following identifies the MAXIMUM amount of time that each individual can stay in the area **WITHOUT** exceeding allowable emergency dose limits?

a. 20 minutes

b. 40 minutes

c. 60 minutes

d. 100 minutes

Answer:

b. 40 minutes

QUESTION NUMBER: 97
TIER/GROUP: ***RO*** ***SRO*** 3
K/A: 2.3.4

Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

K/A IMPORTANCE: ***RO*** ***SRO*** 3.1
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 4

OBJECTIVE: OPS-EP-1-01

DEMONSTRATE an understanding of the Robinson Emergency Plan IAW PLP-007

REFERENCES: PLP-007
EPOSC-04

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number HNP-SRO-2000 97

JUSTIFICATION:

a. Plausible since the normal 10CFR20 limits are 5 Rem annual which would only allow 12 minutes, but for this type of emergency the limits are raised to 10 Rem for a single exposure.

b. **CORRECT** The dose limit for protecting valuable company property is 10 Rem. With a dose rate of 25 Rem, an individual can stay in the area for 0.4 hours, or 24 minutes.

c. Plausible since this would be a valid calculation if the limit for this condition were 15 Rem, but the limit is 10 Rem.

d. Plausible since this would be the limit for lifesaving or protection of large populations is 25 Rem, but the limit for this condition is 10 Rem.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Calculation of stay time in high dose area to determine emergency limits not exceeded

REFERENCES SUPPLIED:

5.4.4.3 (Continued)

- 50 to 100 Rem in 1 day - no impairment likely but some physiological changes, including possible temporary blood changes, may occur. Medical observations would be required after exposure.

- 100 to 300 Rem in 1 day - some physical impairment possible. Some lethal exposures possible.

The following subsections describe the criteria to be considered for life-saving and facility protection actions.

a. Lifesaving Actions

In emergency situations that require personnel to search for and remove injured persons or entry to prevent conditions that would probably injure numbers of people, a planned dose shall not exceed limits as outlined below:

Dose Limit Rem TEDE[1]	Activity	Condition
5	All	
10	Protecting valuable property	Lower dose not practicable
25	Lifesaving or protection of large populations	Lower dose not practicable
>25	Lifesaving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved

[1]Doses to the lens of the eye should be limited to three times the stated TEDE value and doses to any other organ (including skin and body extremities) should be limited to ten times the stated TEDE value.

8.4.3 (Continued)

7. Emergency worker exposure guidelines:

   a. Although an emergency situation transcends the normal requirements for limiting Total Effective Dose Equivalent (TEDE) to workers, guideline levels are established for doses that may be acceptable in emergencies. The (TEDE) received by any worker should not exceed established regulatory limits, to the extent practical. Every reasonable effort will be used to ensure that an emergency is handled in such a manner that no worker exceeds these limits, including the administering of radioprotective drugs.

   b. To assure adequate protection of minors and the unborn, the performance of emergency services should be limited to nonpregnant (pregnancy undeclared) adults.

   c. During emergencies, doses (TEDE) to workers should be limited to 5 Rem.

      – Justification for receiving higher exposures must include the presence of conditions that prevent the rotation of workers or other commonly-used dose reduction methods.

      – Except as noted below, the dose resulting from such emergency exposure should be limited to 10 Rem for protecting valuable property, and to 25 Rem for lifesaving activities and the protection of large populations.

      – In this context, the exposure incurred by workers to protect large populations may be considered justified when the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved.

Question: 97

Given the following conditions:

- An accident has occurred which has resulted in activation of the Emergency Plan.
- A repair team is preparing to enter an area to effect repairs that will protect a piece of valuable company property.
- The dose rate in the area is 25 Rem/hour.

Which of the following identifies the **MAXIMUM** amount of time that each individual can stay in the area without exceeding allowable emergency dose limits?

a. 12 minutes

b. 24 minutes

c. 36 minutes

d. 60 minutes

Answer:

b. 24 minutes

Question: 98

Given the following conditions:

- A reactor trip and safety injection have occurred due to a LOCA on the letdown line and a failure of the letdown line to automatically isolate.
- PATH-1 actions are being performed.
- The following conditions currently exist:
- Containment pressure is 7 psig and slowly decreasing.
- Total AFW flow to the intact SGs is 390 gpm.
- 'A' SG level is 6% and slowly increasing.
- 'B' SG level is 12% and slowly increasing.
- 'C' SG level is 14% and slowly increasing.
- RCS pressure is 1765 psig and rapidly increasing.
- Pressurizer level is 29% and stable.
- Core Exit Thermocouples are 530°F and stable.

Which ONE (1) of the following identifies the parameter that is inadequate to permit terminating SI?

a. Subcooling

b. Secondary heat sink

c. RCS pressure

d. RCS inventory

Answer:

d. RCS inventory

QUESTION NUMBER: 98

TIER/GROUP: ***RO*** ***SRO*** 1/1

K/A: 011EA2.11

Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions for throttling or stopping HPI

K/A IMPORTANCE:	***RO***		***SRO***	4.3
10CFR55 CONTENT:	***55.41(b) RO***		***55.43(b) SRO***	5

OBJECTIVE: PATH-1-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in PATH-1 by explaining the basis of each.

REFERENCES: PATH-1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EPP-008-06 001

JUSTIFICATION:

a. Plausible since subcooling is required to terminate SI, but the requirement for adverse containment conditions is 55 °F and subcooling is currently 89 °F.

b. Plausible since secondary heat sink is required to terminate SI, but the requirement for adverse containment conditions is either one SG level above 18% (all are currently below) or total feed flow of greater than 300 gpm and current flow is 390 gpm.

c. Plausible since pressure is required to terminate SI, but the requirement for adverse containment conditions is 1750 psig and stable or increasing and pressure is currently 1765 psig and increasing.

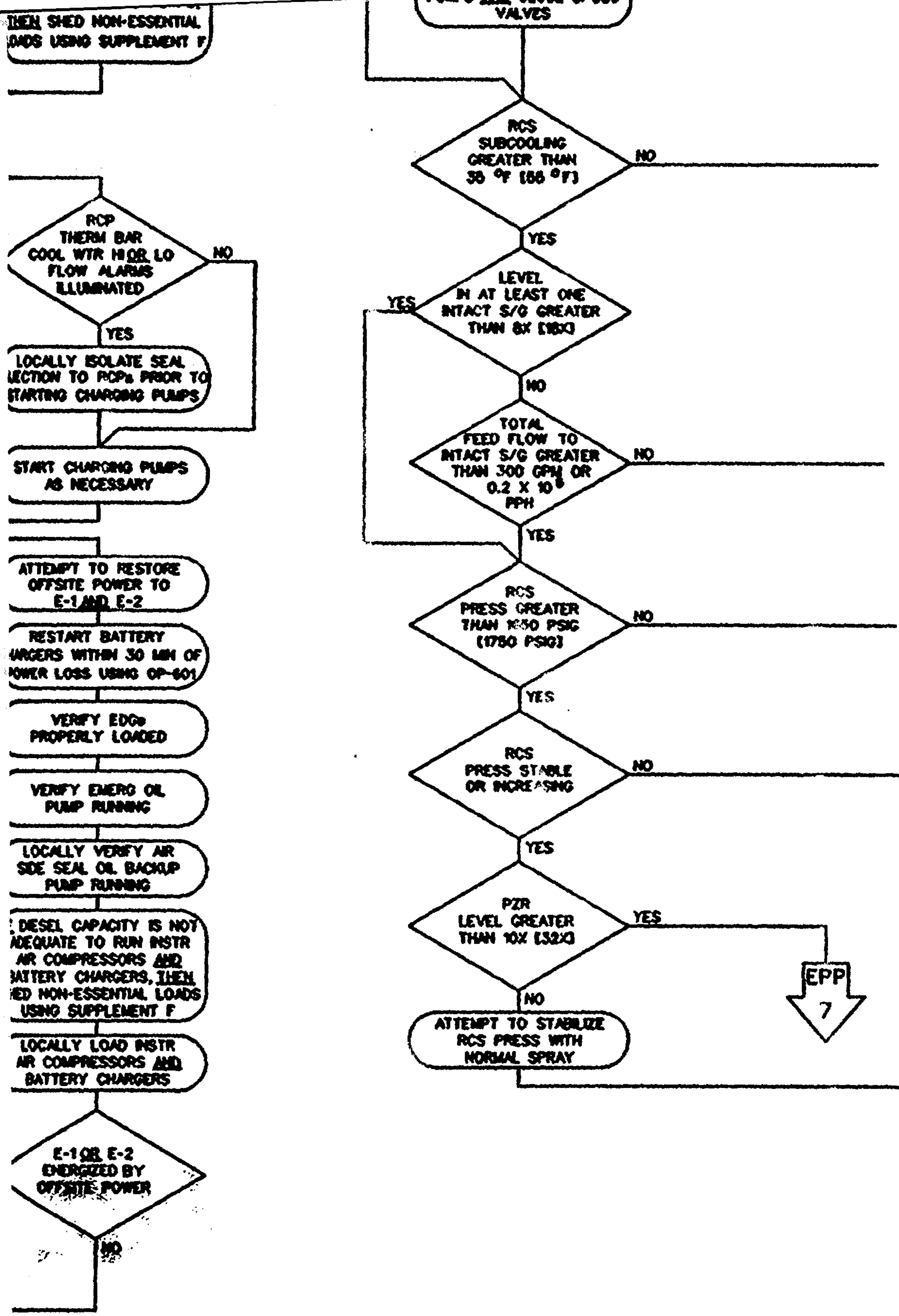
d. **CORRECT** RCS inventory is not adequate since 32% level is required with adverse containment conditions. The crew would be directed to stabilize pressure and transition to EPP-007 when SI flow restores adequate pressurizer level.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of criteria for terminating safety injection

REFERENCES SUPPLIED:


THEN SHED NON-ESSENTIAL
LOADS USING SUPPLEMENT F
VALVES
RCS SUBCOOLING GREATER THAN 35 °F [55 °F]
NO
YES
RCP THERM BAR COOL WTR HI OR LO FLOW ALARMS ILLUMINATED
NO
YES
LEVEL IN AT LEAST ONE INTACT S/G GREATER THAN 8% [18%]
YES
NO
LOCALLY ISOLATE SEAL INJECTION TO RCPs PRIOR TO STARTING CHARGING PUMPS
TOTAL FEED FLOW TO INTACT S/G GREATER THAN 300 GPM OR 0.2 X 10^6 PPH
NO
YES
START CHARGING PUMPS AS NECESSARY
ATTEMPT TO RESTORE OFFSITE POWER TO E-1 AND E-2
RCS PRESS GREATER THAN 1650 PSIG [1750 PSIG]
NO
YES
RESTART BATTERY CHARGERS WITHIN 30 MIN OF POWER LOSS USING OP-601
VERIFY EDGs PROPERLY LOADED
RCS PRESS STABLE OR INCREASING
NO
YES
VERIFY EMERG OIL PUMP RUNNING
LOCALLY VERIFY AIR SIDE SEAL OIL BACKUP PUMP RUNNING
PZR LEVEL GREATER THAN 10% [32%]
YES
NO
EPP 7
IF DIESEL CAPACITY IS NOT ADEQUATE TO RUN INSTR AIR COMPRESSORS AND BATTERY CHARGERS, THEN SHED NON-ESSENTIAL LOADS USING SUPPLEMENT F
ATTEMPT TO STABILIZE RCS PRESS WITH NORMAL SPRAY
LOCALLY LOAD INSTR AIR COMPRESSORS AND BATTERY CHARGERS
E-1 OR E-2 ENERGIZED BY OFFSITE POWER
NO

EPP-008-06 001

Given the following plant conditions:

- A Reactor Trip and Safety Injection have occurred
- As directed by PATH-1, you have transitioned to EPP-8, "Post LOCA Cooldown and Depressurization"
- Two SI Pumps are running
- Both RHR Pumps have been secured
- The crew has reached the step for SI pump reduction

Which ONE (1) of the following describes two conditions, that if both are met, allow you to secure an SI pump?

✓A. RCS Subcooling greater than required and adequate PZR level

B. RCS Subcooling greater than required and at least one Charging pump running

C. RCS Hot Leg temperatures low enough and adequate PZR level

D. RCS Hot Leg temperatures low enough and at least one Charging pump running

Question: 99

Given the following conditions:

- A reactor trip and safety injection have occurred.
- During the performance of PATH-1 a transition has been made to EPP-16, "Uncontrolled Depressurization of All SGs."
- Wide range SG levels are all between 12% and 18% and decreasing slowly.
- SG pressures are all between 180 psig and 200 psig and decreasing slowly.
- Feed flow has been reduced to 80 gpm to each SG per EPP-16 guidance.

Which ONE (1) of the following describes when FRP-H.1, "Loss of Heat Sink," guidance would be implemented to restore SG levels?

a. Wide range level in 2 SGs is still below 26%

b. Narrow range level in 1 SG is still below 10%

c. 2 SGs remain unisolated

d. Total feed flow is below 300 gpm due to other than operator actions

Answer:

d. Total feed flow is below 300 gpm due to other than operator actions

QUESTION NUMBER: 99
TIER/GROUP: ***RO*** ***SRO*** 1/2
K/A: WE05EA2.1

Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) Facility conditions and selection of appropriate procedures during abnormal and emergency conditions

K/A IMPORTANCE: ***RO*** ***SRO*** 4.4
10CFR55 CONTENT: ***55.41(b) RO*** ***55.43(b) SRO*** 5

OBJECTIVE: FRP-H.1-02

RECOGNIZE the selected entry level conditions of FRP-H.1.

REFERENCES: FRP-H.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number FRP-H.1-14 001

JUSTIFICATION:

a. Plausible since this would require feed and bleed if FRP-H.1 were performed, but it is not performed due to flow being limited due to operator action.

b. Plausible since this is within the normal control band for SG level, but FRP-H.1 is not performed due to flow being limited due to operator action.

c. Plausible since this would likely result in 2 SGs being below 26%, requiring feed and bleed if FRP-H.1 were performed, but it is not performed due to flow being limited due to operator action.

d. **CORRECT** FRP-H.1 is not implemented if total feed flow is below 300 gpm due to operator actions.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of operator actions which would prohibit performance of FRP-H.1

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Feed flow is not re-established to any faulted S/G if an intact S/G is available.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Total Feed Flow - LESS THAN 300 GPM DUE TO OPERATOR ACTION	Go To Step 3.
2.	Reset SPDS And Return To Procedure And Step In Effect	
* 3.	Determine If Secondary Heat Sink Is Required As Follows:	
	a. Check RCS pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE	a. Reset SPDS and Go To PATH-1, Entry Point C.
	b. Check RCS temperature - GREATER THAN 350°F [310°F]	b. Perform the following: 1) Place RHR System in service using Supplement I. 2) WHEN adequate cooling with RHR is established, THEN reset SPDS and return to procedure and step in effect.
* 4.	Check Any Two S/G Wide Range Levels - LESS THAN 27% [34%]	IF any two S/G Wide Range Levels decrease to less than 27% [34%], THEN Go To Step 5. Go To Step 6.
5.	Perform The Following: a. Stop all RCPs b. Observe CAUTION prior to Step 30 and Go To Step 30	

Question: 100

Given the following conditions:

- The reactor is defueled.
- Over several days pure water is inadvertently added to the spent fuel pit (SFP).
- The following SFP chemistry exists:
- Boron = 1445 ppm
- Level = 37 ft

Given the supplied references, which ONE (1) of the following is the **MINIMUM** action required to restore key safety functions?

a. Add 1000 pounds of granulated boric acid to the SFP

b. Add 5500 pounds of granulated boric acid to the SFP

c. Drain the SFP a minimum of 8 feet and refill using the RWST

d. Drain the SFP a minimum of 16 feet and refill using the RWST

Answer:

a. Add 1000 pounds of granulated boric acid to the SFP

AVAILABLE CONTINGENCY ACTIONS

4.0 Reactivity Control:

1) Borated makeup sources, and all components necessary to inject the borated water are required to be operable in accordance with OMP-003 when fuel is in the vessel. Other means of borated makeup when the RCS is intact include the flow path through the RCP seals, however this should only be used as a last resort. Normal letdown if available when fuel is in the vessel, may be used to divert displaced inventory to the CVCS Hold Up Tank (HUT). As an alternate means of increasing the Boron Concentration in the Refueling cavity when the vessel head has been removed, 100 lb. bags of Granulated Boric Acid may be added to the cavity. One 100 lb. bag of Granulated Boric Acid will increase the Cavity Boron Concentration approximately 6 ppm. Contact the Reactor Engineer to provide guidance IAW the Reactivity Management Program. (SOER 94-2)

2) When the core is offloaded to the SFP, borated make-up is available from the RWST in accordance with the procedure listed on Attachment 10.2 of this procedure, however if the SFP is at the full level and no more inventory can be added, Boron Concentration may be increased by adding Granulated Boric Acid to SFP locally. One 100 lb. bag of Granulated Boric Acid will increase the Boron Concentration of the SFP approximately 6 ppm. Contact the Reactor Engineer to provide guidance IAW the Reactivity Management Program. (SOER 94-2)

QUESTION NUMBER: 100
TIER/GROUP: *RO* *SRO* 3
K/A: 2.2.26

Knowledge of refueling administrative requirements.

K/A IMPORTANCE: *RO* *SRO* 3.7
10CFR55 CONTENT: *55.41(b) RO* *55.43(b) SRO* 6

OBJECTIVE: OMM-046-04

DEMONSTRATE the use of OMM-046 in maintaining the Key Safety Functions.

REFERENCES: OMM-046
TS 3.7.13

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number OMM-046-04 005

JUSTIFICATION:

a. **CORRECT** Required boron concentration in the SFP is 1500 ppm. Concentration must be raised 55 ppm to established required concentration. Each 100 pound bag of granulated boron will raise SFP level approximately 6 ppm

b. Plausible since with the level in the SFP full normal boration using the RWST cannot be performed, but it would require 1000 pounds at 6 ppm per 100 pounds.

c. Plausible since the RWST normally provides makeup to the SFP, but with level high it cannot be increased and draining the pool would be non-conservative as the diluted water is still removing heat.

d. Plausible since the RWST normally provides makeup to the SFP, but with level high it cannot be increased and draining the pool would be non-conservative as the diluted water is still removing heat.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of the requirements for restoring SFP boron concentration

REFERENCES SUPPLIED: OMM-046, Attachment 10.3

3.7 PLANT SYSTEMS

3.7.13 Fuel Storage Pool Boron Concentration

LCO 3.7.13 The fuel storage pool boron concentration shall be ≥ 1500 ppm.

APPLICABILITY: During new and spent fuel movement activities in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- A.1 Suspend movement of fuel assemblies in the fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify the fuel storage pool boron concentration is within limit.	7 days

INITIAL SUBMITTAL

ROBINSON EXAM 2001-301

MARCH 26 - APRIL 2, 2001

INITIAL SUBMITTAL - RO/SRO COMMON WRITTEN EXAMINATION QUESTIONS

1-50

Question: 1

Given the following conditions:

- The unit is operating at 100% power.
- Annunciators APP-008-E7, S. SW HDR STRAINER PIT HI LEVEL, and APP-008-F7, SOUTH SW HDR LO PRESS, come in simultaneously.

Which ONE (1) of the following actions is required as an immediate action?

a. Stop 'A' and 'B' service water pumps

b. Close SW supply to south header valve V6-12A

c. Close SW supply to north header valve V6-12D

d. Close SW cross-connect valves V6-12B and V6-12C

Answer:

d. Close SW cross-connect valves V6-12B and V6-12C

QUESTION NUMBER: 1

TIER/GROUP: *RO* 1/1 *SRO* 1/1

K/A: 062 2.4.24

Knowledge of loss of cooling water procedures (Service Water).

K/A IMPORTANCE: *RO* 3.3 *SRO* 3.7

10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: AOP-022-05

STATE the immediate action steps of AOP-022

REFERENCES: APP-008
AOP-022

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number AOP-022-05 002

JUSTIFICATION:

a. Plausible since a severe unisolated rupture could result in flooding in critical areas, but this is not an immediate operator action.

b. Plausible since the annunciators address the south header, but this is not an immediate operator action.

c. Plausible since this action would isolate the non-ruptured header from the ruptured header, but this is not an immediate operator action.

d. **CORRECT** Immediate action to close the cross-connect valves to prevent a loss of both headers.

DIFFICULTY:

Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 2

Recall of AOP immediate actions

REFERENCES SUPPLIED:

ALARM

S SW HDR STRAINER PIT HI LEVEL

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Failure of sump pump in south service water strainer pit
2. System leakage in excess of sump pump capacity

OBSERVATIONS

1. Other SW pit Annunciators (D7, D8, & E8)

ACTIONS

1. Refer to AOP-022.

DEVICE/SETPOINTS

1. LS-1652B / 1 foot above floor

POSSIBLE PLANT EFFECTS

1. Continued flooding could jeopardize operability of valves V6-12A, V6-12B, V6-12C, & V6-12D
2. Potential to enter TECH SPEC LCO condition

REFERENCES

1. ITS LCO 3.7.7
2. AOP-022, Loss of Service Water
3. HBR2-11098, SH. 11
4. CWD B-190628, Sh. 832

ALARM

SOUTH SW HDR LO PRESS

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Loss of SW Pump(s)
2. CCW Heat exchanger Outlet Valves open too far
3. Rupture of Service Water Piping

OBSERVATIONS

1. Service Water Pressure (PI-1684, PI-1616)
2. Service Water Pump Breaker(s) Indicating Lights

ACTIONS

1. **IF** an operating SW Pump has tripped, **THEN** perform the following:
    1) START a Standby Pump.
    2) Dispatch operator to check breaker(s)
        - SW Pump A - 480V Bus E1 (CMP 20B)
        - SW Pump B - 480V Bus E1 (CMP 19C)
        - SW Pump C - 480V Bus E2 (CMP 24A)
        - SW Pump D - 480V Bus E2 (CMP 25B)
    3) Throttle CCW Heat Exchanger Return Valves, as necessary, to maintain 40 to 50 psig in the SW Headers.
2. **IF** a rupture in a SW Header has occurred, **THEN** refer to AOP-022.

DEVICE/SETPOINTS

1. PSL-1684 / 40 psig

POSSIBLE PLANT EFFECTS

1. Loss of Service Water
2. Overheat of CCW
3. Possible entry into TECH SPEC LCO

REFERENCES

1. ITS LCO 3.7.7
2. AOP-022, Loss of Service Water
3. CWD B-190628, Sheet 840, cable M
4. Flow Diagram G-190199

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Step 1 is an immediate action step.

1.	Check The Following Alarms - EXTINGUISHED: • APP-008-E7, S SW HDR STRAINER PIT HI LEVEL • APP-008-E8, N SW HDR STRAINER PIT HI LEVEL	Perform the following: a. Close the following SW X-CONN Valves: • V6-12B • V6-12C b. Go To Section F.

NOTE

A SW Header leak may be identified by observing the sequence in which SW Header low pressure alarms are received, and evaluating SW Header pressure indications.

2.	Check Leak Location - IDENTIFIED	Perform local inspections as necessary to determine leak location. <u>WHEN</u> the leak location is identified, <u>THEN</u> Go To Step 3.

Question: 2

Four Operators worked the following schedule at the RTGB position over the past six days:

HOURS WORKED (Shift turnover time not included. Do **NOT** assume any hours worked before or after this period.)

OPERATOR	DAY 1	DAY 2	DAY 3	DAY 4	DAY 5	DAY 6
1	10	14	off	12	12	12
2	14	12	14	10	off	11
3	off	off	off	13	11	14
4	11	13	14	off	11	12

Which ONE (1) of the operators would be permitted to work a 12 hour shift on Day 7 **WITHOUT** requiring permission to exceed normal overtime limits?

a. 1

b. 2

c. 3

d. 4

Answer:

a. 1

QUESTION NUMBER: 2

TIER/GROUP: *RO* 3 *SRO* 3

K/A: 2.1.1

Knowledge of conduct of operations requirements.

K/A IMPORTANCE: *RO* 3.7 *SRO* 3.8

10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: PLP-015-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in PLP-015 by explaining the basis of each

REFERENCES: PLP-015

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number PLP-015-03 002

JUSTIFICATION:

a. **CORRECT** Working a 12 hour shift on Day 7 would result in this operator working 24 hours out of 48, and 72 hours in 7 days, both of which are permissible.

b. Plausible since this operator would not exceed the 24 hours out of 48 limit and has had a recent day off, but would work 73 hours in 7 days which exceeds limit.

c. Plausible since this operator would not exceed the 72 hours in 7 day limit and has several recent days off, but would work more than 24 hours in 48 which exceeds limit.

d. Plausible since this operator would not exceed the 24 hours out of 48 limit and has had a recent day off, but would work 73 hours in 7 days which exceeds limit.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Compare given data to administrative limits to determine which selection would remain within limits

REFERENCES SUPPLIED:

8.2 Improved Technical Specifications

Improved Technical Specifications requirements set forth in detail in Section 5.2.2.e state that administrative procedures shall be developed and implemented to limit the working hours of Plant Staff who perform safety related functions. This procedure applies to the following job categories for individuals on-shift, performing safety-related work activities: all licensed Operators, Auxiliary Operators, RC Technicians, EC Technicians, I&C Technicians, Electricians, Mechanics, and their First Line Supervisors. First Line Supervisors are defined as those individuals who direct safety-related work activities of the above personnel. All other job categories are exempt. This information is intended to clarify and expand upon the requirements defined within definitions 4.1.1.1 and 4.1.1.2, and represents Robinson's interpretation and application of the available regulatory guidance. (ACR 93-211)

8.3 Requirements

Enough plant operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime. The objective is to have operating personnel work a normal shift, based on their work schedule, while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (Reference Improved Technical Specifications 5.2.2.e):

8.3.1 An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

8.3.2 An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

8.3.3 A break of at least eight hours should be allowed between work periods, including shift turnover time.

EXTENDED OVERTIME REQUEST FORM

CONTINUOUS USE

A. Applicable overtime limits which are being exceeded are one or more of the following:

1. Fewer than eight hours between work periods
2. More than 24 hours in a 48 hour period
3. More than 16 hours in one day
4. More than 72 hours in seven days

B. The following person(s) are authorized to exceed the guidelines of Technical Specification 6.2.3.b for the applicable overtime limits (all limits being exceeded shall be indicated by appropriate number under "Limit"):

Name	Limit(s)	Name	Limit(s)
1.		7.	
2.		8.	
3.		9.	
4.		10.	
5.		11.	
6.		12.	

Effective Date: ______________

C. Reason(s) why overtime guidelines are exceeded and length of time exceeded:

____
____
____
____
____

Recommended by (Supervisor): ______________________________ Date __________

Reviewed by (Unit Manager): ______________________________ Date __________

Telephonic Concurrence: ______________________________ Date __________

Approved by Plant General Manager, his designee or higher levels of management:

____ Date __________

Telephonic Concurrence: ______________________________ Date __________

D. A declared emergency is in progress. All affected personnel reporting for emergency response duties are authorized to exceed the technical specifications limits for extended overtime for the duration of the emergency.

______________________ ________

Approved by: Site Emergency Coordinator or Emergency Response Manager Date

Telephonic Concurrence: ______________________________ Date __________

PLP-015-03 002

Given the following plant conditions:

- Operators #1, #2, and #3 worked the below schedule at the RTGB position

(Do not assume any hours worked before or after the seven day schedule shown)

Hours worked (shift turnover time not included)

Operator	Day 1	Day 2	Day 3	Day 4	Day 5	Day 6	Day 7
1	11	13	off	12	12	12	12
2	14	off	13	10	12	11	13
3	off	off	12	13	11	12	12

Which operator(s) exceeded the number of hours a licensed operator may work at the RTGB position?

A. 2 only

B. 3 only

✓C. 2 and 3

D. 1, 2 and 3

Question: 3

Given the following conditions:

- The unit was operating at 100% power when a pipe break occurred inside containment.
- Containment pressure is rising.
- RCS temperature is lowering.

Which ONE (1) of the following differentiates between a non-isolable main feed line break inside containment and a non-isolable main steam line break inside the containment of the same size?

a. RCS heat removal would be greater for the steam line break

b. Containment pressure would be greater for the feed line break

c. Containment radiation levels would be greater for the steam line break

d. RCS depressurization would be greater for the feed line break

Answer:

a. RCS heat removal would be greater for the steam line break

QUESTION NUMBER: 3

TIER/GROUP: *RO* 1/2 *SRO* 1/2

K/A: 054AK1.01

Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): MFW line break depressurizes the S/G (similar to a steam line break)

K/A IMPORTANCE: *RO* 4.1 *SRO* 4.3

10CFR55 CONTENT: *55.41(b) RO* 5 *55.43(b) SRO*

OBJECTIVE: MCD-09-02

DESCRIBE the limiting analysis for the Containment Critical Safety function

REFERENCES: FSAR Accident Analysis
Steam Tables

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number MCD 001

JUSTIFICATION:

a. **CORRECT** Since the latent heat of vaporization would be removed from the RCS as feed water is boiled to steam, a greater amount of heat is removed from the RCS.

b. Plausible since feed water would flash to steam as it entered containment, but the steam break would provide more energy and a higher pressure.

c. Plausible since in the event of a concurrent SGTR gases would escape out the steam break earlier, but would eventually escape to the containment through a feed break once the break is uncovered.

d. Plausible since large amounts of cold feed water would be exiting the break, but the latent heat of vaporization removes more energy from the RCS and results in a greater depressurization.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comparison of different plant responses to different initiating accidents

REFERENCES SUPPLIED:

standby condition have been analyzed.

Main Feedwater System Design

The rapid depressurization that occurs following a rupture may result in large amounts of water being added to the steam generators through the Main Feedwater System. Rapid-closing isolation valves are provided in the main feedwater lines to limit this effect. Also, the piping layout downstream of the isolation valves affects the volume in the feedwater lines that cannot be isolated from the steam generators. As the steam generator pressure decreases, some of the fluid in this volume will flash into the steam generator, providing additional secondary fluid that may exit out the rupture.

The feedwater addition that occurs before closing of the feedwater line isolation valves influences the steam generator blowdown in several ways. First, the rapid addition of feedwater increases the amount of entrained water in large-break cases by lowering the bulk quality of the steam generator inventory. This tends to reduce the amount of energy released to containment because of the lower energy content of water relative to that of steam.

Second, because the water entering the steam generator is subcooled, it lowers the steam pressure, thereby reducing the flow rate out of the break because of a reduced differential pressure. Finally, the increased flow rate causes an increase in the heat transfer rate from the primary to the secondary system because of the increase in delta-T (ΔT) across the steam generator tubes. This results in greater energy being released out the break.

Since these are competing effects on the total mass and energy release, no worst-case feedwater transient can be defined for all plant conditions. In the results presented in the FSAR, the worst effects of each variable have been used. For example, moisture entrainment for each break is calculated, assuming conservatively small feedwater additions so that the entrained water is minimized.

Determination of total steam generator inventory, however, is based on conservatively large feedwater additions. Table 10-5 contains plant-specific design input for the main steam line break analysis. In Table 10-5, the value given for mass added by feedwater pumping assumes that no reduction in feedwater pump turbine speed occurs following a main steam line break and before main feedwater isolation.

The unisolated feedwater line volumes between the steam generators and the isolation valves

Because this lesson is limited to a discussion of the Containment Critical Safety Function, only the bounding containment pressure transient analysis will be considered. The limiting containment pressure transient for the reference plant is a main steam line break inside containment. Table 10-3 lists the spectrum of secondary system pipe ruptures analyzed in the FSAR. Detailed information concerning other analyzed containment pressure transients appears in the facility FSAR.

Table 10-4 lists the initial conditions used in the FSAR containment analysis. The initial containment conditions were selected based on the range of the normal expected conditions within the containment, with consideration given to maximizing the calculated peak containment pressure.

A detailed study of the initial accident conditions was conducted to determine the effects of varying these initial conditions. The results of this study showed that varying the initial containment conditions over a wide range of values changes the calculated peak pressure by less than 1 percent. So the initial containment conditions are relatively unimportant parameters for the containment pressure and temperature analysis.

Steam line ruptures occurring inside a reactor containment structure may result in a significant release of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steam line rupture is dependent upon the many possible configurations of the plant steam system and containment design, as well as the plant operating conditions and the size of the rupture. These variations make difficult a reasonable determination of the single, absolute "worst case" for both containment pressure and temperature evaluations following a steam line break.

As stated, the FSAR analysis of the main steam line break (MSLB) considers 16 different accident scenarios. Refer to table 10-3. For each of the accident scenarios considered, the analysis is sensitive to four major factors that influence the release of mass and energy following a steam line break:

- Steam generator fluid inventory
- Primary-to-secondary heat transfer
- Protective system operation
- State of the secondary fluid blowdown

Question: 4

Given the following plant conditions:

- The RCP Seal Injection filter has just been changed out.
- HP placed the filter in a lead container.
- Prior to placement of the container, R-4, Charging Pump Room Monitor, read 2 mr/hr.
- The container is on a pallet outside of the Charging Pump Room.
- The activity source in the filter is primarily Cobalt-60.
- The container is 5 feet away from R-4 detector, and R-4 reads 10 mr/hr.

If the container is moved to 10 feet away from the R-4 detector, R-4 will indicate ...

a. 4.0 mR/hr.

b. 4.5 mR/hr.

c. 6.0 mR/hr.

d. 7.0 mR/hr.

Answer:

a. 4.0 mR/hr.

QUESTION NUMBER: 4
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 072K5.02

Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation intensity changes with source distance

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 3.2
10CFR55 CONTENT: ***55.41(b) RO*** 12 ***55.43(b) SRO***

OBJECTIVE: AOP-005-03

EXPLAIN the basis of selected steps, cautions, and notes in AOP-005

REFERENCES: GET

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AOP-005-03 012

JUSTIFICATION:

a. **CORRECT** Container contributes 8 mr/hr to reading. If double the distance, then rate falls by factor of 1/r squared, or 4. Thus. final container contribution is 2 mr/hr. Background is still present (2 mr/hr) for a total of 4 mr/hr.

b. Plausible if applies the inverse-square-ratio to the entire reading of 10 mr/hr. If double the distance, then rate falls by factor of 4. Final container contribution is 2.5 mr/hr. Background is still present (2 mr/hr) for a total of 4.5 mr/hr.

c. Plausible if applies a linear ratio to the container contribution of 8 mr/hr. Final container contribution calculated to be 4.0 mr/hr. Background is still present (2 mr/hr) for a total of 6.0 mr/hr.

d. Plausible if applies a linear ratio to the entire reading of 10 mr/hr. Final reading calculated to be 5.0 mr/hr. Background is still present (2 mr/hr) for a total of 7.0 mr/hr.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 4

Calculation based on inverse square ratio using a point source

REFERENCES SUPPLIED:

AOP-005-03 012

Given the following plant conditions:

- Mode 5
- The RCP Seal Injection filter has just been changed out
- HP placed the filter in a one inch thick lead container
- Prior to placement of the container, R-4 read 1 mr/hr
- The container is on a pallet outside of the Charging Pump Room
- There is effectively 2 inches of steel between the container and the R-4 (CHARGING PUMP ROOM MONITOR) detector
- The activity source in the filter is primarily Cobalt-60
- The container is 8 feet away from R-4 detector, and R-4 reads 5 mr/hr

Which ONE (1) of the following identifies the correct R-4 reading if the container is moved to 16 feet away from R-4 detector?

A. 1.25 mr/hr

✓B. 2.0 mr/hr

C. 2.5 mr/hr

D. 3.0 mr/hr

Question: 5

Given the following conditions:

- At 0110, a Reactor Trip and Safety Injection occurred following an accident.
- At 0112, an Alert was declared due to RCS leakage.
- At 0116, a Site Area Emergency was declared.
- At 0120, a General Emergency was declared.

Which ONE (1) of the following identifies the **LATEST** time that the **INITIAL** notification to State/County officials and the NRC must be completed?

	STATE / COUNTY	**NRC**
a.	0125	0210
b.	0127	0212
c.	0131	0216
d.	0135	0220

Answer:

b.	0127	0212

QUESTION NUMBER: 5
TIER/GROUP: ***RO*** 3 ***SRO*** 3
K/A: 2.4.43

Knowledge of emergency communications systems and techniques.

K/A IMPORTANCE: ***RO*** 2.8 ***SRO*** 3.5
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: EPSPA01-03

DEMONSTRATE an understanding of the CR/EOF Emergency Communicator

REFERENCES: EPLCA-01

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EPSPA01-03 001

JUSTIFICATION:

a. Plausible since these times are consistent with the event initiation, but times are based on the declaration time.

b. **CORRECT** Notifications are required within 15 minutes of the initial declaration to the state/county and 1 hour to the NRC.

c. Plausible since these times are consistent with the declaration of the Site Area Emergency, but times are based on the initial declaration time.

d. Plausible since these times are consistent with the declaration of the General Emergency, but times are based on the initial declaration time.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Calculation of time limits based on knowledge of EP requirements

REFERENCES SUPPLIED:

8.1.3 (Continued)

12. If a General Emergency has been declared, formulate a protective Action Recommendation (PAR).

    a. Use guidance in Attachments 8.1.5.1, Initial Protective Action Recommendation Flowchart and Attachment 8.1.5.3, PAR Affected Zones Based on Wind Direction to formulate the initial recommendation and zones to be evacuated based on wind direction.

    b. Subsequent PARs are made by comparing dose projections and environmental monitoring results to Attachment 8.1.5.2, Protective Action Guidelines (PAG) and upgrading the initial recommendations as necessary.

13. Develop and transmit an initial Emergency Notification Form to at least one State and County agency within 15 minutes of emergency declaration.

    a. Follow up notifications are required at least every 30-60 minutes.

14. Within one hour of an Alert (or above) declaration, activate the Emergency Response Data System (ERDS) as noted below:

    a. If the ERDS is not currently operational (ERDS = NORMAL is not displayed at the bottom of an ERFIS terminal), the SEC will ensure that ERDS is activated. Any problems should be reported to Information Technology personnel.

    b. Display the ERDS activation screen by:

       - Depressing the ERDS key on the ERFIS keyboard, or
       - Typing the Turn-On-Code "ERDS" at the input field, or
       - Selecting ERDS from the EP Menu.

EPSPA01-03 001

Given the following plant conditions:

- At 0608 a Reactor Trip and Safety Injection occurred
- At 0610 an Alert was declared due to RCS leakage
- At 0617 a Site Area Emergency was declared
- It is now 0622

Which ONE (1) of the following determines the amount of time remaining to complete the initial notification to State/County officials and the NRC?

✓A. 3 minutes for State/County, 48 minutes for NRC

B. 1 minute for State/County, 46 minutes for NRC

C. 1 minute for NRC, 46 minutes for State/County

D. 3 minutes for NRC, 48 minutes for State/County

Question: 6

Given the following plant conditions:

- An emergency boration is in progress through MOV-350, BA to Charging Pmp Suct, per FRP-S.1, "Response to Nuclear Power Generation / ATWS."
- FI-110, Boric Acid Bypass Flow, indicates 33 gpm.
- FI-122, Charging Line Flow, indicates 75 gpm.
- VCT level is 23 inches.
- VCT Makeup is aligned for automatic operation.
- Normal letdown has been isolated.

VCT level will ...

a. remain essentially unaffected.

b. decrease to the auto makeup setpoint and stabilize.

c. decrease to the low-level setpoint and cause the charging pump suction to switch to the RWST.

d. decrease to the auto makeup setpoint and cycle between the makeup start and stop setpoints.

Answer:

d. decrease to the auto makeup setpoint and cycle between the makeup start and stop setpoints.

QUESTION NUMBER: 6
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: 024AA1.05

Ability to operate and / or monitor the following as they apply to the Emergency Boration: Performance of letdown system during emergency boration

K/A IMPORTANCE: ***RO*** 3.1 ***SRO*** 3.2
10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: CVCS-09

EXPLAIN the normal operation of the CVCS control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-021
FRP-S.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number CVCS-09 008

JUSTIFICATION:

a. Plausible if misconception is that VCT is isolated from charging pump suction during emergency boration, but remains aligned.

b. Plausible since charging exceeds emergency boration flow and VCT level will lower, but makeup capability even with emergency boration flow is greater than the difference between charging and boration.

c. Plausible since charging exceeds emergency boration flow and VCT level will lower, but makeup capability is still available.

d. **CORRECT** Since charging exceeds emergency boration flow, VCT level will decrease. Automatic makeup will occur to cause VCT level to rise.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the effect of performing an emergency boration on the remainder of CVCS

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	Initiate Emergency Boration Of RCS As Follows:	
	a. Verify Charging flowpath established as follows:	
	1) CVC-310B, LOOP 2 COLD LEG CHG - OPEN	1) Open CVC-310A, LOOP 1 HOT LEG CHG.
	2) HIC-121, CHARGING FLOW Controller - DEMAND SIGNAL AT 0%	
	b. Verify Two Charging Pumps - RUNNING AT FULL SPEED	
	c. Verify Boric Acid Pump aligned for blend - RUNNING	c. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT OR • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET. 3) Go To Step 4.f.

(CONTINUED NEXT PAGE)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	(CONTINUED)	
	d. Verify MOV-350, BA TO CHARGING PMP SUCT - OPEN	d. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT <u>OR</u> • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET. 3) Go To Step 4.f.
	e. Check flow on FI-110, BORIC ACID BYPASS FLOW - FLOW INDICATED	e. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT <u>OR</u> • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET.
	f. Verify Charging Flow to RCS on FI-122A	
5.	Verify CONTAINMENT VENTILATION ISOLATION - INITIATED	
* 6.	Check SI - INITIATED	<u>IF</u> An SI Signal occurs, <u>THEN</u> verify auto start of all SI equipment using Supplement L, while continuing with this procedure. Go To Step 8

6.4.1 Emergency Boration

Emergency Boration is required when an uncontrolled cooldown is in progress while shut down, an unexplained or uncontrolled reactivity increase is occurring, or an Anticipated Transient Without Scram (ATWS) event has occurred.

Five supply paths to charging pump suction exist: through FCV-113A and 113B; through LCV-115B or CVC-358 and close LCV-115C; through FCV-113A and 114B; through MOV-350, or through FCV-113A and CVC-356. The path through MOV-350 shall only be used if intent is to shut down the reactor and maintain it shut down.

Operator actions required to initiate emergency boration include aligning the preferred path, shutting LCV-115C if the RWST is being used and verifying boric acid flow to the RCS through normal or alternate charging, auxiliary spray line or RCP seal injection.

6.4.2 Malfunction of Reactor Makeup Control

Emergency low-low level in VCT causes automatic actions - shifts charging pump suction to RWST.

Low level in VCT requires operator to verify automatic operation or initiate manual operations to restore level.

Manual system lineup will be required if the entire makeup system is inoperative.

Operator actions are directed by AOP-003, Malfunction of Reactor Makeup Control

6.4.3 Loss of Instrument Air to CVCS

Letdown isolated due to orifice valves and letdown valves failing closed

Charging isolations to both loops fail open as does charging flow control valve (HCV-121).

All air operated valves in makeup control system close, except for boric acid to blender valve CVC-113A which fails open.

Automatic switchover to the RWST on VCT low-low level is defeated because LCV-115B fails closed.

Any operating charging pump will go to maximum speed. The Foxboro is an AUTO/MANUAL station and not a controller. The Foxboro allows an I/P signal to system controls.

NOTE: The following starting duty limitations apply to the Primary Water pump motor:

1. Maximum number of starts per hour is 20.
2. Minimum time between starts is 2 minutes.

YIC-113, BORIC ACID TOTALIZER, provides a means of setting the amount (in gallons) of boric acid to be added from the RTGB and displays the amount of boric acid actually added. When using the makeup controls in the BORATE mode, it closes FCV-113A and stops the boric acid transfer pump after the desired volume is added.

YIC-114, PRIMARY WATER TOTALIZER, provides a means of setting the amount of Primary Water to be added (in gallons) for a dilution or alternate dilution and provides a display of the amount of primary water actually added. When using the makeup controls in AUTO, DILUTE, and ALTERNATE DILUTE modes, it closes FCV-114A and stops the Primary Water Pump after the desired volume is added.

The RCS Makeup System START/STOP Switch starts and stops makeup when the mode selector switch is not is AUTO and enables/disables auto makeup when the mode selector switch is in AUTO. This switch spring returns to neutral.

The RCS Makeup Mode Selector Switch is a four position (BORATE, AUTO, DILUTE, ALT DILUTE) RTGB switch which controls the mode of makeup to the VCT.

5.2.6 Automatic Makeup (Figure 6, 18 & 19)

The Automatic Makeup mode of operation of the reactor makeup control system provides dilute boric acid solution preset to match the boron concentration in the RCS. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration. It operates on VCT level, starting at 20.2 inches and stopping at 24.4 inches.

Under normal plant operating conditions, the mode selector switch and makeup stop valves (FCV-113A and B, and FCV-114A) are set in the AUTOMATIC MAKEUP position. A preset low level signal from the VCT level controller causes the automatic makeup control action to start a boric acid transfer pump, start a primary water makeup pump, open the makeup stop valve (FCV-113B) to the charging pump suctions, open the concentrated boric acid control valve (FCV-113A) and the primary water makeup control valve (FCV-114A). The flow controllers then blend the makeup stream according to the present concentration in the RCS. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At the preset high level point, the makeup is stopped by the following actions:

- Primary makeup water control valve (FCV-114A) closes
- Boric acid transfer pump is stopped
- Primary water transfer pump is stopped
- Concentrated boric acid control valve (FCV-113A) closes
- Makeup stop valve (FCV-113B) to charging pump suction closes

This operation may be stopped manually by actuating the makeup stop. The blend composition desired in the Automatic Makeup mode of control is established by the plant operator. The primary makeup water flow is controlled at a selectable preset rate when the Automatic Makeup mode is functioning. The boric acid flow is controlled to the rate determined by the plant operator and manually set on the boric acid flow controller. The control of valve FCV-114A for primary makeup water flow is dependent upon the RCS makeup switch position. With the RCS makeup switch and the FCV-114A control station in automatic, valve position is controlled by the HFC-114 setting which is normally set at 100 gpm. With the RCS makeup switch in dilute or alternate dilute and the FCV-114A control station in automatic valve position is controlled by the FCV-114A setting. When the FCV-114A control station is set in manual, regardless of RCS makeup mode switch position, the arrow buttons on the controller are used to set valve position.

The desired boron concentration in the blend is determined on the basis of the present RCS boron concentration. A chart showing the ratio of concentrated boric acid flow to primary makeup water flow vs. boron concentration in the blend stream is used by the operator to determine the boric acid flow and primary makeup water flow control setpoints. The Automatic Makeup blending control functions on demand signals from the VCT level controller.

5.2.7 Dilute (Figure 20 & 21)

The Dilute mode of operation permits the addition of a pre-selected quantity of primary water makeup at a pre-selected flow rate to the RCS. The amount of dilution required to change RCS boron concentration can be estimated using the nomographs in the Station Curve Book. The operator sets the mode selector switch to DILUTE, the primary water makeup flow controller (FC-114) setpoint to the desired flow rate, and the primary water makeup batch integrator (YIC-114) to the desired quantity. Turning the makeup control switch to start opens the primary water makeup control valve (FCV-114A), the VCT makeup stop valve to the VCT (FCV-114B), and starts a primary water makeup pump. Primary water is added to the VCT and thus to the charging pump suction header. Excessive rise of the VCT water level is prevented by automatic actuation of a three-way diversion valve (LCV-115A), which routes the reactor coolant letdown flow to the CVCS holdup tanks. When the preset quantity of primary water has been added, the batch integrator causes the primary water makeup pump to stop and FCV-114A and FCV-114B to close. This operation may be stopped manually by actuating the makeup stop switch.

CVCS-09 008
Given the following plant conditions:

- An emergency boration is in progress
- FI-110, Rapid Boration Flow, indicates 63 gpm
- FI-122, Charging Line Flow, indicates 90 gpm
- VCT level is 35%
- Normal letdown is in service

Which ONE (1) of the following describes the effect emergency boration will have on VCT level?

VCT level will:

A. remain essentially unaffected

B. decrease to the auto makeup setpoint and stabilize

✓C. increase then stabilize after the divert valve is full open

D. decrease to the auto makeup setpoint and cycle between makeup start and stop setpoints

Question: 7

Given the following conditions:

- The unit is operating at 100% power.
- APP-003-C3, PRT HI PRESS and APP-003-D3, PRT HI/LO LVL have alarmed.
- PRT level and pressure are slowly increasing, but there is **NO** appreciable increase in PRT temperature.
- **NO** other annuciators are in alarm.

The PRT response is likely being caused by leakage past ...

a. PCV-455C, PZR PORV.

b. RC-551A, PZR Safety.

c. CVC-203A, High Pressure Letdown Line Relief.

d. CVC-382, Seal Water Return Line Relief.

Answer:

d. CVC-382, Seal Water Return Line Relief.

QUESTION NUMBER: 7
TIER/GROUP: ***RO*** 2/3 ***SRO*** 2/3
K/A: 007A3.01

Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 2.9
10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: PZR-14

EXPLAIN the effect on the PZR and PRT System due to selected failures

REFERENCES: APP-003

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number PZR-03 003

JUSTIFICATION:

a. Plausible since this is a discharge source to the PRT, but incorrect since no accompanying temperature alarms / changes have occurred.

b. Plausible since this is a discharge source to the PRT, but incorrect since no accompanying temperature alarms / changes have occurred.

c. Plausible since this is a discharge source to the PRT, but incorrect since no accompanying temperature alarms / changes have occurred.

d. **CORRECT** Discharges to the PRT and temperature is approximately the same as the normal PRT temperature.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comparison of effect of inputs to PRT on PRT conditions

REFERENCES SUPPLIED:

ALARM

PRT HI PRESS

AUTOMATIC ACTIONS

1. Not Applicable

CAUSE

1. In leakage from Makeup Water, Pressurizer Relief Valves, Pressurizer Safety Valves, RHR Loop Relief Valves, Letdown Relief Valves, Seal Water Return Relief Valve, SI Test Line Relief Valve, or SI Cold Leg Injection Header Relief Valve
2. Failure of N_2 Supply to PRT
3. Opening of Pressurizer Safety or PORV

OBSERVATIONS

1. PRT Level (LI-470)
2. PRT Pressure (PI-472)
3. PRT Temperature (TI-471)
4. Pressurizer Safety Valve Line Temperatures (TI-465, TI-467, TI-469)
5. PORV Discharge Line Temperature (TI-463)

ACTIONS

1. **IF** a PZR PORV or Safety fails open while greater than 350°F, **THEN** Refer To Path-1.
2. **IF** pressure is high, **THEN** vent the PRT as follows:
    1) Open RC-549, PRT VENT
    2) **IF** required, **THEN** verify a Waste Gas Compressor starts.
    3) **WHEN** pressure is less than 3 psig, **THEN** close RC-549.
3. **IF** necessary, **THEN** adjust Nitrogen Regulator to PRT.
4. **IF** necessary, **THEN** drain the PRT using OP-103.

DEVICE/SETPOINTS

1. PC-472 / 5 psig

POSSIBLE PLANT EFFECTS

1. PRT Rupture Disk failure at 100 psig

REFERENCES

1. Path-1, EOP Network
2. CWD B-190628, Sheet 461, Cable P
3. OP-103, Pressurizer Relief Tank Control System

ALARM

PRT HI/LO LVL *** WILL REFLASH ***

AUTOMATIC ACTIONS

1. Not Applicable

CAUSE

High

1. Excessive makeup water added
2. In leakage from Makeup Water, Pressurizer Relief Valves, Pressurizer Safety Valves, RHR Loop Relief Valves, Letdown Relief Valves, Seal Water Return Relief Valve, SI Test Line Relief Valve, or SI Cold Leg Injection Header Relief Valve
3. Opening of Pressurizer Safety or PORV

Low

1. Leakage from PRT to the Reactor Coolant Drain Tank or other area.
2. Excessive draining.

OBSERVATIONS

1. PRT Level (LI-470), Pressure (PI-472), and Temperature (TI-471)
2. Pressurizer Safety Valve Line Temperatures (TI-465, TI-467, TI-469)
3. PORV Discharge Line Temperature (TI-463)

ACTIONS

1. **IF** a PZR PORV or Safety fails open while greater than 350°F, **THEN** Refer To Path-1.
2. **IF** level is high, **THEN** drain the PRT using OP-103.
3. **IF** level is low, **THEN** add Primary Water to the PRT using OP-103.

DEVICE/SETPOINTS

1. LC-470 / 83%
2. LC-470 / 68%

POSSIBLE PLANT EFFECTS

1. None Applicable

REFERENCES

1. Path-1, EOP Network
2. OP-103, Pressurizer Relief Tank Control System
3. CWD B-190628, Sheet 461, Cable M, N

Question: 8

Which ONE (1) of the following conditions would result in a reactor trip?

a. PT-447, First Stage Turbine Pressure, fails low with power level at 22%

b. NI-43, PR Channel N43, fails low with power level at 49%

c. PT-446, First Stage Turbine Pressure, fails high with power level at 1×10^{-8} amps

d. NI-44, PR Channel N44, fails high with power level at at 1×10^{-8} amps

Answer:

c. PT-446, First Stage Turbine Pressure, fails high with power level at 1×10^{-8} amps

QUESTION NUMBER: 8
TIER/GROUP: ***RO*** 2/3 ***SRO*** 2/3
K/A: 045K1.18

Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the RPS

K/A IMPORTANCE: ***RO*** 3.6 ***SRO*** 3.7
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: MT-11

EXPLAIN the reactor trips associated with the MT System. Include purpose and setpoints.

REFERENCES: SD-011

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since P-7 blocks are changed at 10% equivalent power, but 1/2 above 10% enables turbine trip to reactor trip.

b. Plausible since P-7 blocks are changed at 10% power and P-10 provides an input to P-7, but 2/4 above 10% enables turbine trip to reactor trip.

c. CORRECT At this power level the turbine stop valves are closed. With 1/2 First Stage Pressure transmitters failing high, P-7 automatically unblocks the turbine trip to reactor trip signal.

d. Plausible since indicated power above P-7 would cause a reactor trip with the turbine stop valves closed, but coincidence for P-10 input to P-7 is 2/4 above 10% power.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Analysis of the effect of instrument failures on turbine trip to reactor trip circuits

REFERENCES SUPPLIED:

ATTACHMENT 10.1
Page 2 of 3

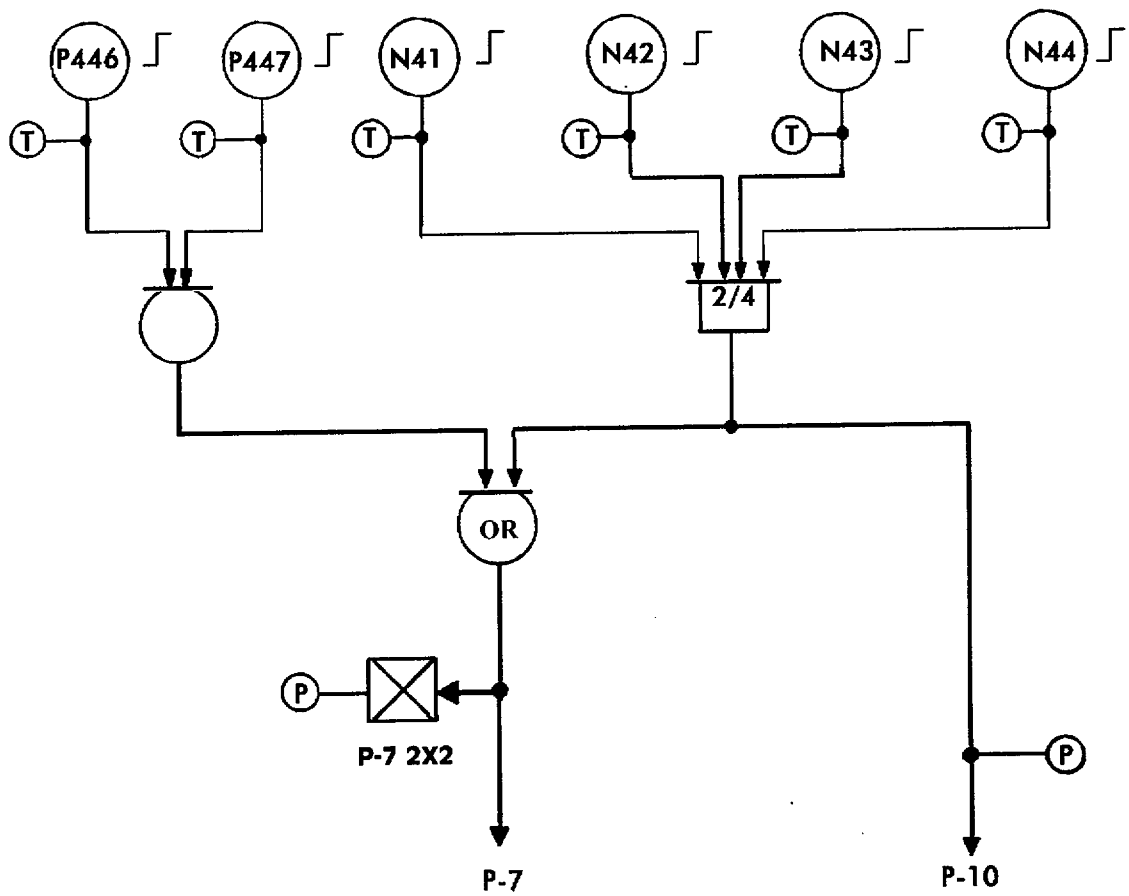
REACTOR PROTECTION SYSTEM PERMISSIVES

PERMISSIVE NUMBER	DERIVATION	FUNCTION
P-7	2/4 Power Ranges above setpoint (10% from P-10) OR 1/2 Turbine First Stage Pressure above setpoint (10%) 3/4 Power Ranges below setpoint (10%) from P-10 AND 2/2 Turbine First Stage Pressure below setpoint (10%)	Enables the following trips: 1. RCS Low Flow 2. RCP Breakers Open 3. UV 4. Turbine Trip 5. PZR low Pressure 6. PZR High Level Blocks the following reactor trips: 1. RCS Low Flow 2. RCP Breakers Open 3. UV 4. Turbine Trip 5.PZR Low Pressure 6.PZR High Level
P-8	2/4 Power Ranges above setpoint (40%) 3/4 Power Ranges below setpoint (40%)	Enables Reactor Trip on low flow in a single loop Blocks Reactor Trip on low flow in a single loop

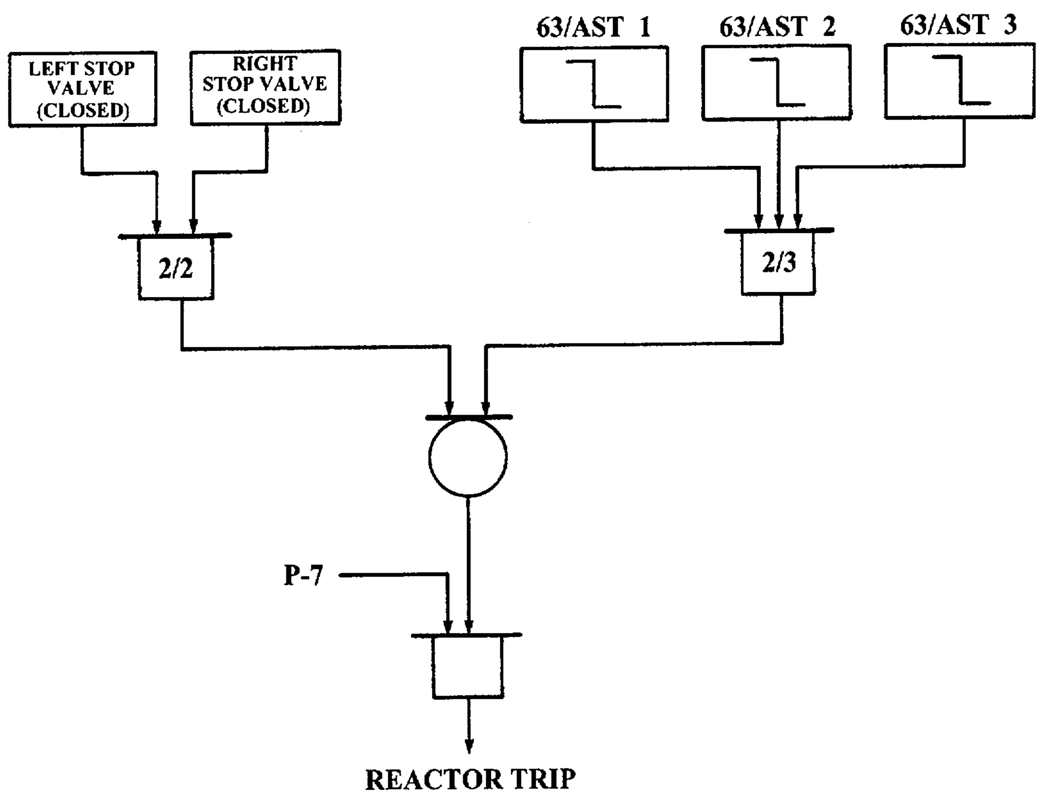
INFORMATION USE ONLY

P-7 AND P-10 PERMISSIVES

RPS-FIGURE-12 (Rev . 1)



TURBINE TRIP/REACTOR TRIP LOGIC



Question: 9

Which ONE (1) of the following describes the reason for RCP restart in FRP-P.1, "Response To Imminent Pressurized Thermal Shock", if the SI termination criteria **CANNOT** be satisfied?

a. Restores PZR spray to allow RCS depressurization in subsequent steps

b. Equalizes S/G pressures to allow simultaneous cooldown of all three loops in subsequent steps

c. Mixes Safety Injection water and RCS water to raise the fluid temperature entering the Reactor Vessel downcomer

d. Transfer core cooling to forced flow allowing the operators to terminate Safety Injection when the criteria are **NOT** satisfied

Answer:

c. Mixes Safety Injection water and RCS water to raise the fluid temperature entering the Reactor Vessel downcomer

QUESTION NUMBER: 9
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: WE08EK3.3

Knowledge of the reasons for the following responses as they apply to the (Pressurized Thermal Shock) Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

K/A IMPORTANCE: ***RO*** 3.7 ***SRO*** 3.8
10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: FRP-P.1-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in FRP-P.1 by explaining the basis of each.

REFERENCES: FRP-P.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number FRP-P.1-03 004

JUSTIFICATION:

a. Plausible since starting an RCP does restore pressure control using normal sprays, but the RCP is started to provide mixing for the SI water.

b. Plausible since during natural circ the SG pressures may vary due to different steaming rates, but cooldowns are not performed during the implementation of this procedure.

c. **CORRECT** Cold SI water flows through the cold leg to the downcomber with no RCPs running to create mixing. This could result in radical drops in temperature along the downcomber wall.

d. Plausible since cooling will be by forced flow, but SI is not terminated unless all conditions are met.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of background information in FRP-P.1

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16.	Determine If An RCP Can Be Started As Follows:	
	a. Check RCS subcooling - GREATER THAN 35°F [55°F]	a. Go To Step 44.
	b. Establish support conditions for running an RCP using OP-101, Reactor Coolant System and Reactor Coolant Pump Startup and Operation	b. Go To Step 44.
	c. Start one RCP using OP-101, Reactor Coolant System and Reactor Coolant Pump Startup and Operation	c. Go To Step 44.
	d. Go To Step 44	

****

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

****

17. Reset The Following Signals:

- SAFETY INJECTION
- CONTAINMENT SPRAY

18. Reset The Following Containment Isolations:

- PHASE A
- PHASE B

RNP STEP	WOG STEP	BASIS/DIFFERENCES

12 4 WOG BASIS

PURPOSE: To specifically note if PRZR PORVs are properly positioned

BASIS:

Depending upon the implementation of the Cold Overpressure Protection System (COPS), the pressure criterion used for checking PORV operations is either PRZR pressure less than the PORV setpoint (if COPS not in service) or RCS pressure less than cold overpressure limit (if COPS in service). If the appropriate pressure criterion is met, the PRZR PORVs should be closed.

RNP DIFFERENCES/REASONS

The step has been formatted as a continuous actions step (see C4 above).

SSD DETERMINATION

This is an SSD per criterion 11.

13 5 WOG BASIS

PURPOSE: To determine if any high-head SI pump is running

BASIS:
If SI is in service, then the SI termination sequence in Steps 6 through 12, which includes stopping SI pumps and establishing charging flow, is appropriate. If SI is not in service, these steps are bypassed.

RNP DIFFERENCES/REASONS

There are no significant differences.

SSD DETERMINATION

This is not an SSD.

14-16 6 WOG BASIS

PURPOSE: To determine if conditions have been established which indicate that full SI flow is no longer required

BASIS:

Following SI actuation, RCS conditions may be restored to within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this guideline than those present in the ORGs since, for an imminent PTS condition, SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

RNP STEP | WOG STEP | BASIS/DIFFERENCES

BASIS (Continued)

14-16 6

The subcooling criterion will ensure subcooled conditions and the RVLIS indication ensures the existence of an adequate vessel inventory such that core cooling is ensured. Refer to document SI TERMINATION/REINITIATION in the Generic Issues section of the Executive Volume.

If either of the termination criteria are not satisfied, then SI is required to ensure core cooling and should not be terminated. Most likely the cold leg/downcomer low temperature condition is due to SI water mixing effects and an RCP restart is attempted. Of the transients considered in PTS, the SBLOCA transient may result in a condition whereby Safety Injection (SI) flow cannot be terminated. In Westinghouse Owners Group (WOG) reports 0G-110 and 0G-117 titled "Evaluation of Alternate RCP Trip Criteria" and "Justification of Manual RCP Trip for Small Break LOCA Events" respectively, a range of SBLOCAs were identified where continued RCP operation or conversely untimely RCP restart could result in increased RCS inventory loss. The loss of additional inventory could ultimately result in deeper core uncovery transients which could in turn result in fuel cladding temperatures in excess of the plant's design basis FSAR analysis result. Therefore, from a SBLOCA standpoint, RCP restart at an inopportune time could result in a degraded core cooling scenario.
In WCAP-10319 titled "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants", numerous transient analyses including those of SBLOCA have been analyzed without RCP restart. The results of the stagnant loop evaluation demonstrate that the total expected frequency of significant flaw extension in a typical W PWR reactor vessel due to PTS, including the contributions from stagnant loop SBLOCA transients, does not exceed the NRC required RTPTS screening value of 270°Ffor axial flaws. Therefore, based on analyses results, RCP restart is not required to meet the NRC PTS risk goal for a typical W plant.

Therefore, an additional support condition, RCS subcooling, in addition to plant specific minimum support conditions is recommended to assure that no potential RCS inventory aggravation will occur due to RCP restart.

An analysis of the effect of an RCP restart has been made to ensure the safety of this action relative to vessel integrity. For conservatism in the analysis the assumption was made that a small preexisting flaw had grown and arrested at 75 percent of wall thickness before RCP start. Starting an RCP was shown not to result in any further flaw propagation and loss of vessel integrity. For a case where a flaw has not grown prior to RCP start, the subsequent heat-up of the downcomer region will decrease the possibility of flaw initiation.

Therefore, in order to mix the cold incoming SI water and the warm reactor coolant water and thereby decrease the likelihood of a PTS condition, an RCP restart is attempted. Whether an RCP is started or not, the next step performed (Step 24), if SI is still required, provides guidance on subsequent cooldown restrictions.

Question: 10

Given the following conditions:

- The plant has experienced a reactor trip.
- The CRSS directs the RO to manually initiate Safety Injection.
- The RO inadvertently depresses **BOTH** Containment Spray pushbuttons.

In addition to Containment Spray, which ONE (1) of the following are **ALL** expected to automatically occur?

a.
- Phase A
- Phase B

b.
- Phase A
- Containment Ventilation Isolation

c.
- Phase B
- Containment Ventilation Isolation

d.
- Phase A
- Phase B
- Containment Ventilation Isolation

Answer:

c.
- Phase B
- Containment Ventilation Isolation

QUESTION NUMBER: 10

TIER/GROUP: *RO* 2/3 *SRO* 2/2

K/A: 103K4.06

Knowledge of containment system design feature(s) and/or interlock(s) which provide for the Containment isolation system

K/A IMPORTANCE: *RO* 3.1 *SRO* 3.7

10CFR55 CONTENT: *55.41(b) RO* 9 *55.43(b) SRO*

OBJECTIVE: CSS-08

EXPLAIN the component operation associated with each switch position for the CSS switches and controls.

REFERENCES: SD-024
SD-006

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number CSS-08 003

JUSTIFICATION:

a. Plausible since Phase B occurring is correct, but Phase A does not occur.

b. Plausible since CVI occurring is correct, but Phase A does not occur.

c. CORRECT Manual actuation of Containment Spray results in Phase B and CVI occurring.

d. Plausible since Phase B and CVI occurring is correct, but Phase A does not occur.

DIFFICULTY:

Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of system actuations in response to manual actuation

REFERENCES SUPPLIED:

5.0 CONTROLS AND PROTECTION

5.1 Containment Spray Actuation

5.1.1 Automatic

Containment Spray Actuation will automatically occur when a Containment Hi-Hi Pressure signal is sensed at 20 psig. This will cause the following:

NOTE: In the year 2000, it is planned to reduce this setpoint to 10 psig to allow the Service Water temperature to be increased without challenging CV pressure. (ESR 99-00153).

1) Steam Line Isolation actuation (closes all three MSIVs)

2) Spray actuation

3) Safety Injection actuation

NOTE: Containment pressure bistables for spray actuation are energize-to-actuate. This differs from other ESF actuations. The purpose is to minimize the possibility for an inadvertent spray signal due to power interruption.

4) Phase "B" Containment Isolation, The following valves close:
CC-716A & B, RCP Clg Wtr Inlet Isols
FCV-626, RCP Thermal Barrier Flow Control
CC-735, RCP Thermal Barrier Outlet Isol
CC-381, RCP Seal Wtr Rtrn Isol
CVC-730, RCP Oil Coolers Outlet Isol

5.1.2 Manual

Containment Spray Actuation can be manually actuated when both Spray pushbuttons are simultaneously depressed. There are Containment Spray Defeat pushbuttons on the RTGB that are not used (abandoned in place). Spray actuation will cause the following:

1) Spray actuation
2) Containment Phase "B"
3) Containment Ventilation Isolation - The following valves will close:
   - Purge Valves
   - Pressure Relief Valves
   - Vacuum Relief Valves

3. Low Pressurizer Pressure
4. Containment High Pressure

5. Manual
6. Containment Hi-Hi Pressure

4.2.2 Safety Injection (SI or S) Signal Actions4.2.2 Safety Injection (SI or S) Signal Actions

The actions caused by a SI signal are listed below:

1. Reactor Trip
2. Emergency diesel generator startup
3. Feedwater isolation
4. Safeguard sequence actuation
5. Phase "A" Containment isolation and IVSW actuation
6. Containment Ventilation isolation
7. Control Room Ventilation shifts to the Emergency Pressurization Mode
8. Close normal dampers for HVH 1-4
9. Align various valves within the SI and RHR systems

4.3 Containment Spray4.3 Containment Spray

4.3.1 Containment Spray (P) Signal4.3.1 Containment Spray (P) Signal

The Containment Spray ("P") signal is initiated by a Hi-Hi containment pressure(10 psig) or manual actuation.

4.3.2 Containment Spray Automatic Signal Actions4.3.2 Containment Spray Automatic Signal Actions

The actions caused by a Containment Spray Automatic signal are listed below:

1. Spray actuation
2. Phase "B" containment isolation
3. Steam line isolation

4.3.3 Containment Spray Manual Signal Actions4.3.3 Containment Spray Manual Signal Actions

The actions caused by a Containment Spray Manual Signal are listed below:

1. Spray actuation
2. Phase "B" containment isolation
3. C.V. ventilation isolation

Question: 11

Given the following conditions:

- The unit is operating at 17% power.
- Condenser backpressure is 5.7 inches Hg Absolute and degrading slowly.
- A power reduction is in progress in an attempt to stabilize backpressure.
- **NO** cause has yet been identified.

Which ONE (1) of the following actions should be taken in accordance with AOP-012, "Partial Loss of Condenser Vacuum or Circulating Water Pump Trip"?

a. Trip the reactor and go to PATH-1

b. Trip the turbine and verify the plant stabilizes on the steam dumps at the point of adding heat

c. Trip the turbine and verify the plant stabilizes on the steam dumps at approximately the current power level

d. Continue the power reduction

Answer:

d. Continue the power reduction

AOP-012-03 010

Given the following plant conditions:

- Vacuum in the main condenser is decreasing
- No cause has yet been identified
- A power reduction has commenced IAW directions in AOP-012, "Loss of Condenser Vacuum"
- Power is presently at 7.0%

Which ONE (1) of the following describes the proper course of action for the above conditions?

A. If condenser backpressure reaches 7 inches Hg, trip the reactor and go to Path-1

B. If condenser backpressure reaches 7 inches Hg, trip the turbine and go to AOP-007, "Turbine Trip Below P-7"

C. If condenser backpressure reaches 10 inches Hg, trip the reactor and go to Path-1

✓D. If condenser backpressure reaches 10 inches Hg, trip the turbine and go to AOP-007, "Turbine Trip Below P-7"

QUESTION NUMBER: 11
TIER/GROUP: *RO* 1/1 *SRO* 1/1
K/A: 051AA2.02

Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

K/A IMPORTANCE: *RO* 3.9 *SRO* 4.1
10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: AOP-012-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of a partial loss of condenser vacuum or a Circulating Water Pump trip as directed by AOP-012.

REFERENCES: AOP-012

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number AOP-012-03 010

JUSTIFICATION:

a. Plausible since at this power level a reactor trip would be required if a turbine trip was required, but a trip is not required until vacuum lowers to 10" Hg Abs.

b. Plausible if misconception that reactor trip is not required at this power level and vacuum calls for turbine trip, but trip is not required until vacuum lowers to 10" Hg Abs.

c. Plausible if misconception that reactor trip is not required at this power level and vacuum calls for turbine trip, but trip is not required until vacuum lowers to 10" Hg Abs.

d. CORRECT With vacuum better than 10" Hg Abs, efforts are continued to lower turbine load and determine the cause of the loss of vacuum. A trip is not yet required.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of required actions in response to loss of vacuum

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

NS-115 is located at the Steam Dump Nitrogen Accumulator.

17.	Dispatch An Operator Perform The Following: a. Close NS-115, STEAM GENERATOR NITROGEN BLANKET ISOLATION b. Close each of the four TURBINE PERFORMANCE TEST CONNECTION VALVEs used for Nitrogen addition • HP Turbine Enclosure two valves between the LP and HP Turbines (North & South sides) • LP Turbine 2 Enclosure two valves between the LP Turbine and Generator (North & South sides)	
18.	Verify Standby Vacuum Pump - RUNNING	
19.	Verify All Available Circulating Water Pumps - RUNNING	
20.	Check Turbine Status - ON LINE	Go To Step 25.
*21.	Check Condenser Back Pressure On PI-1312 <u>AND</u> PI-1313 - GREATER THAN 10 INCHES HG ABS	Go To Step 24.
22.	Check REACTOR TRIP FROM TURB BLOCK P-7 Status Light - ILLUMINATED	Perform the following: a. Trip the Reactor. b. Go To Path-1.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
23.	Perform Turbine Trip Actions As Follows: a. Manually trip the Turbine b. Go To AOP-007, Turbine Trip Without Reactor Trip Below P-7	
*24.	Check Condenser Back Pressure On PI-1310 <u>AND</u> PI-1311 - LESS THAN 5.5 INCHES HG ABS	Perform the following: a. Reduce Turbine load as necessary to maintain Condenser back pressure less than 5.5 inches Hg abs. b. Notify Load Dispatcher of the load reduction. c. <u>IF</u> Condenser backpressure can <u>NOT</u> be maintained less than 5.5 inches Hg abs, <u>THEN</u> begin plant shutdown using GP-006, Normal Plant Shutdown From Power Operations To Hot Shutdown, while continuing with this procedure.
25.	Verify The Following VACUUM BREAKER Valves - CLOSED • MS-70A • MS-70B	
26.	Check Circulating Water Pump Operation - LESS THAN TWO PUMPS RUNNING	Go To Step 29.

Question: 12

Given the following conditions:

- The plant is shutdown following a reactor trip.
- RCPs are all secured.
- The Inadequate Core Cooling Monitor is **NOT** capable of providing subcooling margin.
- Primary Plant parameters indicate the following:

INSTRUMENT	PARAMETER	VALUE
PT-455	PZR Press	1485 psig
PT-456	PZR Press	1465 psig
PT-457	PZR Press	1515 psig
PT-402	RCS Press	1500 psig
PT-405	RCS Press	1525 psig
TI-453	PZR Temp (Surge Line)	524 °F
TI-454	PZR Temp (Vapor)	630 °F
TI-413	RCS Hot Leg WR Temp	538 °F
TI-423	RCS Hot Leg WR Temp	536 °F
TI-433	RCS Hot Leg WR Temp	534 °F
--	Highest Five (5) CETs	548 °F
		544 °F
		542 °F
		542 °F
		541 °F

The margin to saturation is ...

a. 46 °F.

b. 51 °F.

c. 56 °F.

d. 58 °F.

Answer:

a. 46 °F.

QUESTION NUMBER: 12
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 017K4.01

Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Input to subcooling monitors

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.7
10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: ICCM-10

EXPLAIN the operation of the ICCM.

REFERENCES: OP-307

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** Using lowest valid pressure (1465 psig) and highest valid CET (548 °F), saturation temperature for this pressure is 594 °F, resulting in a margin to saturation of 46 °F.

b. Plausible since this is calculated value using lowest pressure and average CET, but highest CET, not average, is used.

c. Plausible since this is calculated value using lowest pressure and highest Thot, but highest CET, not Thot, is used.

d. Plausible since this is calculated value using lowest pressure and average Thot, but highest CET should be used.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Calculation of saturation margin using various indications

REFERENCES SUPPLIED:

3.0 PREREQUISITES

3.1 The Electrical System has been lined up to supply power to the Instrument Buses in accordance with OP-603, Electrical Distribution System, and OP-001, Reactor Coolant and Protection System.

3.2 The Reactor Vessel Level Instrumentation System (RVLIS) sensors and sensing lines have been filled and vented in accordance with MRP-008.

3.3 The RVLIS system has been calibrated in accordance with LP-042.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Operating personnel should refer to Section 8.1 for determining Saturation Margin if the Inadequate Core Cooling System becomes inoperative.

4.2 Under normal conditions, the pressurizer temperature is the saturation temperature corresponding to pressurizer pressure; therefore, the difference between pressurizer temperature and hot leg temperature is approximately the margin to saturation in °F. This may be used as a rapid backup method for determining saturation margin. However, this method will not be valid in the event that the hottest spot in the RCS should shift to another point in the system, such as formation of a void in the reactor vessel head.

4.3 When manually determining saturation margin, each temperature and pressure indication must be carefully evaluated for its validity. The highest valid temperature indication and lowest valid pressure indication should be used for a conservative determination of saturation margin.

INIT

8.0 INFREQUENT OPERATIONS

8.1 Manual Calculation of Margin to Saturation

8.1.1 Initial Conditions

1. This revision has been verified to be the latest revision available.

Name (Print)	Initial	Signature	Date

8.1.2 Instructions for Manual Calculation of Margin to Saturation

1. Determine primary pressure using the lowest valid pressure indication. ________ psig ____
2. Convert pressure to absolute by adding 14.7 psi to value determined in Step 8.1.2.1. ________ psia ____
3. Enter, from the Saturated Steam Tables, the saturation temperature corresponding to the pressure shown in Step 8.1.2.2 to determine the corresponding temperature. ________ °F ____
4. Determine primary temperature using the highest valid core exit thermocouple temperature indication. ________ °F ____
5. Subtract the temperature of Step 8.1.2.4 from the temperature of Step 8.1.2.3 to determine the margin to saturation in °F. ________ °F ____

	Initials	Name(Print)	Date
Performed By:	______	______________________	________
	______	______________________	________
	______	______________________	________

Approved By: ______________________________ ________
Unit 2 - Superintendent Shift Operations Date

Question: 13

Given the following conditions:

- A 25 year old male started working for the Operations department at H.B. Robinson on March 3rd of this year.
- He previously worked this year at Shearon Harris as part of the Maintenance department.
- His exposure for this year at the Harris plant was 1200 mRem.
- He has received **NO** CP&L management exposure extensions and **NO** emergencies exist.

Which ONE (1) of the following is the **TOTAL ADDITIONAL** effective dose equivalent that the individual can receive **WITHOUT** management concurrence at Robinson this year?

a. 300 mRem

b. 800 mRem

c. 2000 mRem

d. 2800 mRem

Answer:

b. 800 mRem

QUESTION NUMBER: 13
TIER/GROUP: *RO* 3 *SRO* 3
K/A: 2.3.1

Knowledge of 10 CFR:20 and related facility radiation control requirements.

K/A IMPORTANCE:	***RO***	2.6	***SRO***	3.0
10CFR55 CONTENT:	***55.41(b) RO***	12	***55.43(b) SRO***	

OBJECTIVE: 10CFR20-03

Identify the Dose Limits for adults including:
a. Occupational Dose Limits
b. Public Dose Limits

REFERENCES: NGGM-PM-002

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number 10CFR20 008

JUSTIFICATION:

a. Plausible if misconception is that administrative limit is 1500 mRem, but limit is 2000 mRem.

b. **CORRECT** Total exposure for the year for all work performed at CP&L plants is 2000 mRem. Harris is a CP&L plant.

c. Plausible since this would be correct exposure at Robinson if previous exposure was at a utility other than a CP&L plant, but Harris exposure counts toward CP&L limit.

d. Plausible since limit is 4000 mRem if previous exposure was at a utility other than a CP&L plant, but Harris exposure counts toward CP&L limit and additional CP&L limit of 2000 mRem would be imposed.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Calculation of exposure limits based on previous exposure

REFERENCES SUPPLIED:

6.7.3 The prior dose history shall be documented on NRC Form 4 or equivalent. The record shall show each period in which the individual received occupational dose and shall be signed by the individual.

6.7.4 As a record of current year dose, a written, signed statement from the individual or the most recent employer may be accepted.

6.7.5 As documentation of cumulative lifetime dose, a written estimate signed by the individual or an up-to-date NRC Form 4 or equivalent signed by the individual or the most recent employer may be accepted.

6.7.6 Prior dose reports may be obtained by letter or electronic means (e.g. fax). However, if the authenticity of the data cannot be ascertained or the reliability is questionable, written verification shall be requested. Orally transmitted dose reports shall not be accepted.

6.7.7 Any period for which the prior dose is not obtained must be noted on the NRC Form 4 or equivalent. In establishing the allowable dose for the current year, assume that the individual received 1.25 rem (TEDE) in each quarter for which records are missing, but do not record the assumed dose values on the NRC Form 4 or equivalent.

6.8 Annual Administrative Dose Limits

6.8.1 The Company goal is that no individual shall exceed the following annual administrative limits for total effective dose equivalent:

1. 0.5 rem CP&L dose if non-CP&L dose for the current year has not been determined (no dose extension permitted).

2. 2 rem CP&L dose and 4 rem total dose if non-CP&L dose for the current year has been determined.

6.8.2 Administrative Dose Limit Extensions

1. The individual's supervisor must provide written justification for the need to extend the individual's dose limit.

2. Site Vice President approval is required to authorize an individual to receive more than 2 rem CP&L dose in a year. This responsibility will not be delegated except during a

10CFR20 008

Given the following conditions:

- A 25 year old male recently started working for the maintenance department at Robinson
- His lifetime dose is currently 31.5 REM total effective dose equivalent
- He has received no radiation exposure for the last 2 years
- No extensions have been approved and no emergencies exists

Which ONE (1) of the following is the TOTAL additional effective dose equivalent that the individual can receive without management concurrence at CP&L this year?

A. 1.5 REM

✓B. 2 REM

C. 3.5 REM

D. 4.5 REM

Question: 14

Given the following conditions:

- A clearance is in effect with two (2) Maintenance department clearance holders (Clearance Holders A and B).
- Clearance Holder A has requested a temporary lift of a portion of the clearance to test equipment for one of the tasks.
- Clearance Holder B is **NOT** available on site and is **NOT** expected back for two (2) days.

Given the provided references, which ONE (1) of the following describes the process to temporarily lift the required portion of the clearance?

a. Obtain permission of Clearance Holder A and the Control Room Shift Supervisor, remove the tags as necessary, and reinstall the tags when complete

b. Obtain permission of Clearance Holder A and Clearance Holder B's supervisor, remove the tags as necessary, and reinstall the tags when complete

c. Obtain permission of Clearance Holder A and the Control Room Shift Supervisor, remove and cancel the entire clearance, and reissue a new clearance with different boundaries

d. Obtain permission of Clearance Holder A and Clearance Holder B's supervisor, remove and cancel the entire clearance, and reissue a new clearance with the same boundaries when complete

Answer:

b. Obtain permission of Clearance Holder A and Clearance Holder B's supervisor, remove the tags as necessary, and reinstall the tags when complete

QUESTION NUMBER: 14
TIER/GROUP: ***RO*** 3 ***SRO*** 3
K/A: 2.2.13

Knowledge of tagging and clearance procedures.

K/A IMPORTANCE: ***RO*** 3.6 ***SRO*** 3.8
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: OMM-005-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in OMM-005 by explaining the basis of each.

REFERENCES: OPS-NGGC-1301

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number OMM-005-03 006

JUSTIFICATION:

a. Plausible since 1/2 clearance holders is available and the CRSS is a licensed SRO, but the CRSS is not authorized to approve removal of this clearance.

b. **CORRECT** If the original clearance holder is not available, the removal of tags requires the approval of the Alternate Clearance Holder or the clearance holder's supervisor. A temporary lift should reinstall the same clearance.

c. Plausible since 1/2 clearance holders is available and the CRSS is a licensed SRO, but the CRSS is not authorized to approve removal of this clearance.

d. If the original clearance holder is not available, the removal of tags requires the approval of the Alternate Clearance Holder or the clearance holder's supervisor, however a temporary lift should reinstall the same clearance.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of administrative requirements regarding clearance removals

REFERENCES SUPPLIED:

9.5 Boundary Changes

9.5.1 Administrative - Boundary Changes

1. Should plant needs dictate the removal of any tags, and the Clearance Holder is not available on site, the removal of the tags require the approval of the designated Alternate Clearance Holder or the individual's supervisor. The original Clearance Holder shall be notified by the individual releasing the clearance as soon as practical.

2. For boundary changes that involve removing clearance tags, the Affected Holders shall be notified of and agree with the boundary change. Once the new clearance boundary has been established, the Affected Holders shall be notified of the new clearance boundary.

3. For boundary changes that involve either temporarily lifting or permanently removing grounds, the Clearance Holders relying on those grounds for protection shall be notified of and agree with the lifting or removal of the grounds.

4. When it is necessary to temporarily lift a clearance tag, the Affected Holders shall be notified of and agree with the lifting of the clearance tag. Once the tag has been reinstalled, the Affected Holders shall be notified of the restoration of the clearance boundary.

5. Temporary Tag Lifts are intended to be for short duration jobs. Temporary Tag Lifts should not exceed the current Operations shift. S-SO concurrence is required to allow a Temporary Tag Lift to extend past the Operations shift.

6. When a boundary change involves only adding tags to a clearance, notification of Clearance Holders is not required.

7. Work activities that will be placed in an unsafe condition during a boundary change shall be suspended until such time that the boundary change is completed.

OMM-005-03 006

Given the following conditions:

- A clearance is in effect with 2 clearance holders (Clearance Holder A and B)
- A partial removal of the clearance is required to test equipment for one of the tasks

Which ONE (1) of the following describes the process the clearance holder will follow?

A. Obtain clearance holders A and B permission, cancel the entire LCTR and assign a new one to be issued.

B. Obtain permission from requesting clearance holder A, sign the clearance as canceled, but tags will not be pulled from the other components until the clearance holder B gives his approval.

✓C. Before tags can be removed, holders A and B shall be notified and agree with the boundary change.

D. Obtain permission from requesting clearance holder B, sign for and remove the tag, give it to the SRO when the rest of the clearance is to be cancelled.

Question: 15

Given the following conditions:

- Fuel is in the vessel.
- RCS temperature is 120°F.
- It is 10 days after the shutdown.
- RCS Level is 8" below the vessel flange.
- RHR cooling is lost.

Given the supplied references, which ONE (1) of the following identifies how much time remains before boiling begins occurring in the RCS?

a. 15.5 minutes

b. 22 minutes

c. 19 minutes

d. 40.5 minutes

Answer:

b. 22 minutes

QUESTION NUMBER: 15
TIER/GROUP: *RO* 1/2 *SRO* 1/2
K/A: 025AK1.01

Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

K/A IMPORTANCE: *RO* 3.9 *SRO* 4.3
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: OMM-046-04

DEMONSTRATE the use of OMM-046 in maintaining the Key Safety Functions.

REFERENCES: OMM-046
Plant Curve 7.19

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number OMM-046-04 009

JUSTIFICATION:

a. Plausible since correct curve is used, but uses 100 hour shutdown line instead of 10 day shutdown.

b. CORRECT Using Curve 7.19, the intersection of the 10 day shutdown line and 120 °F, the time to boiling is 22 minutes.

c. Plausible since correct curve is used, but uses 20 day shutdown line instead of 10 day shutdown.

d. Plausible since correct curve is used, but uses 40 day shutdown line instead of 10 day shutdown.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Application of given data to plant curves to determine time to boiling

REFERENCES SUPPLIED: Plant Curves 7.19, 7.20, 7.21

NOTE: The value listed in the following step is also listed on "Plant Status and Shutdown Safety Function Status" sheet which is located on the LAN. This sheet will be updated by the WCC or Outage Management as plant conditions change.

8.1.5 Verify by the "Time To Boiling" curves in the Plant Curve Book (curves 7.19, 7.20, 7.21, and 7.22 for the RCS; curves 7.23 and 7.24 for the SFP) that the current values listed on "Plant Status and Shutdown Safety Function Status" sheet are correct AND have the values updated as necessary once per 12 hours, OR any time a greater than 10% change in RCS level, or temperature is made.

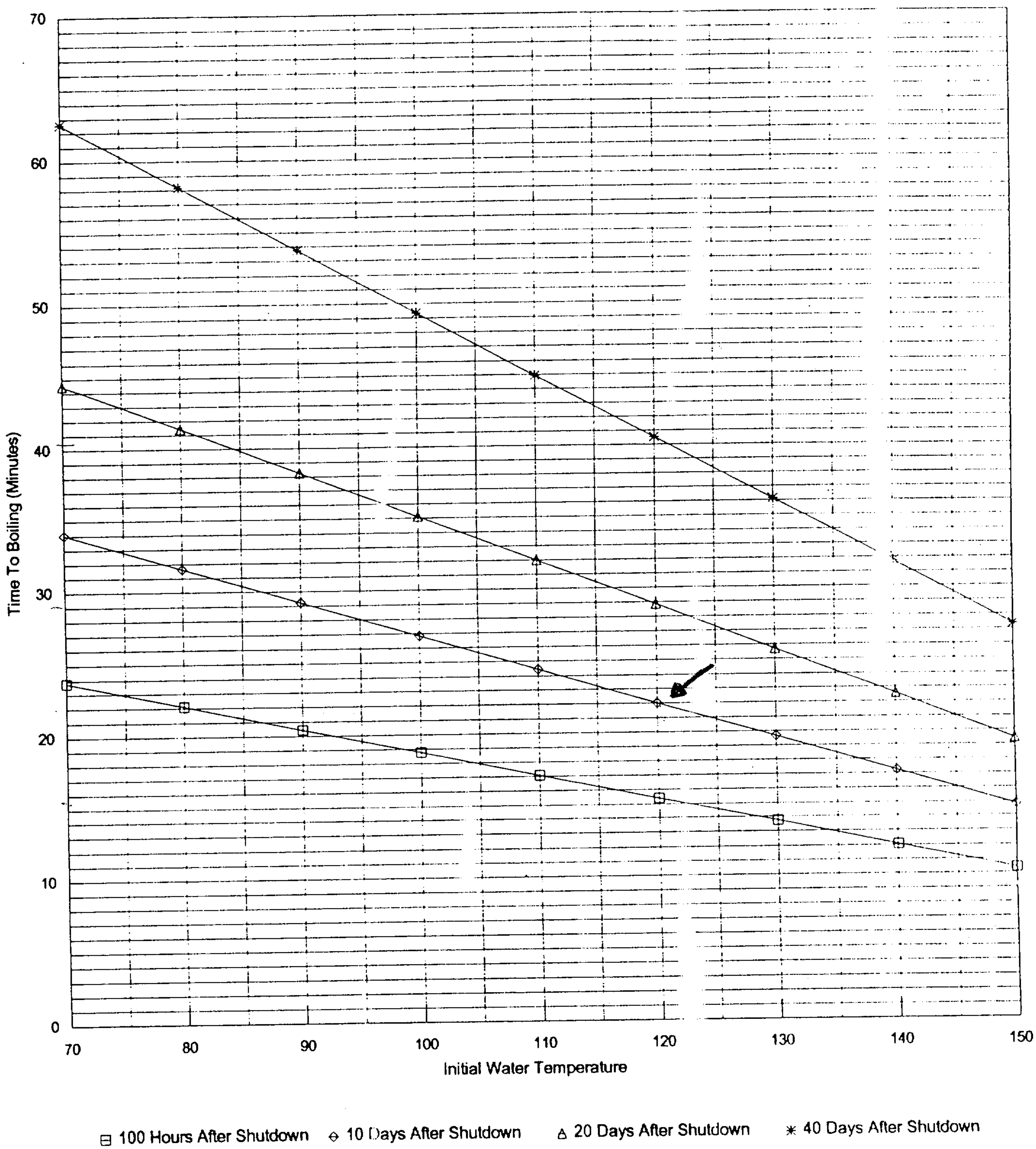
8.1.6 Evaluate, in conjunction with the outage management, the impact of any work request not previously scheduled, on the shutdown safety functions:

- DECAY HEAT REMOVAL
- ELECTRICAL POWER
- INVENTORY CONTROL
- REACTIVITY CONTROL
- RCS PRESSURE CONTROL (with fuel in Containment)
- CONTAINMENT VESSEL STATUS(with fuel in Containment)

8.1.7 Once each shift, verify the minimum Structures, Systems, and Components are in the required status IAW OMP-003, Attachment 10.2, AND initial the correct line on Attachment 10.1.

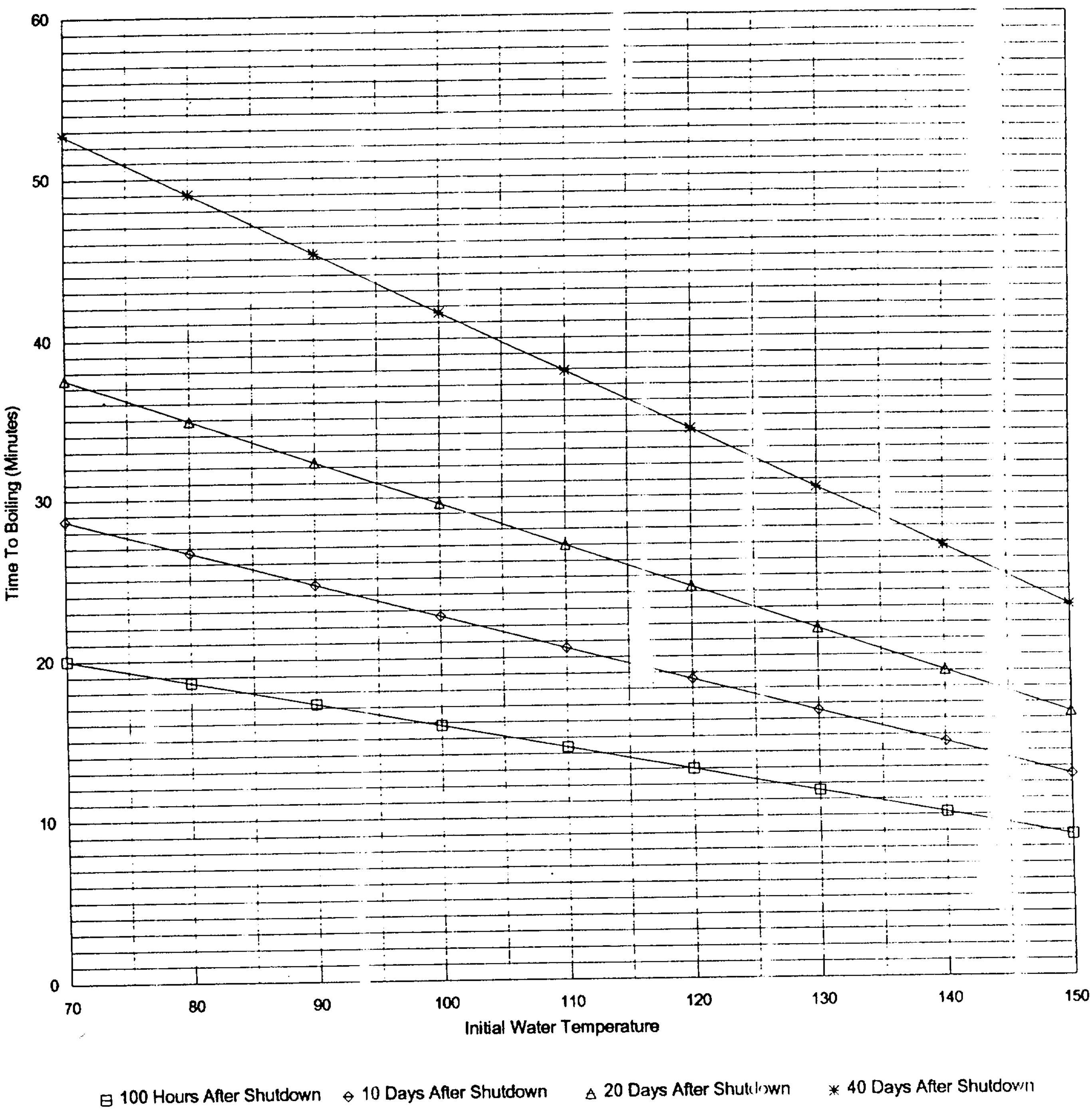
8.1.8 WHEN all the Shutdown Safety Functions are available, THEN completion of Attachment 10.1 may be suspended.

Curve 7.19 - Loss of Residual Heat Removal Cooling Water Level Between 0" to -10" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

Curve 7.20 - Loss of Residual Heat Removal Cooling
Water Level Between -10" to -36" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

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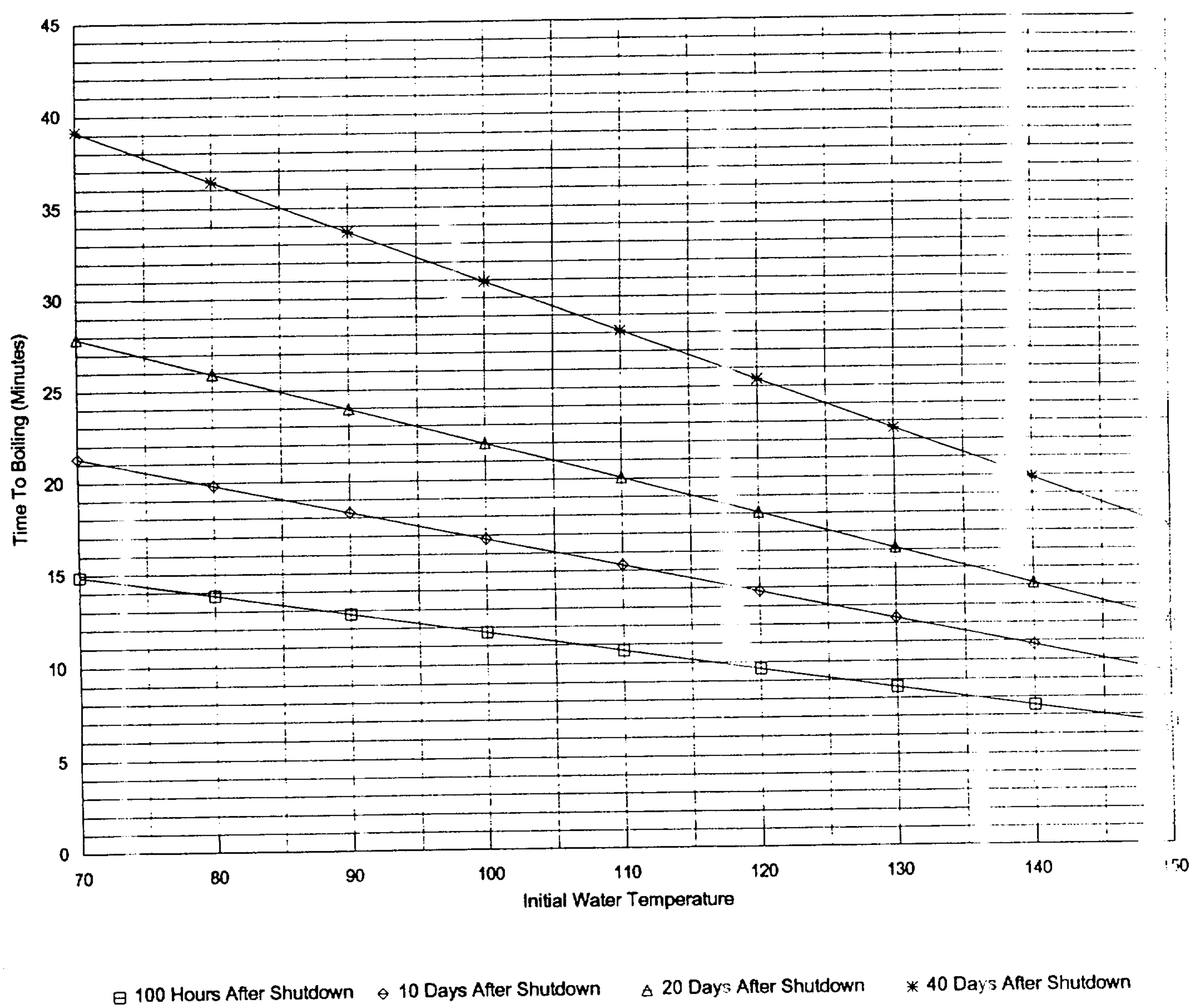
Curve 7.21 - Loss of Residual Heat Removal Cooling Water Level Between -36" to -72" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

OMM-046-04 009

The following plant conditions exist:

- Fuel is in the vessel
- Initial RCS temperature is 150 F
- 40 days after shutdown
- RCS Level is 20" below the vessel flange
- RHR cooling is lost

How long before a steam release occurs?

A. Approximately 20 minutes

✓B. Approximately 23 minutes

C. Approximately 34 minutes

D. Approximately 39 minutes

Question: 21

Given the following conditions:

- A reactor shutdown is in progress.
- APP-005-B2, N-35 LOSS OF COMP VOLT, is received.
- N-35 indicates 6.0x10E-10 amps.
- N-36 indicates 7.0x10E-11 amps.
- N-51 indicates 80 counts.
- N-52 indicates 90 counts.

Which ONE (1) of the following describes the **MINIMUM** action(s) required to obtain Source Range N-31 and N-32 indication?

a. Push **ONLY** the "Train A Source Range Logic Trip Defeat" button

b. Push **ONLY** the "Train A Permissive P-6 Defeat" button

c. Push **BOTH** the "Train A Source Range Logic Trip Defeat" AND the "Train B Source Range Logic Trip Defeat" buttons

d. Push **BOTH** the "Train A Permissive P-6 Defeat" AND the "Train B Permissive P-6 Defeat" buttons

Answer:

d. Push **BOTH** the "Train A Permissive P-6 Defeat" AND the "Train B Permissive P-6 Defeat" buttons

QUESTION NUMBER: 21
TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/2
K/A: 033AA2.11

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Loss of compensating voltage

K/A IMPORTANCE: ***RO*** 3.1 ***SRO*** 3.4
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: NI-08

EXPLAIN the component operation associated with each switch position for the Nuclear Instrumentation System switches and controls.

REFERENCES: APP-005
GP-006

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number NI-08 003

JUSTIFICATION:

a. Plausible since failed IR channel is related to Train A, but both defeat buttons must be pushed.

b. Plausible since failed IR channel is related to Train A, but both defeat buttons must be pushed.

c. Plausible since both buttons must be pushed, but buttons to be pushed are P-6 defeat not trip logic defeat.

d. **CORRECT** Even though only one IR is undercompensated, the circuitry requires that both defeat buttons be pushed to energize the SR instruments.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of NIS system interlocks and controls

REFERENCES SUPPLIED:

5.5 Following a significant (10 ppm or more) change in RCS Boron concentration, additional PZR heaters should be energized. This will permit opening of the PZR spray valves and allow the Boron concentration between the PZR and the RCS loops to equalize.

5.6 Normal spray flow is unlikely or will not occur at all when RCP "C" is stopped and PZR level is less than 30%. Therefore, PZR pressure response may not be as expected. (SCR 90-031)

5.7 Impurities which may be present in the Intermediate Range detectors can prevent the Intermediate Range currents from decreasing to the P-6 reset (10^{-10} amps) in a normal manner. This situation can be identified by observing NI-35, NI-36, NI-51A and NI-52A. If NI-51A and NI-52A are indicating less than 10^2 cps, and NI-35 **OR** NI-36 is not less than 10^{-10} amps, the PERMISSIVE P-6 DEFEAT pushbuttons should be depressed to energize the Source Range detectors. (ACR 92-071)

5.8 When opening disconnect switches, open the blades SLOWLY for the first inch or so, when possible, to make sure there is no power load on it. If there is no load, there will be only a small static discharge, and then the switch may be fully opened. If there is a heavy power arc, the switch should be reclosed to minimize the hazard to the person doing the switching and damage to the equipment. (CP&L Safety Manual)

5.9 ITS SR 3.4.16.2 requires that RCS dose equivalent I-131 specific activity be verified $\leq$ 1.0 µCi/gm within 2 to 6 hours after every thermal power change of $\geq$15% in any one hour period.

- Every time the power level of the Reactor is changed 15% or more in any one hour, E&C shall be notified of the power change, including the time started and the expected duration of the transient. Sample results shall be compared with ITS limits and logged according to Chemistry Procedures. Additionally, E&C shall be notified when the transient is completed.

- A power level change shall be defined for sampling purposes as an absolute value of 15%/hr. in one direction only, (i.e. 95% to 80% = 15%, or 95% to 85% to 90% = 10%). This includes controlled changes, runbacks, transients, and trips that result in changes greater than 15% in any one hour period.

- E&C shall be notified after every 15% power change that is completed in less than one hour. Do not wait until after an hour of changing power before notifying E&C.

ALARM

N-35 LOSS OF COMP VOLT

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Loss of Compensating Voltage on NI-35

OBSERVATIONS

1. Intermediate Range NI

ACTIONS

1. **IF** NI-35 has failed, **THEN** remove NI-35 from service in accordance with OWP-011.
2. **IF** a unit shutdown occurs, **THEN** Source Range NIS will require manual activation.

DEVICE/SETPOINTS

1. None Applicable

POSSIBLE PLANT EFFECTS

1. NI-35 will read higher than actual causing failure of automatic Source Range activation.

REFERENCES

1. ITS Table 3.3.1-1, Item 3
2. CWD B-190628, Sh 441, Cable AL
3. OWP-011, Nuclear Instrumentation (NI)

Question: 22

Given the following conditions:

- The unit is operating at 100% power.
- **NO** scheduled releases are in progress.
- A small leak develops from the bottom of Waste Condensate Tank "A".
- All ventilation systems are in a normal configuration.

An indication that would alert the operators of the accidental liquid release in progress is an increase in the level of monitor ...

a. R-3, PASS Panel Area Monitor.

b. R-4, Charging Pump Room Area Monitor.

c. R-9, Letdown Line Area Monitor.

d. R-14C, Plant Effluent Noble Gas, Low Range Monitor.

Answer:

d. R-14C, Plant Effluent Noble Gas, Low Range Monitor.

QUESTION NUMBER: 22

TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/1

K/A: 059AK2.02

Knowledge of the interrelations between the Accidental Liquid Radwaste and Radioactive-gas monitors

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 2.7

10CFR55 CONTENT: ***55.41(b) RO*** 11 ***55.43(b) SRO***

OBJECTIVE: RM-14

EXPLAIN the effect on the RM System due to selected failures.

REFERENCES: AOP-005
SD-019

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number RM-01 003

JUSTIFICATION:

a. Plausible since the PASS Panel is in the general vicinity of WCT "A". The liquid from the leak will be collected in a sump and will not spill out to the PASS Panel area.

b. Plausible since the Letdown Line Area is in the general vicinity of WCT "A". The liquid from the leak will be collected in a sump and will not spill out to the Letdown Line area.

c. Plausible since the Charging pump room is in the general vicinity of WCT "A". The liquid from the leak will be collected in a sump and will not spill out to the Charging Pump room.

d. **CORRECT** The liquid form the leak will be collected in a sump but the gas that comes out of solution will be exhausted past R-14C by the Auxiliary building exhaust.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of the proximity of radiation monitors to leakage source

REFERENCES SUPPLIED:

c. MD-12(series) Beta/Gamma (GM tube) Detector (R-15)
These detectors use a halogen quenched geiger tube to absorb both beta and gamma particles. The signal is then processed through the preamplifier circuit to the ratemeter. The Overload P.C. Board provides a positive upscale indication of high range instead of the downscale indication normally associated with G.M. Tubes after saturation.

d. MD-51 Gamma Scintillation Detector (R-18)
Similar to MD-5D. This detector uses a sodium iodide (NaI) crystal.

3. Process channels R-30, R-31A, R-31B, and R-31C utilize a model DRM-200A (V15) microprocessor base ratemeter; manufactured by NRC. They use Area monitoring equipment housed in a shielded assembly to measure a specific radiation source.

3.2.2 Nuclear Measurements Corporation (NMC) Monitors (channels R-14A, R-14B, R-14C, R-14D, R-14E, R-22A, R-22B, R-22C, R-23A, R-23B, R-23C, R-38A, R-38B, and R-38C)

3.2.2.1 NMC Channels R-14A, R-14B, R-14C, R-14D, R-14E

Channels R-14A, R-14B, R-14C, R-14D and R-14E. Each channel has a detector which measures the radiation levels and provides a signal to the Programmable Input Output Processor System (PIOPS). In the Low Range Flow Path (normal operation) this monitor will collect and monitor airborne particulate, iodine, and noble gases. In the High Range Flow Path this monitor will collect airborne particulate and iodine prior to the monitoring of Nobles Gases by the intermediate (mid) and high range channels. The major components for the R-14 Skid monitoring system are (FIGURE 9):

1. SKID (Components common to both the Low and High Range Flow Paths)
   a. Sample Inlet
   The inlet and outlet lines to the monitor are 1 inch diameter stainless steel tubing. A "Y" is provided to allow diversion of the sample to the normal range channels or to the high range channels. Each leg of the outlet of the "Y" will have a remote actuated full port valve. The sample is drawn from the plant stack via eight sampling nozzles.

   b. Shield Assembly
   Each detector and its associated collector/sample chamber or prefilter is housed in a lead shield.

   c. Heat Tracing
   - Sample lines from the Plant Stack to the Skid.
   - Sample lines within the Skid.

Accident Channels.

Accident Channels are defined as detector/drawer arrangements, either area or process, that are designed to provide indication during and after an accident when radiation levels and/or environmental specifications of the other area and process channels may be exceeded. (The other Area and Process channels will however, continue to provide indication during and after an accident until the above mentioned limitations are exceeded.)

1. The area RMS (system # 7005).
   Defined as a detector/drawer arrangement in which the detector is exposed or subject to general area radiation.

2. The process RMS (system # 7005).
   Defined as a detector/drawer arrangement in which the detector is housed in a shielded assembly where only a specific radiation source is monitored.

2.3.1 Area RMS (FIGURE 2)

This system consists of twelve channels which monitor radiation levels in various areas of the plant. Two of these channels (R-32A and R-32B) are designated as accident channels.

Channel	Area Monitored
R-1	Control Room
R-2	CV Low Range Monitor
R-3	PASS Panel Area
R-4	Charging Pump Room
R-5	Spent Fuel Building
R-6	Sampling Room
R-7	CV In-core Instrumentation Room
R-8	Drumming Station
R-9	Letdown Line Area
Channel	Area Monitored
R-32A	CV High Range
R-32B	CV High Range
R-33	Monitor Building Area

A typical area channel consists of a detector and a ratemeter. This monitoring system utilizes fixed-position, gamma-sensitive G-M tube detectors (except R-32A and R-32B which use Ion Chambers). The radiation level is indicated locally near the detector (except R-32A and R-32B) and in the Control Room on the ratemeter digital display (R-32A and R-32B have an analog display). Radiation levels are recorded by a multi-point recorder. High-radiation levels and Trouble alarms are annunciated on the RTGB and

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 3

AREA MONITOR R-3 - HEALTH PHYSICS WORK AREA

(Page 2 of 2)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Check Reason For Alarm - KNOWN	With assistance from RC personnel, visually inspect the PASS Panel area for radioactive spills.
10.	Check Radioactive Spill - DETECTED	Go To the main body, Step 1, of this procedure.
11.	Start One Of The Following AUX BUILDING CHARCOAL EXH FANs: • HVE-5A • HVE-5B	
12.	Coordinate With RC Personnel To Control The Spill And Limit Spread Of Contamination	
13.	Go To The Main Body, Step 1, Of This Procedure	

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 4

AREA MONITOR R-4 - CHARGING PUMP ROOM

(Page 1 of 2)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Place VLC Switch To EMERG Position	
2.	Place And Hold EVACUATION ALARM Switch To LOCAL Position For 15 SECONDS	
3.	Announce The Following Over Plant PA System: "ATTENTION ALL PERSONNEL. ATTENTION ALL PERSONNEL. A HIGH RADIATION ALARM HAS BEEN RECEIVED ON CHARGING PUMP ROOM AREA MONITOR, R-4. ALL NON-ESSENTIAL PERSONNEL EVACUATE THE CHARGING PUMP ROOM UNTIL FURTHER NOTICE"	
4.	Repeat PA Announcement	
5.	Place VLC Switch To NORM Position	
6.	Contact RC Personnel To Perform A Survey, As Necessary, To Determine Magnitude Of Radiation Source	
7.	Check Reason For Alarm - KNOWN	With assistance from RC personnel, visually inspect Charging Pump Room for radioactive leaks.
8.	Check Charging Pump Room - LEAK IDENTIFIED	Go To the main body, Step 1, of this procedure.
9.	Start One Of The Following AUX BUILDING CHARCOAL EXH FANs: • HVE-5A • HVE-5B	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 9

AREA MONITOR R-9 - LETDOWN LINE AREA

(Page 1 of 2)

1. Place VLC Switch To EMERG Position

2. Place And Hold EVACUATION ALARM Switch To LOCAL Position For 15 SECONDS

3. Announce The Following Over Plant PA System:

   "ATTENTION ALL PERSONNEL. ATTENTION ALL PERSONNEL. A HIGH RADIATION ALARM HAS BEEN RECEIVED ON LETDOWN LINE AREA MONITOR R-9. ALL NON-ESSENTIAL PERSONNEL EVACUATE AUXILIARY BUILDING UNTIL FURTHER NOTICE"

4. Repeat PA Announcement

5. Place VLC Switch To NORM Position

6. Contact RC Personnel To Perform A Survey In The Following Areas To Determine Magnitude Of Radiation Source:

   - Lower level Aux Building

   - VCT area

7. Verify LTDN ORIFICE Valve(s) - LESS THAN OR EQUAL TO ONE OPEN

8. Control Charging Flow To Maintain PZR Level

Question: 23

Given the following conditions:

- The Control Room has filled with dense smoke from a fire on Unit 1.
- The reactor has been tripped manually by operators.
- The Control Room has been evacuated due to the dense smoke.

Which ONE (1) of the following identifies the procedure(s) that will be **INITIALLY** used to stabilize the plant?

a. EOP Path-1 and EPP-004, Reactor Trip Reponse

b. DSP-002, Hot Shutdown Using the Dedicated/Alternate Shutdown System

c. AOP-004, Control Room Inaccessibility

d. GP-006, Normal Plant Shutdown from Power Operation to Hot Shutdown

Answer:

c. AOP-004, Control Room Inaccessibility

QUESTION NUMBER: 23
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: 068 2.4.11

Knowledge of abnormal condition procedures (Cont Room Evac).

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.6
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: AOP-004-02

RECOGNIZE the selected entry level conditions of AOP-004.

REFERENCES: AOP-004

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-004-02 001

JUSTIFICATION:

a. Plausible since a reactor trip is performed in accordance with AOP-004, but the EOP network is not implemented in the event of a control room evacuation.

b. Plausible since entry may be directed to DSP-002 by AOP-004, but not used initially.

c. **CORRECT** Entry conditions to AOP-004 are met due to requiring evacuation due to life-threatening dense smoke.

d. Plausible since GP-006 is used for normal shutdowns, but actions taken outside the control room are outside the scope of GP-006.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of entry requirements / purpose for abnormal procedures

REFERENCES SUPPLIED:

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE:

This procedure provides instructions in the event conditions in the Control Room require evacuation as deemed necessary by the SSO.

The following assumptions apply to this procedure:

- Offsite power is available.
- All RTGB controls are operational and no failures are expected to occur to the RTGB which preclude safe operation of equipment from outside the Control Room.
- No other accident condition exists within the primary plant requiring the Emergency Operating Procedures OR any other AOP.
- The plant is NOT in Cold Shutdown.

2. ENTRY CONDITIONS:

- DSP-001, Alternate Shutdown Diagnostic.
- Toxic gas in the Control Room.
- Confirmed bomb threat in or adjacent to the Control Room.
- Other life threatening conditions, as determined by the SSO or his designee, that cause the Control Room to be uninhabitable.

- END -

Question: 24

Given the following conditions:

- The unit is operating at 40% power.
- OST-011, "Rod Cluster Control Exercise & Rod Position Indication Monthly Interval," is being performed.
- Annunciator APP-005-E2, ROD CONT SYSTEM URGENT FAILURE, alarms just as Control Bank 'C' rods are being withdrawn.

Which ONE (1) of the following describes this condition and / or the actions that should be taken?

a.
- This is an expected alarm.
- Continue withdrawing Control Bank 'C' rods.

b.
- This makes more than one rod inoperable.
- Trip the reactor and go to PATH-1.

c.
- Place the ROD BANK SELECTOR switch in Manual.
- Restore Tavg to Tref by raising turbine load.

d.
- Place the ROD BANK SELECTOR switch in Manual.
- Restore Tavg to Tref by dilution.

Answer:

d.
- Place the ROD BANK SELECTOR switch in Manual.
- Restore Tavg to Tref by dilution.

QUESTION NUMBER: 24
TIER/GROUP: *RO* 2/1 *SRO* 2/1
K/A: 001K3.01

Knowledge of the effect that a loss or malfunction of the CRDS will have on the CVCS

K/A IMPORTANCE: *RO* 2.9 *SRO* 3.0
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: AOP-001-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to dropped rod, misaligned rod, immovable rod, IRPI failure as directed in AOP-001

REFERENCES: AOP-001

SOURCE: *New* ☒ *Significantly Modified* ☐ *Direct* ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible since this is an action that would be taken if a dropped rod recovery were being performed due to this being an expected alarm, but it is not expected and rods will not move.

b. Plausible since this is an action that would be taken if multiple rods were dropped, but an urgent failure does not indicate that any rods are dropped.

c. Plausible since turbine load adjustments to restore Tavg are permissible, but turbine load should be lowered, not raised.

d. CORRECT Rod bank selector is to be placed in Manual and Tavg restored by adjusting boron concentration (dilution) or turbine load (load reduction).

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of conditions during rod control surveillance to determine required actions

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

IMMOVABLE/MISALIGNED RODS

(Page 1 of 31)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Current Plant Status - MODE 1	Observe the NOTE prior to Step 49 and Go To Step 49.
2.	Verify ROD BANK SELECTOR Switch Position - M (Manual)	
3.	Check Tavg - WITHIN +0.5 TO -2.5°F OF TREF	Adjust Turbine load OR RCS boron concentration to maintain Tavg to within +0.5 to -2.5°F of Tref prior to continuing.
4.	Stop Any Evolutions That Change Reactor Power Except As Called For By This procedure • Turbine load changes • Boron concentration changes	
5.	Check APP-005-E2, ROD CONT SYSTEM URGENT FAILURE - ILLUMINATED	Perform one of the following: • IF an entire bank of rods will NOT move, THEN Go To Step 64. OR • IF individual rod(s) indicate misalignment OR will NOT move, THEN Go To Step 18.
6.	Dispatch An Operator To The Rod Control Room To Check Indications And Alarms At The Following: • Each Rod Control Power Cabinet • Rod Control Logic Cabinet	

Question: 25

Given the following conditions:

- A turbine runback is in progress.
- Power is currently at 93% and lowering as the turbine runback occurs.
- APP-005-D5, OTΔT/OPΔT TURBINE RUNBACK ROD STOP, is illuminated.
- APP-004-E3, OVERTEMP ΔT TRIP, is illuminated.
- All loop ΔT's indicate less than the OTΔT and OPΔT setpoints.
- All OTΔT and OPΔT bistables are extinguished.

Which ONE (1) of the following describes the actions to be taken?

a. Verify the turbine runback stops when power lowers to 90%

b. Verify the turbine runback stops when power lowers to 70%

c. Place the turbine in MANUAL due to a runback circuitry failure

d. Trip the reactor and go to PATH-1

Answer:

d. Trip the reactor and go to PATH-1

QUESTION NUMBER: 25

TIER/GROUP: *RO* 3 *SRO* 3

K/A: 2.4.45

Ability to prioritize and interpret the significance of each annunciator or alarm.

K/A IMPORTANCE: *RO* 3.3 *SRO* 3.6

10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: OMM-001-15-02

DISCUSS the major sections of OMM-001-15

REFERENCES: OMM-001-15

SOURCE: *New* ☒ *Significantly Modified* ☐ *Direct* ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible since a turbine runback is designed to reduce any excessive ΔT condition and a ΔT runback will only occur until the condition is cleared, but the first out alarm during a transient requires a reactor trip.

b. Plausible since a turbine runback is designed to reduce any excessive ΔT condition and some turbine runbacks lower power to 70%, but the first out alarm during a transient requires a reactor trip.

c. Plausible since no indications support the requirement for a turbine runback, but the first out alarm during a transient requires a reactor trip.

d. **CORRECT** With the plant in a transient, any Reactor Trip First Out annunciator requires a reactor trip.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of given conditions to determine first out annunciator requires trip

REFERENCES SUPPLIED:

8.4.2 (Continued)

2. RTGB Annunciators

a. APP-004, First-Out Reactor Trips, annunciators are an indication that a condition exists that has resulted, or should have resulted, in a signal to effect a Reactor Trip or Safeguards actuation. First-Out annunciator alarms require the highest priority and the following immediate response:

1) Announce the alarm.

2) Scan the RTGB for confirmation of a Reactor Trip.

3) **IF** the plant is in a transient condition, **THEN** immediately trip the Reactor and actuate Safeguards as required.

4) **IF** the plant is at steady state conditions, **THEN** perform the following:

.a) **IF** a Reactor Trip has **NOT** occurred, **THEN** scan the RTGB for confirmation that the First-Out annunciator is valid. The scan should include bistable status lights, other annunciators, and process and control indications such as levels and pressures that input to the Reactor protection system.

.b) **IF** the scan supports the need for a Reactor Trip or Safeguards actuation, **THEN** the operator should immediately trip the Reactor and actuate Safeguards as required.

.c) **IF** the scan does **NOT** support a Reactor Trip or Safeguards actuation, **THEN** the operator should clearly and quickly communicate the condition to the SSO/CRSS, who is expected to assist in the diagnosis. Any indication that supports the diagnosis that a trip is required should result in an immediate Reactor Trip. Only if **NO** supporting indication is present is it acceptable to remain at power while troubleshooting and repairs are made.

Question: 25

A "Blue Dot" adjacent to a RTGB instrument indicates the instrument is ...

a. out-of-service for calibration.

b. environmentally qualified.

c. a Technical Specification indication.

d. out of tolerance from a channel deviation check.

Answer:

d. out of tolerance from a channel deviation check.

QUESTION NUMBER: 25

TIER/GROUP: *RO* 3 *SRO* 3

K/A: 2.4.45

Ability to prioritize and interpret the significance of each annunciator or alarm.

K/A IMPORTANCE: *RO* 3.3 *SRO* 3.6

10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: OMM-001-11-02

EXPLAIN the requirements for maintaining operations records and logs in accordance with OMM-001-11

REFERENCES: OMM-001-11

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number OMM-001-??-04 001

JUSTIFICATION:

a. Plausible since the instrument is in an unusual configuration, but blue stickers are reserved for instruments with an identified unacceptable deviation.

b. Plausible since identifying these instruments is vital to post-accident response, but blue stickers are reserved for instruments with an identified unacceptable deviation.

c. Plausible since identifying these instruments is vital to post-accident response, but blue stickers are reserved for instruments with an identified unacceptable deviation.

d. **CORRECT** The blue sticker is used to designate an instrument which has an unacceptable deviation identified.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of administrative requirements for identifying out of service indicators

REFERENCES SUPPLIED:

8.2.4 Corrective Action

1. Once an unacceptable deviation is identified, corrective action consistent with the Plant TECH SPECS, QA requirements, and OWP should be initiated. For instrumentation which also provides an input to an automatic safety function, "Operator Actions" described in TECH SPECS shall be met until proper operation of the safety circuit can be verified.

2. A blue sticker will be placed adjacent to the respective RTGB instrument as a visual reminder to the operator that an unacceptable deviation has been identified.

3. One or more of the following options should be used as a guide in initiating a Work Request (WR).

   a. Declare the channel inoperable, perform applicable "Operator Actions" described in TECH SPECS, and initiate Work Request.

   b. Declare the channel out of tolerance but still available as a trend indicator, perform applicable "Operator Actions" described in TECH SPECS, and initiate Work Request.

   c. Declare the channel deviating by a known constant amount. If the deviation is in a conservative direction, the "Operator Actions described in TECH SPECS need not be inserted; if the deviation is in a nonconservative direction, perform applicable "Operator Actions" described in TECH SPECS, and initiate Work Request.

NOTE: Any trips inserted may be reset after the protection portion of the channel is verified to be operating properly.

4. When maintenance on the instrument has been completed, the blue sticker adjacent to it is removed from the RTGB and discarded and the instrument is no longer carried in the applicable section of the RO/BOP Operators Turnover Checklist.

Question: 26

Given the following conditions:

- A valid alarm has been acknowledged for R-1, Control Room Area Monitor.
- The CRSS has entered AOP-005, Radiation Monitoring System.
- Step 3 of Attachment 1 has the operator stop the HVS-1 Auxiliary Building Supply Fan by opening the supply breaker on MCC-5.

Which ONE (1) of the following is the basis for this step?

a. Ensures that any air-flow will be from the Control Room to the Auxiliary Building

b. Ensures that the air-borne contaminants in the Control Room will be exhausted to the Auxiliary Building for cleanup

c. Ensures that personnel in the Auxiliary Building will **NOT** be exposed to high airborne activity for a prolonged period

d. Ensures that personnel in the Control Room will **NOT** be exposed to high radiation condition for a prolonged period of time

Answer:

a. Ensures that any air-flow will be from the Control Room to the Auxiliary Building

QUESTION NUMBER: 26

TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/2

K/A: 061AK3.02

Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: Guidance contained in alarm response for ARM system

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.6

10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: AOP-005-03

EXPLAIN the basis of selected steps, cautions, and notes in AOP-005.

REFERENCES: AOP-005

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-005-03 005

JUSTIFICATION:

a. **CORRECT** Ensures CR pressure is higher than AB pressure to ensure air flow is out of the CR and into the AB.

b. Plausible since it would be desirable to clean up airborne contaminants, but contaminants are prevented from entering the CR due to the high pressure.

c. Plausible if misconception that AB is maintained at higher pressure, but CR is maintained at higher pressure than AB.

d. Plausible since it would be desirable to maintain low levels of airborne radiation in the CR, but contaminants are prevented from entering the CR due to the high pressure.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of the basis for actions taken in response to a radiation alarm

REFERENCES SUPPLIED:

ATTACHMENT 1, AREA MONITOR R-1 - CONTROL ROOM

Step	Description
1	Upon high radiation alarm on Area Monitor R-1, Control Room ventilation automatically shifts to emergency pressurization mode. This step verifies that the automatic shift has occurred. The discharge dampers are verified closed by observing the pink and blue status lights (when power available) or by locally observing the damper positions (when power is unavailable to status lights).
2	Radiation Control personnel are contacted in response to all abnormal radiation conditions at the Plant. This step provides instruction to contact RC personnel and provides instruction on minimum radiological surveys to be performed.
3	This step opens Breaker AUX BUILDING SUPPLY FAN HVS-1 (MCC-5, CMPT-7J). This breaker must be opened to ensure that Control Room pressure is higher than Aux Building pressure, thereby ensuring that any air flow will be out of the Control Room and into the Auxiliary Building. (This is an NRC commitment per CP&L memo RNP/94-1689.)
4	This standard step provides transition back to the procedure body to address other Radiation Monitor alarms or to exit the procedure.

ATTACHMENT 2, AREA MONITOR R-2 - CV AREA

Step	Description
1	If personnel are not in CV, then performing a CV evacuation is not necessary. This step checks if personnel are in CV and RNO 1 provides transition around steps performing CV evacuation if not required.

Question: 27

Given the following conditions:

- A large break (DBA) LOCA has occurred.
- EPP-15, Loss of Emergency Coolant Recirculation, is being implemented.
- One SI Pump and one RHR pump are running.
- Time after trip and SI is 20 minutes.
- SI **CANNOT** be terminated due to insufficient subcooling.

Given the supplied references, which ONE (1) of the following states the **MINIMUM** SI flow for these conditions?

a. One RHR pump injecting, with flow manually throttled to approximately 260 gpm

b. One RHR pump injecting, with flow manually throttled to approximately 130 gpm

c. One SI pump injecting, with flow manually throttled to approximately 260 gpm

d. One SI pump injecting, with flow manually throttled to approximately 130 gpm

Answer:

c. One SI pump injecting, with flow manually throttled to approximately 260 gpm

QUESTION NUMBER: 27

TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/2

K/A: WE11EK2.2

Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the systems.

K/A IMPORTANCE: ***RO*** 3.9 ***SRO*** 4.3

10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: EPP-015-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to EPP-15.

REFERENCES: EPP-15

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EPP-015-08 001

JUSTIFICATION:

a.		Plausible since determined flow is correct, but flow should be established with SI pump, not RHR pump.
b.		Plausible if 200 minute line is incorrectly used, but actual flow should be maintained above 260 gpm.
c.	**CORRECT**	Using EPP-15, Attachment 1, intersection of 20 minute line with curve identifies minimum required flow as 260 gpm. The RHR pumps are both stopped under these conditions.
d.		Plausible if 200 minute line is incorrectly used, but actual flow should be maintained above 260 gpm.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Application of given data to plant curves to determine SI flow requirements

REFERENCES SUPPLIED: EPP-15, Attachment 1

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

33. Check If SI Can Be Terminated As Follows:

a. Check RCS subcooling - GREATER THAN 85°F [105°F]

b. Check RVLIS indication - GREATER THAN REQUIRED FROM TABLE

RCP STATUS	**REQUIRED RVLIS INDICATION**
ONE RUNNING	40% DYNAMIC RANGE
NONE RUNNING	69% FULL RANGE

RESPONSE NOT OBTAINED:

Establish minimum SI flow to remove decay heat as follows:

1) Determine required SI flow from Attachment 1, Required Flow Rate Versus Time After Reactor Trip.

2) Verify BOTH RHR pumps are stopped.

3) Check SI flow rate on FI-943, COLD LEG HEADER FLOW.

4) Start an additional SI Pump as necessary.

5) Establish communications with operators stationed at the breakers for the BIT OUTLET COLD LEG INJECTION Valves

- SI-870A - MCC-5 (CMPT 10M)
- SI-870B - MCC-6 (CMPT 13J)

6) Coordinate with local personnel to OPEN the appropriate breaker WHEN flow reaches the required value.

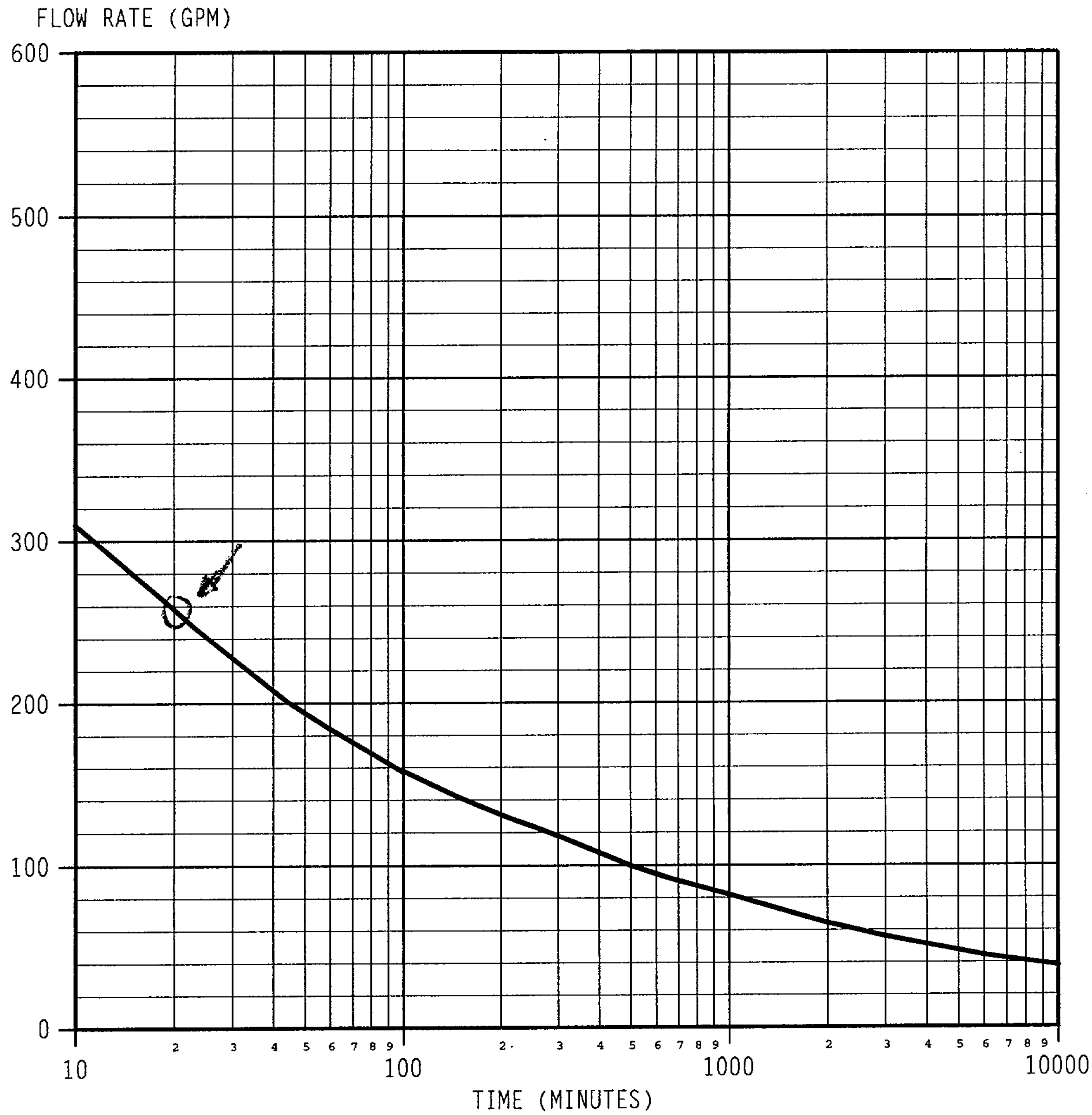
7) As necessary, individually CLOSE BIT OUTLET Valves, SI-870A AND SI-870B.

8) IF necessary, THEN locally throttle SI-870A OR SI-870B.

9) Go To Step 38.

34. Reset CONTAINMENT ISOLATION PHASE A AND PHASE B

ATTACHMENT 1
REQUIRED FLOW RATE VERSUS TIME AFTER REACTOR TRIP
Page 1 of 1



EPP-015-08 001

The following plant conditions exist for a large break (DBA) LOCA when the operators begin to implement EPP-15 Loss of Emergency Coolant Recirculation.

* One SI and One RHR pumps are running
* Time after trip and SI is 30 minutes
* SI cannot be terminated due to insufficient subcooling

Which ONE (1) of the following states the minimum SI flow for these conditions?

A. Only one SI pump and one RHR pump injuecting at full flow.

B. Only one SI pump injecting at full flow.

✓C. Only one SI pump injecting with flow manually throttled to approximately 230 gpm.

D. Only one SI pump inujecting with flow manually throttled to approximately 120 gpm.

Question: 28

Given the following conditions:

- The unit is operating at 24% power during a plant startup.
- Rods are being withdrawn to raise RCS temperature.
- When the IN-HOLD-OUT lever is released, rods continue to step outward.

Which ONE (1) of the following actions should be taken?

a. Place the ROD BANK SELECTOR switch in Automatic and verify rod motion stops

b. Place the ROD BANK SELECTOR switch in Manual and verify rod motion stops

c. Manually trip the reactor in anticipation of an Intermediate Range High Flux Trip and go to PATH-1

d. Manually trip the reactor in anticipation of a Power Range High Flux (Low Setpoint) Trip and go to PATH-1

Answer:

a. Place the ROD BANK SELECTOR switch in Automatic and verify rod motion stops

QUESTION NUMBER: 28
TIER/GROUP: *RO* 1/2 *SRO* 1/1
K/A: 001AA2.03

Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place

K/A IMPORTANCE: *RO* 4.5 *SRO* 4.8
10CFR55 CONTENT: *55.41(b) RO* 6 *55.43(b) SRO*

OBJECTIVE: AOP-001-05

STATE the immediate action steps of AOP-001

REFERENCES: AOP-001

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. CORRECT Automatic rod withdrawal is physically disabled, so placing the switch in Automatic should stop all rod withdrawal.

b. Plausible since automatic rod control is capable above 15% power, but only for automatic rod insertion as automatic rod withdrawal is physically disabled.

c. Plausible since a reactor trip would be required if below 15% or if the correct actions failed to stop rod motion, but IR trip would have been blocked by this point.

d. Plausible since a reactor trip would be required if below 15% or if the correct actions failed to stop rod motion, but PR trip would have been blocked by this point.

DIFFICULTY:
Comprehensive/Analysis [ ] *Knowledge/Recall* [X] *Rating* 3

Knowledge of immediate operator actions for continuous rod motion

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
	NOTE Steps 1 through 4 are immediate actions.	
1.	Check Unwarranted Rod Motion - IN PROGRESS	Go To Step 6.
2.	Check Reactor Power - GREATER THAN 15%	Trip the Reactor and Go To Path 1.
3.	Attempt To Stop Rod Motion As Follows:	
	a. Check ROD BANK SELECTOR Switch position - A (AUTO)	a. Place the ROD BANK SELECTOR Switch in A (Auto). Go To Step 4.
	b. Place ROD BANK SELECTOR Switch in M (Manual)	
4.	Check Unwarranted Rod Motion - STOPPED	Trip the Reactor and Go To Path 1.
5.	Go To Section C, Continuous Rod Motion	
6.	Determine If Multiple Rods Have Dropped As Follows:	
	a. Analyze Indications For Multiple Rod Drop • Prompt Drop - PRESENT • More than 1 Rod Bottom Light - ILLUMINATED • More Than 1 IRPI - INDICATES ON BOTTOM	
	b. Check Multiple Dropped Rods - PRESENT	b. Go To Step 9.
7.	Check Reactor Status - MODE 1 <u>OR</u> 2	Go To Section A, Dropped Rod

DISCUSSION (Continued)

The following are possible indications that a bank has failed to move:

- No change in IRPI or Group Step Counter readings when motion is demanded
- With rods in auto, following a Turbine load or boron concentration decrease, any of the following:
    - Increasing Tavg, RCS pressure and pressurizer level
    - Tavg/Tref deviation indication and alarm

In the case of an individual rod that indicates misalignment or lack of motion, it may or may not be fully known by the Operator the exact nature of the failure. The problem could be a stuck rod, IRPI failure, or misaligned rod. Depending on core location, core flux patterns may not be sufficiently abnormal to indicate a rod alignment problem on the ex-core detectors. Incore flux maps and thermocouple readings would be necessary to confirm a misaligned rod. In the case of a stuck rod, it may not be confirmed until actions have been taken to validate the IRPI and attempts to realign the rod prove unsuccessful.

The procedure has arrange the sections in the main Body to assist the operator in diagnosing an IRPI failure vs a Misaligned or Stuck Rod. The operator should always keep in mind that if not sure that a problem is an IRPI problem, then it should be treated as a misaligned rod.

Section C - Continuous Rod Motion

This section of the procedure is intended to provide the direction necessary to diagnose the cause of unwarranted rod motion and comply with ITS requirements if the rod movement occurred while in Individual bank Select Mode. This section also assures plant power is maintained below 100%. Possible causes of unwarranted rod motion are:

- Out motion - Failure of IN/OUT manual station
- In motion - Failure of IN/OUT manual station
  - Failure of the Automatic Control System

In most cases of unwarranted rod motion, the cause would be accompanied by alarms and indications of the failure. The alarms that would occur and the rate of rod motion are proportional to the cause and extent of the failure. it is not possible to experience a failure in the Automatic Control System causing continuous rod withdrawal. The leads for automatic rod withdrawal have been physically lifted.

INDIVIDUAL STEP DESCRIPTION:

Main Body

Step	Description
N1	This note reminds the operator that the first four steps are immediate actions.
1	This step provides transition for actions in the event of uncontrolled rod movement. The step is noted as "unwarranted" rod movement. It is expected that the operator is familiar with the setup of the rod control system so that he may observe plant conditions and determine that rod motion should not be called for.
2	This step checks plant power for applicability of actions for unwarranted rod movement. If NOT greater than 15% the reactor will be tripped. If less than 15% a startup (or shutdown) is in progress with rods in manual. Uncontrolled rod movement could result in an uncontrolled criticality. Rods can not be placed in automatic when below 15% power, therefore the reactor is tripped. If in Mode 3 the rods are in manual with rod movement, most likely, not in progress (this would be classified as spontaneous uncontrolled manual rod movement). Once again, rods can not be placed in automatic below 15%.
3	This step attempts to stop rod motion. If rods are in manual initially and rod movement begins, the failure is most likely in the in or out selector switches. Note that this includes having the switch in the manual "bank select" position. Placing the rods in automatic will remove those switches from the circuit. If the rods are in automatic when movement occurs placing the switch in manual will remove the automatic circuitry from service.
4	If after moving the switch to a different position, the uncontrolled rod movement continues, a reactor trip is required. Rods movement without control of the operator is a serious condition which could lead to flux anomalies and fuel damage if left unattended.

Question: 29

A Containment Purge is in progress.

Which ONE (1) of the following will automatically terminate the purge on a high radiation signal?

a. R-11, Containment Air and Plant Vent Particulate

b. R-14A, Plant Effluent Particulate

c. R-14C, Plant Effluent Noble Gas Low Range

d. R-16, Containment HVH Cooling Water Radioactive Liquid

Answer:

a. R-11, Containment Air and Plant Vent Particulate

QUESTION NUMBER: 29
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 073A4.01

Ability to manually operate and/or monitor in the control room: Effluent release

K/A IMPORTANCE: ***RO*** 3.9 ***SRO*** 3.9
10CFR55 CONTENT: ***55.41(b) RO*** 9 ***55.43(b) SRO***

OBJECTIVE: RM-09

EXPLAIN the normal operation of the RM control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: AOP-005
SD-019

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number RM-09 003

JUSTIFICATION:

a. **CORRECT** On high radiation level automatically closes CV purge supply and exhaust, as well as the pressure and vacuum relief valves.

b. Plausible since R-14A monitors vent exhaust and CV purge exhaust is monitored by this rad monitor, but no auto actions are associated with R-14A.

c. Plausible since R-14A monitors vent exhaust and CV purge exhaust is monitored by this rad monitor, but auto actions associated with R-14C are to isolate waste gas tank release.

d. Plausible since this would detect a containment high radiation condition, but only if leakage into the cooling water also existed and there are no automatic actions for this monitor.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of automatic actions associated with radiation monitors

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 12

PROCESS MONITOR R-11/R-12 - CV AIR & PLANT VENT

(Page 2 of 3)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Check CONTAINMENT VENTILATION ISOLATION Valves - CLOSED	Perform the following: a. Depress H.V. OFF on R-11 OR R-12 to initiate Containment Ventilation Isolation. b. IF any CONTAINMENT VENTILATION ISOLATION Valve fails to close, THEN locally verify penetration is isolated from outside CV.
9.	Place The Following CV IODINE REMOVAL FAN Control Switches To PREPURGE Position: • HVE-3 • HVE-4	
10.	Check RCS Temperature - GREATER THAN 200°F	Initiate CV closure using OMM-033, Implementation Of CV Closure.
11.	Request RC To Perform A Background Radiation Check At Radiation Monitors R-11 AND R-12	

ATTACHMENT 10.2
Page 1 of 2

RMS INSTRUMENT CONTROL FUNCTIONS

MONITOR	MEDIUM MONITORED	FUNCTION
R-1	Control Room Air	Switches Control Room ventilation into the emergency pressurization operating mode.
R-11	CV Air or Stack Particulate	Closes C.V. purge supply and exhaust; pressure and vacuum relief valves.
R-12	CV Air or Stack Gas	Same function as R-11
R-14C	Stack Gas (Low Range)	Closes waste gas decay tank release valve (RCV-014); swaps R-14 Skid over to high range (two different setpoints).
R-14D	Stack Gas (Mid Range)	Swaps R-14 Skid over to low range.
R-18	Liquid Waste Disposal	Closes waste disposal system liquid release valve (RCV-018)
NOTE The blowdown tank release isolation valve (V1-31) will close if all three SG monitors (R-19A, R-19B and R-19C) are in alarm.		
R-19A	SG "A" Blowdown	Closes; blowdown isolation valves FCV-1930A & FCV-1930B, sample isolation valves FCV-1933A & FCV-1933B, rate flow control valve FCV-4204A.

INFORMATION ONLY

RM-09 003

Which ONE (1) of the following describes process radiation monitor channels that initiate automatic actions?

The following titles are associated with the channel numbers:

R-11:	Containment Air Particulate
R-12:	Containment Air Gas
R-14C	Plant Vent Gas
R-17:	CCW Monitor

	R-11	R-12	R-14C	R-17
A.	Yes	No	No	Yes
B.	No	Yes	Yes	Yes
✓C.	Yes	Yes	Yes	No
D.	No	No	No	No

Question: 30

Given the following conditions:

- Reactor power is 35%.
- All control systems are in automatic.
- Pressurizer level transmitter LT-459 is selected for control.
- A small leak develops across the differential pressure bellows for LT-459, resulting in pressure equalizing across the bellows.

Assuming **NO** operator actions, which ONE (1) of the following describes the instrumentation and plant response to this leak?

	LI-459 PZR LVL	**LI-460 PZR LVL**
a.	Increases	Increases
b.	Increases	Decreases
c.	Decreases	Increases
d.	Decreases	Decreases

Answer:

b.	Increases	Decreases

QUESTION NUMBER: 30
TIER/GROUP: ***RO*** 1/3 ***SRO*** 1/3
K/A: 028AK1.01

Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leak abnormalities

K/A IMPORTANCE: ***RO*** 2.8 ***SRO*** 3.1
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: CVCS-09

EXPLAIN the effect on the CVCS due to selected failures.

REFERENCES: SD-059
Pressurizer LP

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number CVCS-09 019

JUSTIFICATION:

a. Plausible since indicated level on the failed instrument will increase, but actual pressurizer level will lower.

b. **CORRECT** Pressure equalizing across the cell would indicate that water level in the pressurizer is equal to the height of the reference leg. Since this would indicate a high level, charging pump speed would lower, and actual level would lower.

c. Plausible if misconception is that indicated level decreases as differential pressure decrease, but indicated level will increase.

d. Plausible if misconception is that indicated level decreases as differential pressure decrease, but indicated level will increase.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 4

Analysis of pressurizer level malfunction to determine indicated and actual plant response

REFERENCES SUPPLIED:

There is one alarm associated with each channel of LTOPP. It actuates for 3 reasons: (1) RCS temperature is <360°F and LTOPP is not selected on the key switch for OVERPRESSURE PROTECTION, (2) The PORV has received an actuation signal based upon current pressure and temperature or (3) the associated Block valve is shut.

5.1.5 PZR Level Control (PZR-Figure 12)

PZR level is controlled by controlling charging pump speed. The level is programmed to ramp up as Tavg increases by LC-459G. This maintains approximately constant mass in the RCS as Tavg is increased and the coolant in the RCS expands. Level program is 22.2% at Tavg of 547°F and 53.3% at Tavg of 575.4°F.

There are 3 PZR level channels LT-459, LT-460 and LT-461. LC-459G the PZR level controller is normally fed by level channel LT-459 but can be replaced by LT-461 with a selector switch on the RTGB. The output of LC-459G is then fed to the charging pump speed controllers to control speed of the charging pump if their controllers are selected to Auto.

If PZR level increases 5% above program LC-459D will turn on the backup heaters and sound an annunciator for High Level Heaters on.

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would deenergize and any backup heater in manual would remain energized.

(1) Examples:

(a) Break in reference leg - reference pressure decreases - indicated level increases

(b) High temperature in CV - reference pressure decreases - indicated level increases

(c) D/P cell rupture - reference pressure decreases - indicated level increases

c) Cold cal. level at NOP/NOT - reference pressure remains constant - variable leg density/pressure decreases - cold cal. level indicates lower

5. AVAILABILITY OF SPRAY FLOW FOR VARIOUS RCP COMBINATIONS — PZR-FIGURE-14

a) Spray valve 455B off of "C" loop

b) Spray valve 455A taps off of "B" loop

c) Spray performance may be improved with any combination of pumps by raising PZR water level. Normal spray flow is unlikely or will not occur at all when RCP "C" is stopped and PZR level is less than 30%

d) When operating only one RCP in a loop with a spray valve, the idle loop spray valve should be left shut to prevent "short-cycling" of flow back to the idle loop

e) RCP/Spray combinations shown in Figure-14

6. PRESSURE CHANNEL FAILURE — **OBJ. #14**

a) Automatic systems will act same as any pressure change

b) Channel failing high will result in full spray and PORV failing open

c) Must manually shut PORV to prevent trip on low pressure

a) Vent if hydrogen or oxygen >4% by

b) Gaseous Waste Vent Header

c) RC-549

C. ABNORMAL OPERATIONS

1. PZR automatically responds to all abnormal conditions

   a) Automatic controls

   b) Safety valves

2. CONTROLLING CHANNEL FOR PZR LEVEL FAILS HIGH (NO OPERATOR ACTION) — **OBJ. #14**

   a) Charging pump speed ↓ causing PZR level ↓

   b) Letdown isolates and all PZR heaters trip @ 14.4%

   c) Letdown will reinitiate as level ↑, heaters will not

Q	Why will heaters not automatically re-energize ?
A	Breakers must be reset by taking the control switch to OFF then back to AUTO or on

   d) PZR level oscillates and pressure ↓ (no heaters), reactor trip on low pressure, OTΔT runback may be experienced

3. PZR LEVEL AND PRESSURE RESPONSE FOLLOWING A 15 % LOAD REDUCTION

   a) Level and pressure increase

   b) Pressure increase stops before level increase stops

4. PZR LEVEL INDICATION CHANGES DUE TO ABNORMAL TRANSMITTER CONDITIONS — **Draw a picture of a vessel variable and reference legs**

   a) Transmitter works on differential pressure

   b) Anything that causes reference leg pressure to decrease/increase relative to variable leg causes indicated level to go up/down

CVCS-09 019

Given the following plant conditions:

- VCT level is at 20 inches and automatic makeup is in progress
- The level transmitter associated with the automatic level controller (LT-115) fails HIGH
- The Hagan rack switch is in the NORMAL position

Which ONE (1) of the following describes the CVCS system response?

A. LT-112 will override the input from LT-115 for LCV-115A; therefore actual VCT level will remain constant.

B. Actual VCT level will increase once an auto markup signal is established due to input from LT-112.

C. LCV-115C, VCT Outlet valve, will CLOSE and LCV-115B EMERG MU TO CHG SUCT valve will open.

✓D. The operating charging pump(s) will become air bound due to gas intrusion from the VCT.

Question: 31

Given the following conditions:

The plant is being shutdown because of high vibrations on Condensate Pump "A".

- The plant is currently at 65% power.
- Two Main Feedwater Pumps, two Condensate Pumps and a Heater Drain Tank Pump are in service.
- Condensate Pump "A" trips.

Which ONE (1) of the following actions should be taken?

a. Attempt to stabilize the plant at the current power level

b. Attempt to lower turbine load at a rate between 1% minute and 5% per minute and stablize the plant at or below 60% power

c. Attempt to lower turbine load at a rate between 1% minute and 5% per minute and stablize the plant at or below 50% power

d. Trip the reactor and go to PATH-1

Answer:

c. Attempt to lower turbine load at a rate between 1% minute and 5% per minute and stablize the plant at or below 50% power

QUESTION NUMBER: 31
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 056K1.03

Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW

K/A IMPORTANCE: ***RO*** 2.6 ***SRO*** 2.6
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: AOP-010-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in AOP-010 by explaining the basis of each.

REFERENCES: SD-027
APP-007
AOP-010

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AOP-010-03 002

JUSTIFICATION:

a. Plausible since this power level is below the trip requirement for one condensate and one feed pump, but power is to be reduced to less than 50%.

b. Plausible since this would be the correct action if only one feed pump tripped, but power is to be reduced to less than 50% with both a condensate and feed pump trip.

c. CORRECT Under these conditions, a trip of one condensate pump will cause a trip of one FW pump. Maximum allowable power level for one condensate and one feed pump is 50%. A trip is not required since power is below 70%.

d. Plausible since this action would be required if power level was above 70%, but a trip is not required at this level.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Application of given conditions to determine response required to remain within condensate limitations

REFERENCES SUPPLIED:

- High-High steam generator level (2/3 ≥75%), the bypass valve associated with the high-high level closes

4.0 INSTRUMENTATION

4.1 Main Feedwater Pumps

The control switches and indicating lights for the main feedwater pumps are located on the RTGB. The following requirements must be met before a pump will start under normal plant conditions: A condensate pump running, sufficient suction pressure, and sufficient lube oil pressure (8 psig). In addition, if no feedwater pumps are running, then all three block valves must be closed prior to the first feedwater pump start. To start a feedwater pump following a feedwater isolation, all of the above must be met plus the Reactor Trip breakers must be closed and Safety Injection reset, OR place the Feedwater Isolation OVRD/RESET key switches in OVRD/RESET.

A feedwater pump will trip from the following:
(refer to Attachment 10.4 for applicable switches and setpoints)

- Electrical overload
- Undervoltage on its bus
- Minimum flow - blocked for 30 secs. after starting (935 gpm)
- Loss of condensate pump
- Low lube oil pressure (6 psig)
- Low suction pressure (235 psig) coincident with low flow (3100 gpm)
- Safeguards actuation (SI signal)
- Hi-Hi steam generator level (2/3 in 1/3 S/G ≥75%)

To run both main feedwater pumps, both condensate pumps must be running. If only one main feedwater pump is running and it trips due to minimum flow, low lube oil pressure, or electrical overload, the non-running pump will automatically start providing a condensate pump is still running.

Flow switches are provided for each pump to control its recirculating valve (FCV-1444 and FCV-1445) and annunciate alarm conditions. Each valve can be selected to AUTO or OPEN on the RTGB. When in AUTO the valve opens on low flow and closes at a higher flow. This valve will not automatically open unless its associated pump is running, and fails open on loss of power. The valves purpose is to maintain minimum flow through the main feedwater pumps to ensure pump cooling. The valves will open when a low flow condition (1475 gpm) is sensed (e.g. following a reactor trip and feedwater isolation). Upon low flow, the white light adjacent to the control switch on the RTGB also illuminates. When flow increases to 3100 gpm the valve will close.

ALARM

COND PMP A MOTOR OVLD/TRIP

AUTOMATIC ACTIONS

1. **IF** COND PUMP "A" has tripped **AND** COND PUMP "B" is in standby, **THEN** COND PUMP "B" will start.
2. **IF** COND PUMP "A" trips, **THEN** one Feed Pump will trip if two Condensate Pumps **AND** two Feed Pumps were running.

> **NOTE:** The 51ϕB device is set at a lower current value than 51ϕA and 51ϕC. A slow current increase will cause an alarm prior to reaching the long term **OR** short term overcurrent trip setting.

CAUSE

1. Electrical fault trip of Pump Breaker
2. Electrical overload (without breaker trip)

OBSERVATIONS

1. COND PUMP "A" Status Lights
2. S/G Level trends

ACTIONS

1. **IF** COND PUMP "A" has tripped **AND** the Main Generator is in parallel with the grid, **THEN** refer to AOP-010.
2. **IF** COND PUMP "A" has tripped **AND** the Unit is shutdown, **THEN** perform the following:
    1) Verify Automatic Actions listed above occur.
    2) **IF** required, **THEN** feed the S/Gs using AFW Pump(s)
    3) **IF** the cause of the trip is **NOT** known, **THEN** dispatch personnel to inspect the pump **AND** breaker for indications of the cause.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	Go To The Appropriate Step from The Table Below:	

EVENT	STEP
Main Feed Pump Trip	Step 5
Condensate <u>AND</u> Feed Pump Trip	Step 9
Condensate Pump Trip Without MFP Trip	Step 45
Heater Drain Pump Trip	Step 14
Pipe Break / Leak	Step 20
Other	Step 23

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Check Reactor Power - LESS THAN 80%	Trip the Reactor and Go To Path-1.
6.	Check Reactor Power - GREATER THAN 60%	Go To Step 12.
7.	Reduce Turbine Load At 1%/MIN To 5%/MIN To Achieve Less Than 60% Reactor Power	
8.	Go To Step 12	
9.	Check Reactor Power - LESS THAN 70%	Trip the Reactor and Go To Path-1.
10.	Check Reactor Power - GREATER THAN 50%	Go To Step 12.
11.	Reduce Turbine Load At 1%/MIN To 5%/MIN To Achieve Less Than 50% Reactor Power	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
12.	Check Main Feed Pumps - AT LEAST ONE RUNNING	IF Reactor Power is greater than 10%, THEN trip the Reactor and Go To Path-1. IF Reactor Power is less than 10%, THEN trip the Turbine and Go To AOP-007, Turbine Trip Below P-7.
13.	Go To Step 36	
14.	Reduce Turbine Load At 1%/MIN To 5%/MIN To Achieve Reactor Power Less Than The Target Power Per The Following Table:	

PUMPS RUNNING			**TARGET POWER**
Main FWP	Cond	Htr Drn	Percent
2	2	1	85%
2	2	0	80%
1	2	0 or 1	60%
1	1	0 or 1	50%

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
15.	Check Main FW Pumps - TWO PUMPS RUNNING	Observe NOTE prior to Step 17 and Go To Step 17.
*16.	Check HCV-1459, LP HEATERS BYP - OPEN	Perform the following: a. Monitor Condensate Pumps header pressure on PI-1458. b. IF pressure decreases to less than the applicable setpoint, THEN verify OPEN HCV-1459. • Any HDP Running - 300 psig • No HDPs Running - 350 psig

AOP-010-03 002

Given the following plant conditions:

- The plant is being shutdown because of high vibrations on Condensate Pump "A"
- The plant is currently at 30% power
- Two Main Feedwater Pumps, two Condensate Pumps and a Heater Drain Tank Pump are in service
- Condensate Pump "A" trips

Which ONE (1) of the following describes the expected plant response?

A. Both Main Feedwater Pumps will trip resulting in a Reactor trip due to low Steam Generator level.

✓B. One Main Feedwater Pump will trip but sufficient Feedwater flow exists to maintain Steam Generator level.

C. One Main Feedwater Pump will trip which will result in insufficient Feedwater flow to maintain Steam Generator level.

D. A Turbine run back will occur bringing Steam flow in-line with Feedwater flow.

Question: 32

Given the following excerpt from OP-922, "Post Accident Containment, Hydrogen Reduction/Venting System", and the following conditions:

- A design basis LOCA occurred 90 days ago.
- Hydrogen Concentration (Hydrogen Monitor Reading) is 2.5%.
- The H_2 Recombiner System is unavailable for Containment Hydrogen Reduction.

From OP-922:

"5.2.8 Determine the following data:

1. H_2 generation rate from Curve Book, Curve 7.16, Total Hydrogen Generation Rate From All Sources.

- Time following DBA _____ Days
- H_2 Generation Rate _____ SCFM (Curve 7.16)

2. H_2 Concentration from Containment Hydrogen Monitor located in the Control Room or from analysis of Containment samples:

- H_2 Concentration _____ %

5.2.9 Calculate the required exhaust flow:

1. $$Qe = 2400 \frac{G}{C}$$

- Qe is exhaust flow in SCFM
- G is H_2 Generation rate
- C is H_2 Concentration

Required exhaust flow _____ SCFM

NOTE: The Containment Air Exhaust Line (PACV "B") should be used in preference to the Pressure Relief Line (PACV "A").

Given the supplied references, in order to provide required exhaust flow through preferred exhaust path (Containment Air Exhaust), Containment pressure should be raised to approximately ...

a. 0.9 psig.

b. 1.1 psig.

c. 3.7 psig.

d. 4.6 psig.

Answer:

a. 0.9 psig.

QUESTION NUMBER: 32
TIER/GROUP: *RO* 2/3 *SRO* 2/2
K/A: 028A1.02

Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure

K/A IMPORTANCE: *RO* 3.4 *SRO* 3.7
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: CVHVAC-09

EXPLAIN the normal operation of the CV HVAC, PACV and H^2 Reombiner control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES:
OP-922
Plant Curve 7.6
Plant Curve 7.16

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number CVHVAC-09 009

JUSTIFICATION:

a. **CORRECT** Using Curve 7.16, determine intersection of 90 day line as 0.25 scfm. Performing calculation, determine required vent flow rate is 240 scfm. Using Curve 7.6, determine intersection of 240 scfm and PACV-B to be 0.9 psig.

b. Plausible since performed correct until using Curve 7.6 and uses PACV-A instead of PACV-B, which is the preferred method.

c. Plausible if misread Curve 7.16 as 0.5 instead of 0.25. Calculation would then result in 480 scfm. Using PACV-B on Curve 7.6 would result in this response.

d. Plausible if misread Curve 7.16 as 0.5 instead of 0.25. Calculation would then result in 480 scfm. Then using PACV-A on Curve 7.6 would result in this response.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 4

Calculation of containment pressure requirements based on application of given conditions to use of plant curves

REFERENCES SUPPLIED: Plant Curves 7.6 and 7.16

INIT

5.2.7 At least one of the following operable for pressurizing the CV: (N/A method not used)

1. Station Air System components to pressurize the CV, including Station Air Compressor, piping and valves to the CV. ____

2. Instrument Air System components to pressurize the CV, including an Instrument Air Compressor, piping and valves to the CV (an Instrument Air Prefilter and Dryer should be used if available, however, they can be bypassed). ____

> **NOTE:** The Containment H_2 Concentration can be determined by chemical analysis of samples collected IAW Section 8.1 if the Containment Hydrogen Monitor is inoperable.

5.2.8 Determine the following data:

1. H_2 generation rate from Curve Book, Curve 7.16, Total Hydrogen Generation Rate From All Sources.

   - Time following DBA ______ Days 40

   - H_2 Generation Rate __2.5__ SCFM (Curve 7.16) ____

2. H_2 Concentration from Containment Hydrogen Monitor located in the Control Room or from analysis of Containment samples:

   - H_2 Concentration __2.5__ % Samples / Monitor (Circle one) ____

INIT

5.2.9 Calculate the required exhaust flow:

1. $Qe = 2400 \frac{G}{C}$

- Qe is exhaust flow in SCFM

- G is H_2 Generation rate

- C is H_2 Concentration

Required exhaust flow _____ SCFM ____

> **NOTE:** The Containment Air Exhaust line (PACV "B") should be used in preference to the Pressure Relief line (PACV "A").

5.2.10 Determine the required Containment pressure from Curve Book, Curve 7.6, System Resistance Curve to obtain required exhaust flow.

Required pressure for PACV "B" = ________ psig

Required pressure for PACV "A" = ________ psig ____

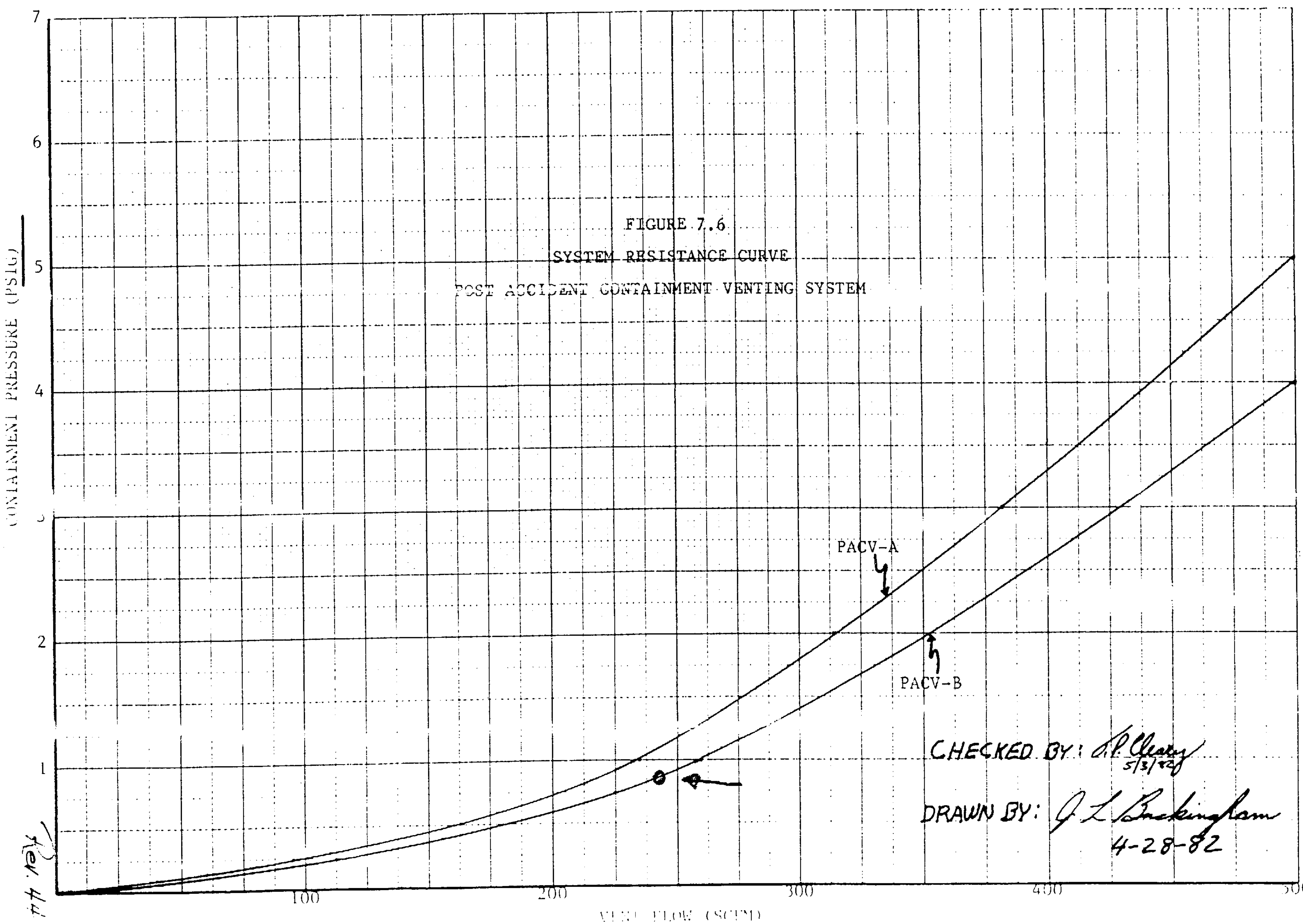
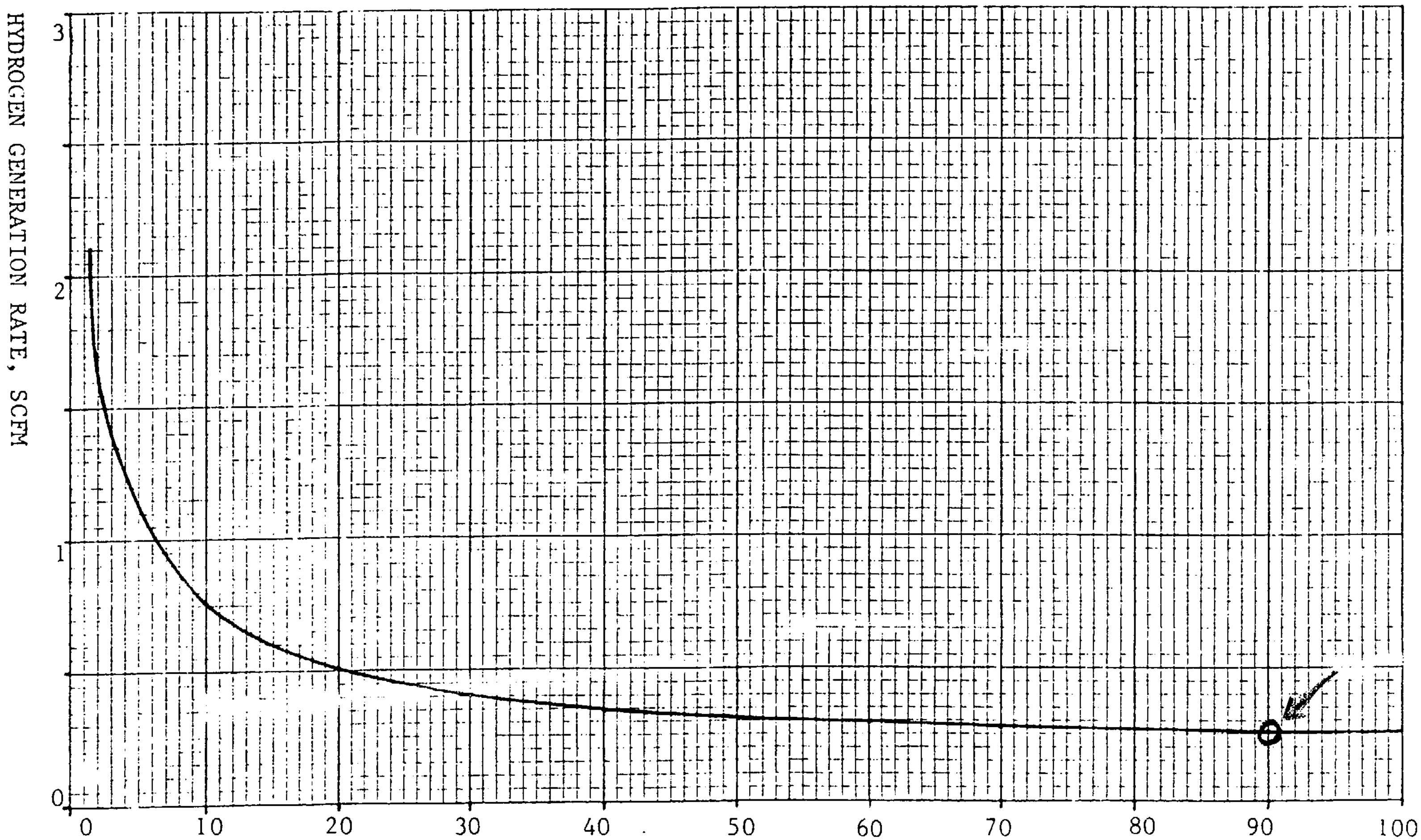


FIGURE 7.6
SYSTEM RESISTANCE CURVE
POST ACCIDENT CONTAINMENT VENTING SYSTEM
CONTAINMENT PRESSURE (PSIG)
VENT FLOW (SCFM)
PACV-A
PACV-B
1
2
3
4
5
6
7
100
200
300
400
500
CHECKED BY: 5/3/82
DRAWN BY: 4-28-82
Rev. 44

9/



CURVE 7.16 TOTAL HYDROGEN GENERATION RATE FROM ALL SOURCES.

830 Day Full Power TID Core
DBA Conditions
Hydrogen Sources:

- Zirconium-Water Reaction
- Aluminum Corrosion
- Core Solution Radiolysis
- Sump Solution Radiolysis

Drawn By: James M. [illegible] 10-19-84

Checked By: [illegible] 10/19/84

CVHVAC-09 009

A design basis LOCA has occurred. The following data has been obtained:

- Time following the LOCA 10 days
- Hydrogen Concentration (Hydrogen Monitor Reading) 5.8%
- Containment pressure 0.2 psig

Which ONE (1) of the following is the correct pressure that the containment should be raised in order to vent the containment using the preferred path? (OP-922 is provided for reference).

✓A. 1.5 psig

B. 1.9 psig

C. 2.0 psig

D. 2.5 psig

Question: 33

Which ONE (1) of the following Fire Brigade qualified personnel would normally serve as the Fire Brigade Team Leader in the event of a fire in the Auxiliary Building of Unit 2?

a. Fire Protection Auxiliary Operator

b. WCC Senior Reactor Operator

c. Unit 1 Superintendent Shift Operations

d. Environmental & Radiation Control Supervisor

Answer:

b. WCC Senior Reactor Operator

QUESTION NUMBER: 33

TIER/GROUP: *RO* 3 *SRO* 3

K/A: 2.4.26

Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.

K/A IMPORTANCE: *RO* 2.9 *SRO* 3.3

10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: OMM-002-03

DISCUSS each section of OMM-002, when possible, using the information given in each section of the procedure.

REFERENCES: OMM-002

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number OMM-002-03 002

JUSTIFICATION:

a. Plausible since function is to be on-shift fire protection expert, but leader must be a licensed operator.

b. **CORRECT** Normally the WCC SRO fills this position, although any licensed operator can serve as leader if qualified.

c. Plausible since this position acts as an advisor to the leader during any fire on Unit 1, but leader must be a licensed operator.

d. Plausible since this position will provide guidance for the radiological considerations associated with a fire in an RCA, but leader must be a licensed operator.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of administrative requirements for makeup of fire brigade

REFERENCES SUPPLIED:

4.0 **PREREQUISITES**

4.1 Definitions

4.1.1 Operations Staff Support (Fire Protection Program) - Personnel assigned responsibility for the implementation of the Fire Protection Program.

4.1.2 Fire Brigade - Those persons designated by the Plant General Manager who comprise the Plant Fire Brigade for each shift of operations. Each shift Fire Brigade is comprised of a Team Leader and at least four qualified brigade members of which at least two must have additional knowledge as described in step 8.6.2.

4.1.3 Fire Brigade Member - Those persons that have been designated by the Plant General Manager and maintain required security access, current medical qualification, active status by successful completion of required training and participation in drills.

4.1.4 Fire Brigade Team Leader - Unit 2 - Normally, the Work Control Center Senior Reactor Operator who is qualified as a Team Leader and is in charge of the Fire Brigade and the emergency scene. Any licensed Operator can serve as Team Leader if qualified. The Unit 1 Superintendent Shift Operations/Shift Leader will act as advisor to the Fire Brigade Team Leader concerning a fire on Unit 1.

3.2 The Fire Protection Auxiliary Operator is responsible to the Superintendent Shift Operations for:

3.2.1 Performing routine fire inspections of Unit 2 to ensure compliance with fire protection procedures.

3.2.2 Conducting and documenting periodic inspections, shift rounds, tests, and preventive minor maintenance of fire protection systems and equipment to ensure proper operational condition.

3.2.3 Being qualified as a Fire Brigade member. During a fire emergency will function as an advisor to the Fire Brigade Team Leader and can function in any capacity on the Fire Brigade as directed by the Team Leader.

3.2.4 Supervising and following-up all valve closures or impairments to any fire protection systems or equipment to ensure adequate back-up protection is provided as required by FP-012 and to prevent extended or unnecessary impairments. Notifying the Superintendent Shift Operations of the above situations.

3.2.5 Functioning as advisor to the Superintendent Shift Operations concerning any fire protection matter.

3.2.6 Preparing Fire Reports in accordance with FP-002.

3.2.7 Complete OMM-001-12 and OMM-007 as applicable.

3.3 The on-duty Superintendent Shift Operations is responsible for:

3.3.1 Operation of the fire detection and fire suppression systems in accordance with FP-012 and FP-013 and established procedures.

3.3.2 Ensuring at least five Fire Brigade members are available in accordance with step 8.6.2 and OMM-001.

3.3.3 Providing general direction and support to the Fire Brigade Team Leader in the event of a fire. (If the Emergency Coordinator is activated in accordance with PLP-007 of the Plant Operations Manual, this general guidance and support may be provided by the Emergency Coordinator in lieu of the Superintendent Shift Operations).

3.11.2 Appendix R Safe Shutdown Engineer:

1. Has overall responsibility for the Appendix R Safe Shutdown Program, providing direction to other plant Sections who support and assist in the implementation, maintenance and surveillance of the program.

2. Review procedural or programmatic changes that affect Safe Shutdown.

3. Ensures compliance with regulatory requirements concerning safe shutdown.

4. Provides design and engineering functions for safe shutdown systems.

5. Reviews plant modifications for impact on the Appendix R Program.

6. Prepare and maintain procedures and instructions which the Appendix R Engineer sponsors.

3.12 Environmental & Radiation Control (E&RC) is responsible for:

3.12.1 Providing a minimum of one fire brigade member per shift as needed to support the minimum Fire Brigade compliment.

3.12.2 Periodic monitoring of the breathing air quality.

3.12.3 Distribution of dosimetry to arriving off-site fire fighting personnel.

3.13 Maintenance is responsible for:

3.13.1 Maintenance of fire protection equipment and systems.

3.13.2 Installing and maintaining the fire barrier penetration seals, fire barrier materials, fire wraps and insulating materials.

3.13.3 Providing a minimum of one fire brigade member per shift as needed to support the minimum Fire Brigade compliment.

OMM-002-03 002

Which ONE (1) of the following personnel would normally serve as the Fire Brigade Team Leader in the event of a fire in Unit 2?

A. Fire Protection Technical Aide.

✓B. Off Control Operator.

C. Unit 1 Shift Supervisor.

D. RC Fire Support.

Question: 34

Given the following conditions:

- The unit is operating at 100% power.
- APP-001-F7, INST AIR HDR LO PRESS, has illuminated.
- AOP-017, "Loss of Instrument Air", is being implemented.
- Instrument air pressure currently reads 79 psig and slowly decreasing.
- The Station Air Compressor is running.

SA to IA cross connect ...

a. valve, SA-5 will automatically OPEN to pass SA through the IA aftercoolers and separators to remove contaminants prior to passing into the IA header.

b. bypass filter isolation valves, SA-220 & SA-221, will automatically OPEN to pass SA through a filter to remove contaminants prior to passing into the IA header.

c. valve, SA-5 will be manually OPENED to pass SA through the IA aftercoolers and separators to remove contaminants prior to passing into the IA header.

d. bypass filter isolation valves, SA-220 & SA-221, will be manually OPENED to pass SA through a filter to remove contaminants prior to passing into the IA header.

Answer:

d. bypass filter isolation valves, SA-220 & SA-221, will be manually OPENED to pass SA through a filter to remove contaminants prior to passing into the IA header.

QUESTION NUMBER: 34
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 079K1.01

Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: IAS

K/A IMPORTANCE: ***RO*** 3.0 ***SRO*** 3.1
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: AIR-14

EXPLAIN the effect on the Instrument and Station Air System due to selected failures.

REFERENCES: AOP-017

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AIR-03 007

JUSTIFICATION:

a. Plausible because SA-5 is opened in AOP-017 as an RNO, but it is must be done manually.

b. Plausible because SA-220 & 221 are opened in AOP-017, but they must be done manually.

c. Plausible because SA-5 is opened in AOP-017 as an RNO, but it does not go through the IA aftercoolers and separators.

d. **CORRECT** The preferred method is to open SA-220, SA-221 and verify open IA-18. This will allow the Service Air to pass through a filter to remove contaminants prior to passing into oil free Instrument Air Header.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of IA / SA system automatic actions

REFERENCES SUPPLIED:

Step	Description
2	Step 2 checks Instrument Air (IA) Header pressure less than 60 psig. Through research it has been determined that major system components required for power operation (i.e. feed reg valves, letdown valves) will start to drift closed. This step is to satisfy an INPO comment that stated "The common industry approach is to direct the operators to manually trip the reactor at a specific decreasing Instrument Air pressure". If pressure does decrease to less than 60 psig, then the operator is directed to trip the reactor in Step 3.
3	This step accomplishes two very important action items. First the reactor is manually tripped by the operator. This is in anticipation of loss of control to various air operated valves and their subsequent failing closed (air pressure less than 60 psig). The second action item transitions the operator to concurrently perform PATH-1 while continuing with this procedure.
4-5	During normal plant operations Air Compressor D or the Primary Air Compressor should operate to maintain header pressure. Air Compressor D is usually the lead compressor running continuously to maintain system pressure. This step is intended for the operator to start any available compressor that is in standby. It is assumed that Instrument Air Compressors A & B will be running in Auto if power is available.
6	This continuous action step checks pressure less than 80 psig. If less than 80 psig the operator is directed to steps that would further increase the supply of air into the IA system.
N7	This note describes the location of IA-3821 to help expedite the task performed.
7	Entering into this step signifies that Instrument Air problems have deteriorated to a point where air pressure is now less than 80 psig. The operator should be prepared for this since he would have received an Instrument Air low pressure alarm at 85 psig. Exiting this step we should find that: (1) Station Air Compressor is backing up the Instrument Air System. (2) Air dryers have been bypassed. (3) Station Air and Instrument Air Compressors are running The first substep requires the verification that the Station Air Compressor is running. If the compressor can not be started the RNO will bypass steps that cross-connect Station Air with Instrument Air. The second and third substep directs the operator to cross-connect the Station Air Header with the Instrument Air Header. Two methods are available to achieve this step: (1) The preferred step is addressed in the left column of this procedure (Open SA-220, SA-221 and verify open IA-18). This will allow the Service Air to pass through a filter to remove contaminants prior to passing into oil free Instrument Air Header.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

IA-3821 is located on IA Dryer D.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Dispatch Operator(s) To Perform The Following:	
	a. Verify Station Air Compressor - RUNNING	a. Go To Step 7.c.
	b. Verify the following SA TO IA CROSS CONNECT BYPASS FILTER ISOLATION Valves - OPEN: • SA-220 • SA-221	b. Open SA-5, STATION AIR TO INST AIR CROSS CONNECT.
	c. Verify IA-18, AIR DRYER "A" & "B" BYPASS - OPEN	
	d. Verify the following Compressors - RUNNING • STATION AIR COMP • INST AIR COMP A • INST AIR COMP B	
	e. Check FCV-1740, AIR DRYER HIGH DP FLOW CONTROL Valve - OPEN	e. Open IA-3665, AIR DRYER "A" & "B" BYPASS.
	f. Open IA-3821, INSTRUMENT AIR DRYER "D" BYPASS	

AIR-03 007

Given the following plant conditions:

- The Unit is at 100% power
- APP-001-F7, INST AIR HDR LO PRESS, has illuminated
- AOP-017, LOSS OF INSTRUMENT AIR, is in use
- Instrument air pressure currently reads 79 psig and slowly decreasing

Which ONE (1) of the following describes the correct response to the decreasing air pressure?

SA to IA cross connect:

A. valve, SA-5 will automatically OPEN to pass SA through the IA aftercoolers and separators to remove contaminants prior to passing into oil free IA header.

B. bypass filter isolation valves, SA-220 & SA-221, will automatically OPEN to pass SA through a filter to remove contaminants prior to passing into oil free IA header.

C. valve, SA-5 will be manually OPENED to pass SA through the IA aftercoolers and separators to remove contaminants prior to passing into oil free IA header.

✓D. bypass filter isolation valves, SA-220 & SA-221, will be manually OPENED to pass SA through a filter to remove contaminants prior to passing into oil free IA header.

Question: 35

Given the following conditions:

- The unit was operating at 100% with bank D rods at 218 steps when a failure of 'B' inverter occurred.
- Instrument bus 3 de-energized.
- **NO** reactor trip occurred.
- Rods **CANNOT** be withdrawn.

Which ONE (1) of the following is preventing rod motion?

a. Power range flux rod stop

b. Intermediate range flux rod stop

c. Overtemperature ΔT rod stop

d. Overpower ΔT rod stop

Answer:

a. Power range flux rod stop

QUESTION NUMBER: 35

TIER/GROUP: *RO* 1/1 *SRO* 1/1

K/A: 057AA2.20

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: Interlocks in effect on loss of ac vital electrical instrument bus that must be bypassed to restore normal equipment operation

K/A IMPORTANCE: *RO* 3.6 *SRO* 3.9

10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: NI-06

LIST power supplies for the major components of the Nuclear Instrumentation System as listed in the EDPs.

REFERENCES: AOP-024

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number NI-09 009

JUSTIFICATION:

a. CORRECT Loss of power to PR channel 3 causes 1/4 overpower rod stop actuation.

b. Plausible since IR channels can prevent rod withdrawal, but IR channels not powered by IB 3.

c. Plausible since OT ΔT can prevent rod withdrawal, but does not actuate on power loss and coincidence is 2/4.

d. Plausible since OP ΔT can prevent rod withdrawal, but does not actuate on power loss and coincidence is 2/4.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of the effect of the loss of a single instrument bus on rod control

REFERENCES SUPPLIED:

CONTINUOUS USE

ATTACHMENT 4

EXTENDED LOSS OF INSTRUMENT BUS 3 (AND 8)

(Page 2 of 3)

4. Restore Rod Control as follows:

   a. At the Miscellaneous Control And Indication Panel, place ROD STOP BYPASS Selector Switch (for PR 41 & PR 43) to BYPASS PR 43 position.

   b. Position the Control Rods as necessary to control Axial Offset and RCS temperature.

5. IF CHARGING PUMP C is in service, THEN perform the following:

   a. Start CHARGING PUMP A OR B.

   b. Stop CHARGING PUMP C.

6. Place The Control Switch For R-11/R-12 Vacuum Pump To STOP.

NOTE
Lost instrumentation/control is described in EDP-008, Instrument Busses.

7. Select and monitor alternate instrumentation.

8. Notify Maintenance to determine and correct the cause of Instrument Bus failure.

* 9. IF power can NOT be restored to Instrument Bus 3 within 2 HOURS THEN place the Plant in Mode 3 within the following 6 HOURS AND Mode 5 within the following 36 HOURS.

10. IF APP-005-B5, ROD BANKS A/B/C/D LO LIMIT, is ILLUMINATED, THEN Borate to clear the alarm using OP-301, Chemical and Volume Control System (CVCS), while continuing with this procedure.

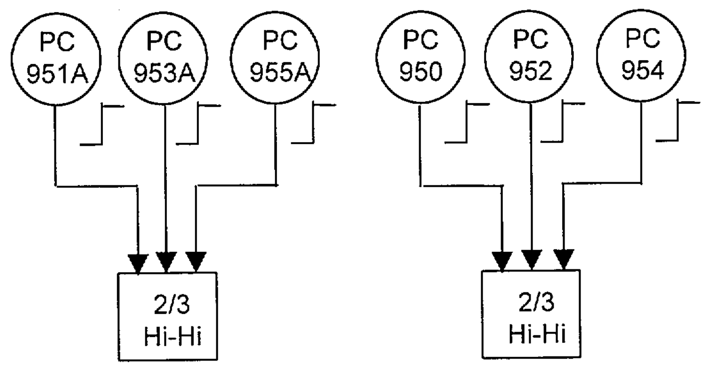
11. Maintain stable plant conditions until Instrument Bus 3 (and 8) power is restored.

12. WHEN power is restored to Instrument Bus 3 (and 8), THEN observe the NOTE prior to Step 13 and Go To Step 13.

Question: 41

Given the following conditions:

- Power has been lost to Containment Pressure channel 954.
- Containment Pressure transmitter PT-950 has failed low.
- **NO** actions in OWP-032, "Containment Pressure," have been performed.
- A large break LOCA occurs and actual Containment Pressure reaches 21 psig.



Which ONE (1) of the following describes the response of the Containment Spray system?

a. **NEITHER** train of Containment Spray will automatically actuate

b. **ONLY** Train 'A' of Containment Spray will automatically actuate

c. **ONLY** Train 'B' of Containment Spray will automatically actuate

d. **BOTH** trains of Containment Spray will automatically actuate

Answer:

a. **NEITHER** train of Containment Spray will automatically actuate

QUESTION NUMBER: 41
TIER/GROUP: *RO* 2/2 *SRO* 2/2
K/A: 026A1.01

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment pressure

K/A IMPORTANCE:	*RO*	3.9	*SRO*	4.2
10CFR55 CONTENT:	*55.41(b) RO*	8	*55.43(b) SRO*	

OBJECTIVE: CSS-09

EXPLAIN the normal operation of the CSS control systems. Include function, instrumentation, interlocks, annunciators, and set points.

REFERENCES: SD-024

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** Two-of-three high pressure conditions on both sets of pressure transmitters are required to generate a Containment Spray signal. Bistables are energized to actuate so only one set will generate the required signal.

b. Plausible since the minimum coincidence is met for a single train of pressure transmitters, but require both sets tripped to generate a signal. Bistables are energized to actuate so only one set will generate the required signal.

c. Plausible since the minimum coincidence is met for a single train of pressure transmitters, but require both sets tripped to generate a signal. Bistables are energized to actuate so only one set will generate the required signal.

d. Plausible since the minimum coincidence is met for a single train of pressure transmitters, but require both sets tripped to generate a signal. Bistables are energized to actuate so only one set will generate the required signal.

DIFFICULTY:
Comprehensive/Analysis [X] *Knowledge/Recall* [ ] *Rating* 3

Analysis of failures on Containment Spray actuation signal

REFERENCES SUPPLIED:

4.2.2 Spray Header Flow "A", FT-958A and "B", FT-958B

The purpose of these flow transmitters is to provide Spray Header "A" and "B" flow indication. They are located on the RTGB and have a range of 0-1500 gpm.

4.3 CV Pressure

NOTE: The CV pressure transmitters are not part of the Spray system but are listed for information. (See SD-006, Engineered Safety Features)

There are nine (9) transmitters located in the Aux. Bldg. near the IVSW tank area. Three are used for the (2 out of 3) HI pressure SI signal at 4 psig (PC-951B, 953B, & 955B). Six supply the HI-HI pressure signal actuation at 20 psig (PC-950, 951A, 952, 953A, 954 & 955A). (CSS-Figure-3) (**NOTE:** 2 groups with 3 transmitter each, 2/3 transmitters from 2/2 groups generates the HI-HI signal.)

There is one narrow range pressure transmitter that is used for RTGB indication and alarm, PI-950B. There are two Wide Range Accident channels that are used for indication, PI-956 & 957 and are located on the Core Cooling Monitor Panels.

4.4 Local Instrumentation

There are local pressure indicators on the discharge of Spray Pumps “A” and “B”. There is also a local Spray Pump test line flow indicator.

4.5 Alarms

APP-002-D1 SPRAY ACTUATION and **APP-002-D2** CV ISOL PHASE B, Both will alarm at **20 psig** from PC-950, PC-951A, PC-952, PC-953A, PC-954, PC-955A. These alarms come in if 2/3 Hi-Hi Containment Pressure Bistable on both channels or if manual initiation has been actuated by depressing 2 pushbuttons simultaneously.

APP-002-E1 CV SPY PMP COOL WTR LO FLOW, Alarms at **30 gpm** from FIC-657. This alarm is caused by loss of component cooling to the pump (s).

APP-002-F1 CV SPY PMP MOTOR OVLD, Alarms when the 19A-74 relay is energized (Spray pump A) or when the 25C-74 relay is energized (Spray pump B). This alarm is caused by an overload on Spray Pump Motor.

APP-002-F2 SPRAY ADD TANK LO LEVEL, Alarm at **36%** from LC-949. ITS limit is 35.5% (2505 gallons). The tank should be filled to normal level.

5.0 CONTROLS AND PROTECTION

5.1 Containment Spray Actuation

5.1.1 Automatic

Containment Spray Actuation will automatically occur when a Containment Hi-Hi Pressure signal is sensed at 20 psig. This will cause the following:

NOTE: In the year 2000, it is planned to reduce this setpoint to 10 psig to allow the Service Water temperature to be increased without challenging CV pressure. (ESR 99-00153).

1) Steam Line Isolation actuation (closes all three MSIVs)

2) Spray actuation

3) Safety Injection actuation

NOTE: Containment pressure bistables for spray actuation are energize-to-actuate. This differs from other ESF actuations. The purpose is to minimize the possibility for an inadvertent spray signal due to power interruption.

4) Phase "B" Containment Isolation, The following valves close:
CC-716A & B, RCP Clg Wtr Inlet Isols
FCV-626, RCP Thermal Barrier Flow Control
CC-735, RCP Thermal Barrier Outlet Isol
CC-381, RCP Seal Wtr Rtrn Isol
CVC-730, RCP Oil Coolers Outlet Isol

5.1.2 Manual

Containment Spray Actuation can be manually actuated when both Spray pushbuttons are simultaneously depressed. There are Containment Spray Defeat pushbuttons on the RTGB that are not used (abandoned in place). Spray actuation will cause the following:

1) Spray actuation
2) Containment Phase "B"
3) Containment Ventilation Isolation - The following valves will close:
   - Purge Valves
   - Pressure Relief Valves
   - Vacuum Relief Valves

Question: 42

Given the following conditions:

- The unit is operating at 100% power.
- Normal letdown is in service.
- Pressurizer level control is in automatic
- Leakage passed the hydrogen pressure regulator to the VCT causes pressure in VCT to increase.

Which ONE (1) of the following describes the effect of this on RCP seal flow?

	No. 1 SEAL LEAKOFF FLOW	**No. 2 SEAL LEAKOFF FLOW**
a.	Increases	Increases
b.	Decreases	Decreases
c.	Decreases	Increases
d.	Increases	Decreases

Answer:

c.	Decreases	Increases

QUESTION NUMBER: 42

TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1

K/A: 003A2.05

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) use procedures to correct, control, or mitigate the consequences: Effects of VCT pressure on RCP seal leakoff flows

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 2.8

10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: CVCS-14

EXPLAIN the effect on the CVCS due to selected failures.

REFERENCES: SD-001
APP-003

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number CVCS-14 010

JUSTIFICATION:

a. Plausible since a change in VCT pressure will affect the RCP seal leakoff flows, but #1 seal leakoff flow will decrease.

b. Plausible since a change in VCT pressure will affect the RCP seal leakoff flows, but #2 seal leakoff flow will increase.

c. **CORRECT** Raising VCT pressure causes pressure against the #1 seal flow to increase, increasing pressure between the #1 and 2 seals and d/p across the #2 seal, causing #2 seal flow to increase. #1 seal flow decreases slightly due to more pressure in the VCT.

d. Plausible since a change in VCT pressure will affect the RCP seal leakoff flows, but #1 seal leakoff flow will decrease and #2 seal leakoff flow will increase.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the relationship between VCT pressure and RCP seal flows

REFERENCES SUPPLIED:

through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is the labyrinth seal which contains the thermal barrier heat exchanger. The thermal barrier limits heat transfer between hot system water and seal injection water. The thermal barrier heat exchanger provides sufficient cooling of the pump bearing if seal injection is lost.

The thermal barrier heat exchanger, though functioning as a boundary between RCS and Component Cooling Water (CCW), is only designed for a high differential pressure from O.D. to I.D. (CCW side). The pressure inside the heat exchanger should not exceed 200 psi. If a heat exchanger tube leak were to occur, the CCW isolation valves are designed to withstand full RCS pressure.

3.1.2 Seals (RCS-Figure-6)

Type	Westinghouse controlled leakage seal assembly
Seal water injection	8 gpm
Seal water return	3 gpm

The shaft seal section consists of three devices. They are the No. 1 controlled leakage, film-riding face seal, and the No. 2 and 3 rubbing face seals.

During normal system operation the charging pump(s) provide approximately 8 gpm injection flow to each RCP. The injection enters the pump between the thermal barrier and the pump bearing. The flow is then divided with approximately 5 gpm flowing down past the thermal barrier into the RCS and approximately 3 gpm flowing up past the pump bearing. The outlet from the No. 1 seal discharges to the Volume Control Tank (VCT). The VCT maintains a back pressure of at least 15 psig to ensure a flow through the No. 2 seal. The No. 2 seal discharges approximately 3 gph to the associated RCP standpipe. The standpipe overflows at midplane to the reactor coolant drain tank. The standpipe is located so that it maintains at least a seven-foot head to ensure flow through the No. 3 seal. The No. 3 seal discharges approximately 10cc - 100cc/hr to the containment sump.

When starting a RCP, RCS pressure low limit must be greater than 325 psig. With the pump operating, RCS pressure is allowed to decrease to a minimum of 210 psid on the seals before the pump must be secured. The 210 psid limitation is to ensure the #1 RCP seal has proper separation between surfaces. Additional information may be located in GP-001, "Fill and Vent of the Reactor Coolant System".

During heatup and cooldown, when the system water pressure is 1000 psig or below, leakoff flow may be insufficient to cool the bearing and seal components. When leakoff flow is below 1 gpm, the seal bypass valve should be opened. This permits a limited flow to bypass the No. 1 seal through a nonadjustable orifice block (external to the pump

ALARM

VCT HI/LO PRESS *** WILL REFLASH ***

AUTOMATIC ACTIONS

1. Not Applicable

CAUSE

1. Failure of N_2 or H_2 Regulator
2. Abnormal high level in VCT
3. Failure of CVC-258, VCT VENT

OBSERVATIONS

1. Volume Control Tank Pressure (PI-117)

ACTIONS

NOTE:	Minimum VCT pressure for RCP operation is 15 psig.

1. **IF** VCT pressure is high, **THEN** open CVC-258, VCT VENT.
2. **IF** VCT pressure is low, **THEN** verify closed CVC-258, VCT VENT.
3. Verify proper operation of N_2 and H_2 regulators.

DEVICE/SETPOINTS

1. PC-117 / 65 psig
2. PC-117 / 15 psig

POSSIBLE PLANT EFFECTS

1. Decreased number 1 seal leakoff (high)
2. Decreased number 2 seal leakoff (low)

REFERENCES

1. CWD B-190628, Sheet 473, Cables H, J

CVCS-14 010

Given the following plant conditions:

- The unit is at 100% power
- One charging pump with 45 gpm of letdown is in service
- VCT level is at 20 inches
- All make-up controls are in their normal configuration
- An Automatic makeup is in progress, 70 gpm primary water and 3 gpm boric acid
- LT-115 fails HIGH

Which ONE (1) of the following explain the effect of this failure on VCT level with no operator action?

A. VCT level will continue to cycle between 20.2"- 24.5" due to make-up system operation

B. VCT level will continue to increase at a slow rate to 24.5" and stabilize

✓C. VCT level will decrease to 0" due to lack of make-up capability

D. VCT level will decrease to 0, make-up continues

Question: 43

Given the following conditions:

- A reactor trip occurred from 20% power as a result of a low-low level in 'A' SG.
- Coincident with the reactor trip, 480V Bus E-1 deenergized and was subsequently energized by the EDG.
- Twenty (20) seconds following the trip, SG levels are:

SG	LEVEL
'A'	12%
'B'	28%
'C'	26%

Which ONE (1) of the following describes the expected condition of the Auxiliary Feed Water pumps 20 seconds following the trip?

	MDAFW PUMP 'A'	MDAFW PUMP 'B'	SDAFW PUMP
a.	Running	Running	Off
b.	Off	Running	Running
c.	Off	Running	Off
d.	Off	Off	Running

Answer:

c.	Off	Running	Off

QUESTION NUMBER: 43
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 013A4.03

Ability to manually operate and/or monitor in the control room: ESFAS initiation

K/A IMPORTANCE: ***RO*** 4.5 ***SRO*** 4.7
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: AFW-10

EXPLAIN the operation of the AFW System.

REFERENCES: SD-042
APP-004

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since this would be the expected condition if the EDG were not carrying the bus, but the auto start on low-low level is blocked for 'A' pump.

b. Plausible since this is the expected condition of the MDAFW pumps, but the SDAFW pump requires 2/3 low-low levels or a loss of both E-1 and E-2 to start.

c. **CORRECT** Both MDAFW pumps would normally start on low-low level, but the 'A' pump low-low level start is blocked and it will start at 39.5 seconds by the sequencer. The SDAFW pump requires 2/3 low-low levels to start.

d. Plausible since the EDG carrying the bus blocks the auto start on low-low level, but only the affected MDAFW pump.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Analysis of effect of loss of power on automatic operation of AFW pumps

REFERENCES SUPPLIED:

The auxiliary oil pump runs constantly to insure adequate lubrication to the turbopump.

A knurled knob on the speed governor can be used to control the speed of the turbine at a setpoint less than full speed. For normal automatic operation, the knob is set at the highest point so the turbine will operate at the maximum set speed during emergency conditions.

The SDAFW Pump may be manually tripped by pushing the "RED" trip button (AFW Figure 8). Reset the trip by pushing in the trip lever elbow.

The SDAFW pump will trip on overspeed and also trips on low discharge pressure to protect the pump from a loss of suction supply.

- 650 psig setpoint, 2/2 coincidence
- Shuts steam inlet valves (V1-8A, B and C)
- Note overspeed trip shuts V1-8A, B C due to 650 psig discharge pressure

The signals that will automatically start this pump are covered in Section 6.1 below.

6.0 SYSTEM OPERATION

6.1 Normal Operation

During normal plant operation the AFW system is not in service, except to augment startup, shutdown and cooldown. The Main Feedwater System is used whenever possible. The system must be operable under normal operations and periodic testing is performed to assure its operability.

This system will start automatically as follows: (See AFW Figure 11, AFW Pump Auto-start Logic).

If 2 of the 3 level detectors on any one of the S/Gs indicate low-low level, both MDAFW pumps will start. AMSAC will also cause an automatic start of the MDAFW Pumps. If the breakers of both main FW pumps open the MDAFW pumps will start. Blackout and safeguard conditions (SI) signal the MDAFW pumps to start on a timed sequence. The three motor operated discharge valves (AFW-V2-16A, 16B and 16C) will automatically open. The SGBD isolation valves (FCV-1930A and B, FCV-1931A and B, and FCV-1932A and B) will automatically close when either one of the MDAFW pumps automatically start. The common discharge valves at the MDAFW pumps discharge (V2-20A and B) can be used to have one pump feed two of the S/Gs in case of a break in the discharge line of the other pump, and these valves should normally be open.

The automatic AFW pump starts on S/G low-low level, AMSAC and both main FW pump breakers open can be blocked by key switch operations. The key switches are in the back of the RTGB.

Both the low-low S/G, AMSAC and main FW pump breakers automatic starts are blocked when the respective emergency buss is de-energized (loss of power) or when the respective EDG output breaker is shut. In this situation the MDAFW pumps will be started by the safeguards and blackout sequence logic.

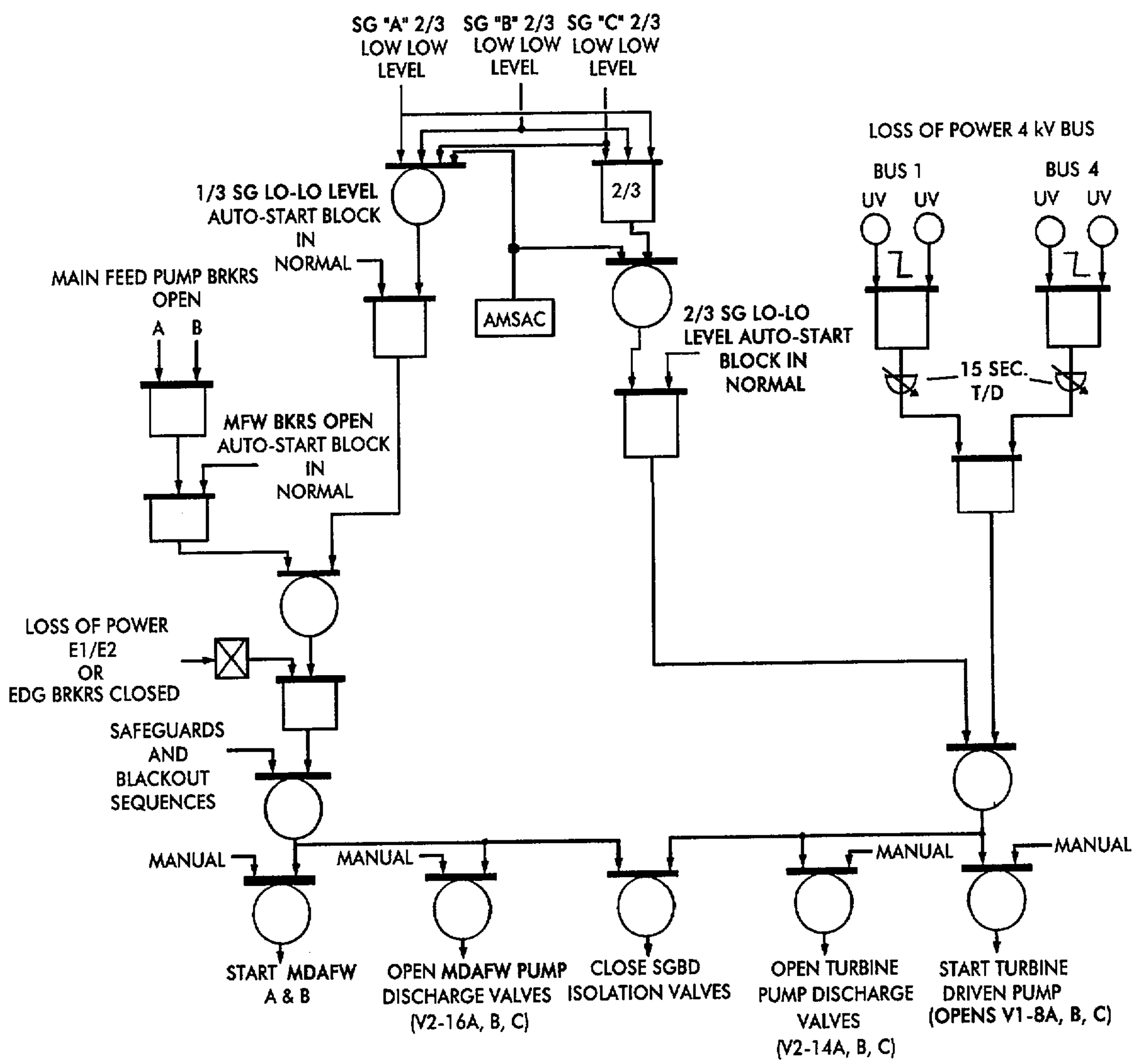
The SDAFW pump will automatically start if 2 of 3 level detectors on 2 of 3 S/Gs indicate low-low level. The SDAFW pump will also start from an under voltage on 4160V busses 1 and 4. AMSAC will cause the SDAFW pump to automatically start. A signal to start the SDAFW pump will open the three motor operated steam supply valves (MS-V1-8A, 8B and 8C), open the three motor operated discharge valves (AFW-V2-14A, 14B and 14C) and close the SGBD isolation valves (FCV-1930A and B, FCV-1931A and B, and FCV-1932A and B). The MOVs have individual control switches on the RTGB, so the operator may selectively feed any combination of S/Gs.

During normal plant operation, periodic testing will be performed to assure the AFW pumps ability to function when required. In addition, if for any reason the AFW Pumps are desired, they can be started and operated in the Manual Mode. The proper sequence to follow when securing an MDAFW Pump is, first, stop the pump, allow it to stop rotating, then close the motor operated discharge valves (V2-16A, V2-16B, V2-16C). This sequence will allow proper seating of the check valves and allow the discharge valves to fully seat which prevents back leakage through all these valves.

A possible consequence of check valve or discharge valve backleakage is steam binding of the AFW Pumps. Steam binding of the MDAFW Pumps may be indicated by warm discharge piping between the discharge check valve(s) and the V2-16(s). Steam binding of the SDAFW Pump may be indicated by a warm pump casing. If steam binding of any of the AFW Pumps is suspected, refer to the Infrequent Operation in OP-402.

The CST should be kept full. If the CST level decreases to 10% during AFW operation, a backup water supply should be placed in service. Service Water should be used as first backup supply to AFW Pumps. If Service Water is not available, Deepwell Water should be used as backup to AFW Pumps. Both isolation valves on the Service Water (SW-118 and AFW-24) and the deep well water backup (DW-19 and DW-21) will normally be locked closed with the telltale drain valves (AFW-24A and DW-20) open to prevent the plant condensate from being contaminated with untreated water. If the backup water supplies are required, the appropriate drain valves will be closed and the associated block valves opened. The flowrates from backup water could be limited IAW OP-402, see Attachment 10.2, "Backup Water Flow Limits".

AFW PUMP AUTO-START LOGIC
AFW-FIGURE-11 (REV. 1)



ALARM

S/G A LO-LO LVL TRIP

AUTOMATIC ACTIONS

1. Reactor Trip
2. Motor-Driven AFW Pumps start
3. Blowdown Isolation Valves close

CAUSE

1. Any sustained feedwater/steamflow mismatch.
2. S/G Shrink caused by sudden reduction in steam demand

OBSERVATIONS

1. Reactor trip breaker position
2. S/G "A" Level (LI-474, LI-475, LI-476)

ACTIONS

1. **IF** the Reactor has tripped, **THEN** refer to the EOP Network.
2. **IF** the Reactor is **NOT** tripped **AND** a plant transient is in progress, **THEN** trip the Reactor **AND** refer to the EOP Network.
3. **IF** the Reactor is **NOT** tripped **AND** the plant is stable, **THEN** perform the following:
    1) Scan the RTGB for confirmation that a trip is **NOT** required.
    2) Inform the CRSS **OR** SSO of plant conditions to assist in diagnosis.
    3) **IF** no supporting indications show a plant trip is required, **THEN** the plant may remain at power for troubleshooting and repairs.

DEVICE/SETPOINTS

1. LC-474A, LC-475A, LC-476A / 16% (2/3 Channels)

REFERENCES

1. EOP Network
2. 5379-2758, Logic Diagram
3. 5379-3440, Block Diagram
4. CWD B-190628 SH 440 Cable H

Question: 44

Given the following conditions:

- The plant is operating at 50% power.
- All control systems are operating in automatic.
- The First Stage Pressure Channel Selector switch is aligned to the PT-447 position.
- First Stage Pressure Transmitter PT-446 fails low.

Which ONE (1) of the following plant reponses is expected?

a. Feedwater Regulating Valves throttle closed

b. Control Rods step inward

c. Automatic rod control is blocked

d. Steam Dumps have a demand signal

Answer:

d. Steam Dumps have a demand signal

QUESTION NUMBER: 44
TIER/GROUP: *RO* 2/2 *SRO* 2/2
K/A: 035K4.01

Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the S/G level control

K/A IMPORTANCE: *RO* 3.6 *SRO* 3.8
10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: SG-08

EXPLAIN the component operation associated with each switch position for the Steam Generator System switches and controls.

REFERENCES: SD-033

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number MT-08 003

JUSTIFICATION:

a. Plausible since this would be the response if PT-446 were selected, but with PT-447 selected there is no response in feed water.

b. Plausible since this would be the response if PT-446 were selected, but with PT-447 selected there is no response in rod control.

c. Plausible since this would be the response if PT-446 were selected, but with PT-447 selected there is no response in rod control.

d. **CORRECT** The Tref signal for steam dumps is provided only by PT-446 (not selectable). With a low failure, Tavg would be higher than Tref, creating a steam dump demand. Dumps remain closed unless armed.

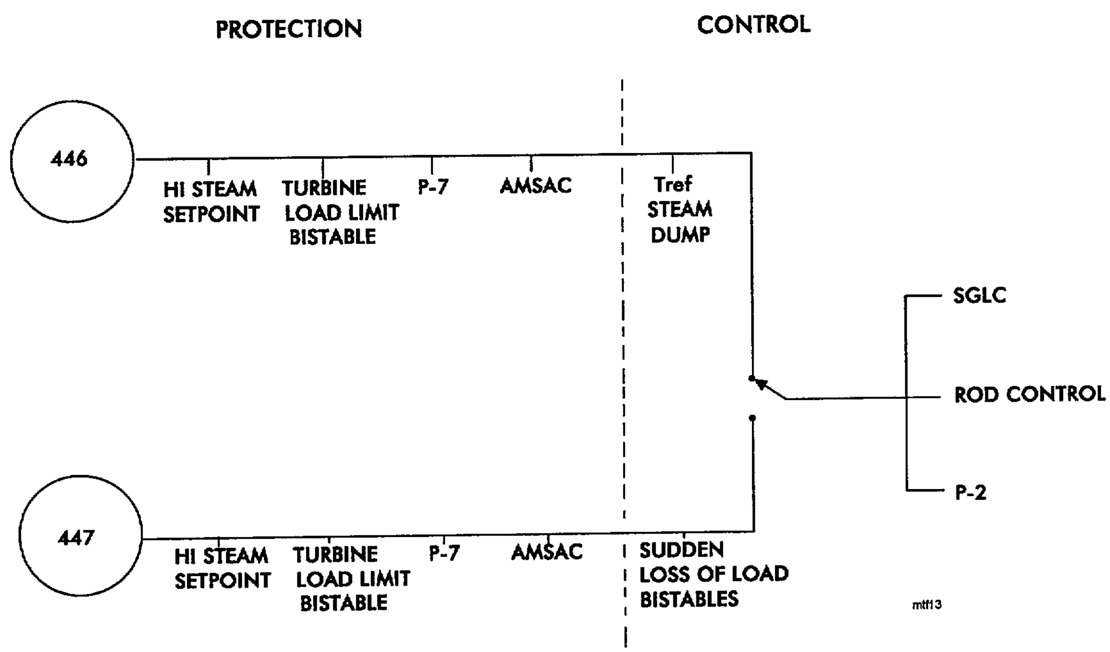
DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of instrument alignment to determine effect of first stage pressure failure

REFERENCES SUPPLIED:

FIRST STAGE PRESSURE

MT-FIGURE-13 (Rev . 0)



MT-08 003

Which ONE (1) of the following choices is supplied by the selector switch from either turbine first stage pressure channel PT-446 or 447?

A. Permissive P-7

B. Turbine load 70% bistables

C. Steam Dump Control System

✓D. Steam Generator Level Control

Question: 45

Given the following conditions:

- Due to low heat loads and extremely cold outside temperatures, Spent Fuel Pool (SFP) water temperature is 65°F.
- CC-775, CC FROM SPENT FUEL PIT HX BUTTERFLY Valve, has been throttled to the maximum allowed closed position.

Which ONE (1) of the following actions should be taken to raise Spent Fuel Pool water temperature?

a. Place the SFP on recirc to the RWST

b. Throttle the discharge valve of the in-service SFP Cooling pump

c. Shutdown the in-service SFP Cooling pump

d. Start an additional SFP Cooling pump

Answer:

c. Shutdown the in-service SFP Cooling pump

QUESTION NUMBER: 45

TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2

K/A: 033K3.03

Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Spent fuel temperature

K/A IMPORTANCE: ***RO*** 3.0 ***SRO*** 3.3

10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: SFP-09

EXPLAIN the normal operation of the spent fuel pit and purification system control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: OP-910

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number SFP-09 003

JUSTIFICATION:

a. Plausible since many systems heat up on recirc, but flow would continue through the SFP HX resulting in a lowering temperature.

b. Plausible since this would create a flow resistance and cause the water to heat up, but this would be offset by the increased heat removal from the SFP HX.

c. **CORRECT** The normal method of control is using CC-775, but if throttled to max position the SFP pump must be stopped to stop flow through the HX.

d. Plausible since this would provide additional pump heat, but this would be offset by the increased heat removal from the SFP HX.

DIFFICULTY:

Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of procedural requirements to adjust SFP temperature

REFERENCES SUPPLIED:

8.4.3 Adjusting Spent Fuel Pit Temperature (ACR 92-420)

1. Initial Conditions

   a. Spent Fuel Pit Cooling is in operation in accordance with Section 8.1.1 of this procedure.

2. Raising SFP Temperature

   a. Throttle closed CC-775, CC FROM SPENT FUEL PIT HX BUTTERFLY, to raise SFP temperature to between 74 °F and 121 °F.

   b. **IF** CC-775 has been throttled to the maximum allowable closed position **AND** the SFP Temperature continues to decrease **THEN** Go To Step 8.4.3.4.

3. Lowering SFP Temperature

   a. Throttle open CC-775, CC FROM SPENT FUEL PIT HX BUTTERFLY, to lower SFP temperature to between 74 °F and 121 °F.

4. Controlling SFP Temperature Under Low Heat Load Conditions

   a. Shutdown the Spent Fuel Pit Cooling Loop by stopping the running SFPC Pump.

Question: 46

Given the following conditions:

- The plant is operating at 68% power.
- Power Range channel N-43 is out of service for repairs.
- N-43 has been removed from service in accordance with the OWP.
- While working on N-43, the technician causes the Control Power fuses to blow.

Which ONE (1) of the following describes the effect of this on the plant?

a. **NO** effect since the OWP places the DROPPED ROD MODE switch in the "Bypass" position

b. **NO** effect since the Dropped Rod Runback requires two-of-four (2/4) coincidence to actuate

c. The turbine will runback for 9 seconds at 200% per minute

d. The turbine will runback at a cyclic rate of 200% per minute until power is $\leq$ 70%

Answer:

c. The turbine will runback for 9 seconds at 200% per minute

QUESTION NUMBER: 46
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 015K6.04

Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Bistables and logic circuits

K/A IMPORTANCE: ***RO*** 3.1 ***SRO*** 3.2
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: NIS-14

EXPLAIN the effect on the Nuclear Instrumentation System due to selected failures.

REFERENCES: SD-010

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible since the switch is placed in bypass, but control power is required to maintain the bypass condition.

b. Plausible since all other PR NIS actuations require a 2/4 coincidence, but the dropped rod runback / rod stop is 1/4.

c. **CORRECT** Even though the switch is in bypass, control power is required to maintain the bypass condition. The runback lasts for 9 seconds and will not recur until the signal is reset.

d. Plausible since a runback will occur, but the runback will be continuous for 9 seconds. The cyclic runback is caused by OT and OP ΔT signals.

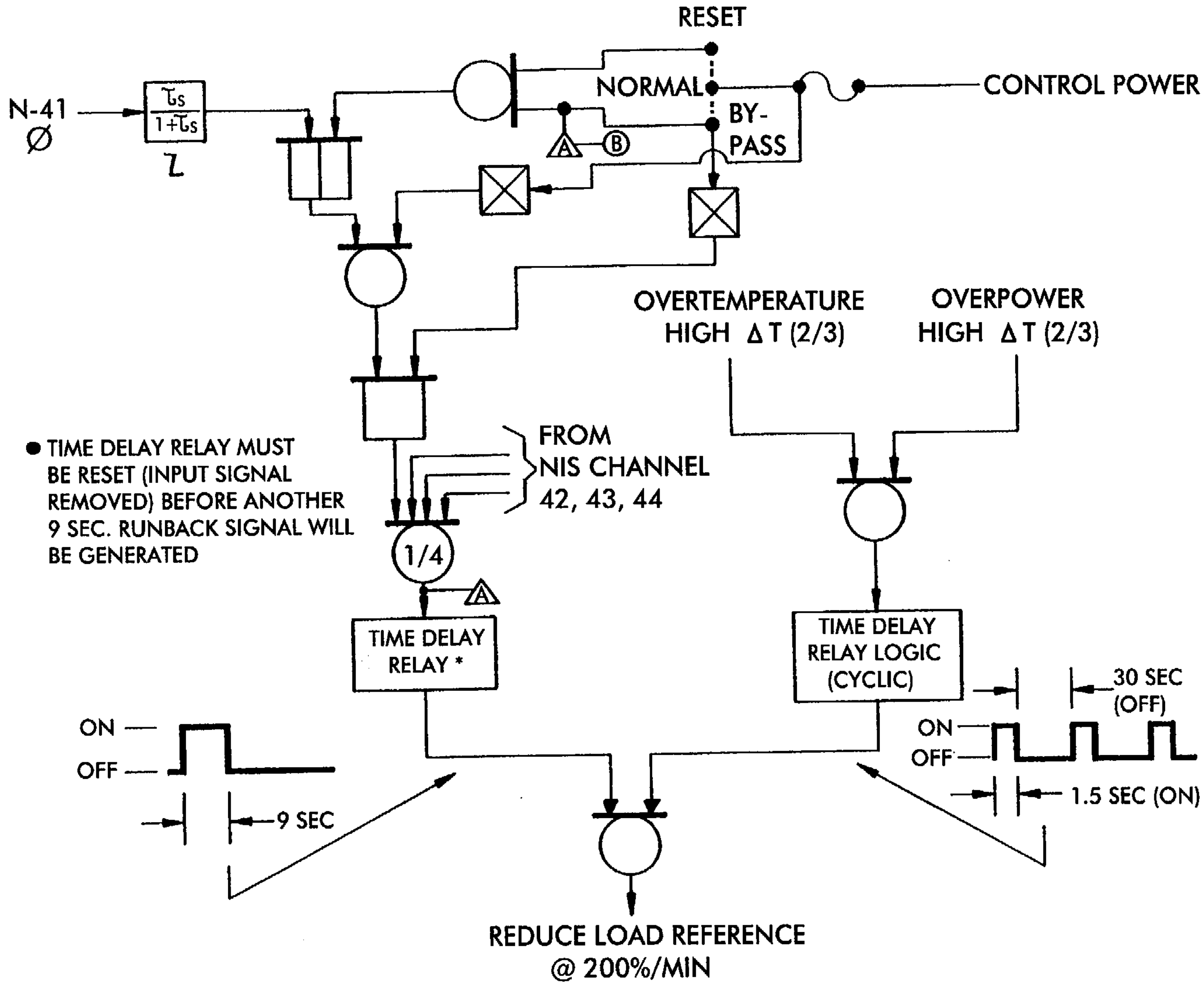
DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 4

Analysis of effect of failure on rod drop runback circuitry

REFERENCES SUPPLIED:

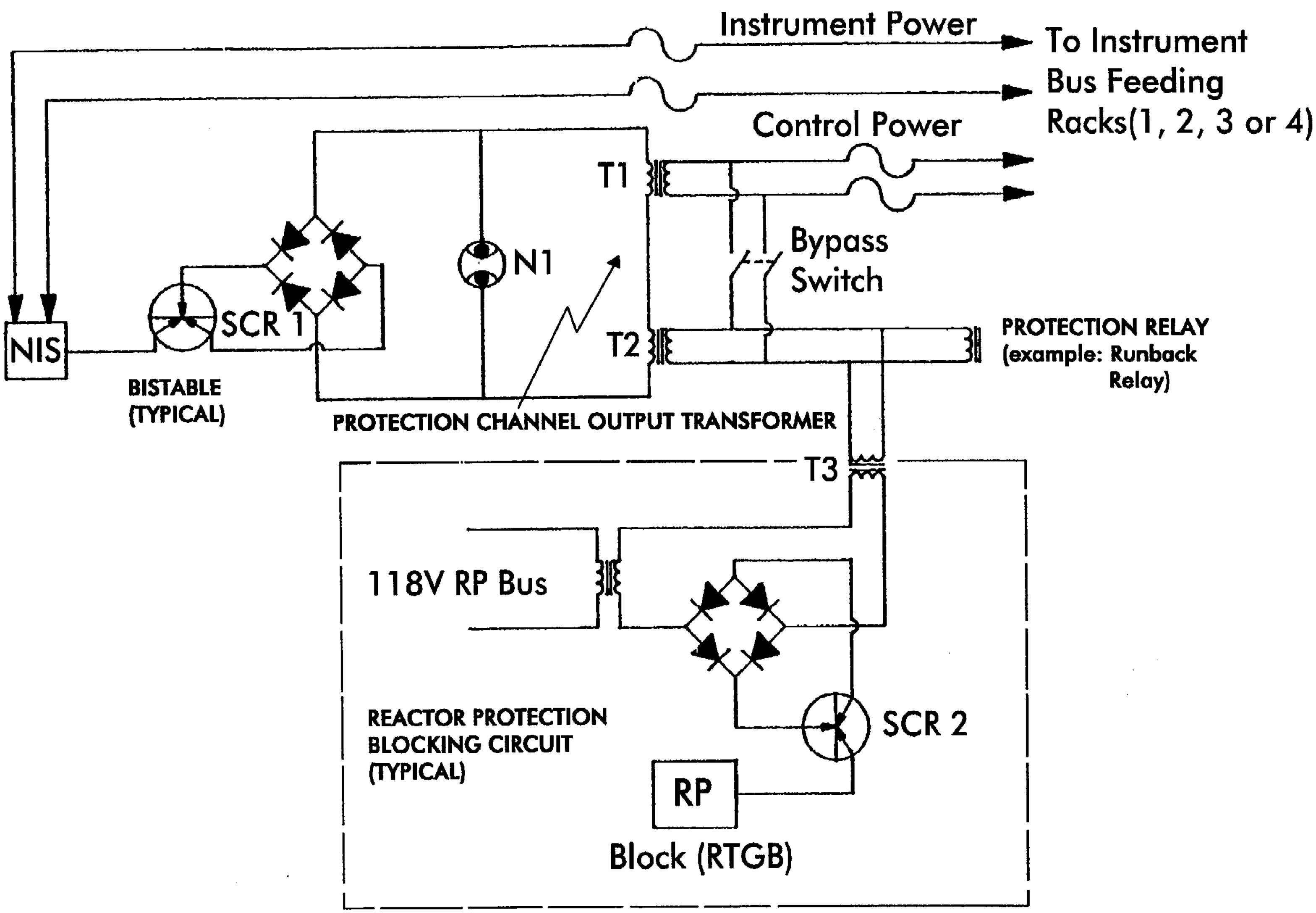
NIS RUNBACK LOGIC

NI-FIGURE-28 (Rev . 0)



NIS PROTECTION REPRESENTATION

NI-FIGURE-29 (Rev . 0)



runback relay reenergizes and stops the runback.

Question 3A:

With runback RESET, what will happen if the control power fuses are removed?

A runback will be initiated.

Question 3B:

Why does a runback occur when the control power fuses are removed?

The removal of the control power fuses causes the channel output transformers to deenergize which will initiate a runback.

Question 3C:

What will happen if the control power fuses are reinstalled within seconds?

The runback will stop.

Question 3D:

Why does the runback stop when the control power fuses are reinstalled?

The runback stops because the channel output transformers are reenergized when the fuse is installed.

Question 4A:

With NIS channel in **ROD DROP BYPASS**, what will happen if the control power fuses are pulled?

A runback will be initiated.

Question 4B:

Why does a runback occur if the control power fuses are pulled when the NIS channel is in rod drop bypass?

The removal of the control power fuses causes the channel output transformers to

deenergize which will initiate a runback and the runback will not be blocked because the removal of the control power fuses will cause the runback block relay to deenergize and the runback block will be removed.

Question 4C:

What will happen if the control power fuses are reinstalled within a few seconds while the runback is in progress?

The channel output transformers will reenergize and the runback will stop.

Question 4D:

Why does the runback stop when the control power fuses are reinstalled?

The runback stops because the channel output transformers are reenergized and the runback relay reenergizes and stops the runback.

Question 5A:

With a NIS channel in Rod Drop Bypass, what will happen if the instrument power fuses are removed?

The channel rod drop bistable will deenergize but no runback will occur.

Question 5B:

Why doesn't a Runback occur?

The channel rod drop bistable will deenergize and this should cause a runback to occur; however, with the Rod Drop Bypass switch in the BYPASS position, the runback will be blocked and will not occur.

Question 5C:

What will happen if the instrument power fuses are reinstalled within seconds?

The rod drop bistable will automatically reset and return to the energized state.

Question 5D:

Why does the rod drop bistable automatically reset when the fuses are reinstalled?

With the rod drop bypass switch in the bypass position, the bistable will reset

Question: 47

Given the following conditions:

- A LOCA has occurred inside containment.
- Due to electrical problems an entry was made to EPP-15, "Loss of Emergency Coolant Recirculation."
- One (1) Containment Spray pump was operating upon exiting EPP-15, with containment pressure at 16 psig.
- Subsequently, an entry was made to FRP-J.1, "Response to High Containment Pressure," due to containment pressure being at 14 psig and lowering slowly.

Which ONE (1) of the following describes the actions that are to be taken regarding the Containment Spray system?

a. Return to EPP-15 to determine Containment Spray system requirements

b. Stop the running Containment Spray pump

c. Maintain the current Containment Spray system configuration

d. Start the second Containment Spray pump

Answer:

c. Maintain the current Containment Spray system configuration

QUESTION NUMBER: 47
TIER/GROUP: *RO* 1/1 *SRO* 1/1
K/A: WE14EK1.2

Knowledge of the operational implications of the following concepts as they apply to the (High Containment Pressure) Normal, abnormal and emergency operating procedures associated with (High Containment Pressure).

K/A IMPORTANCE: *RO* 3.2 *SRO* 3.7
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: FRP-J.1-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to high containment pressure as directed in FRP-J.1.

REFERENCES: FRP-J.1

SOURCE: *New* ☒ *Significantly Modified* ☐ *Direct* ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible since EPP-15 has priority over FRP-J.1 for containment spray operation, but no conditions merit re-entry into EPP-15.

b. Plausible since containment pressure is lowering, but containment spray is maintained in operation until pressure is < 10 psig.

c. CORRECT Upon entry to FRP-J.1, if containment spray is being operated per EPP-15, no change should be made to the configuration.

d. Plausible since containment pressure is still above 10 psig, but EPP-15 has priority over FRP-J.1 for containment spray operation.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of priority of containment spray operations under abnormal conditions

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check CONTAINMENT ISOLATION PHASE A Valves - CLOSED	Perform the following: a. Momentarily depress either of the CONTAINMENT ISOLATION Pushbuttons. b. IF any CONTAINMENT ISOLATION PHASE A Valve fails to close, THEN locally isolate the affected penetration.
2.	Check CONTAINMENT VENTILATION ISOLATION Valves - CLOSED	Perform the following: a. Momentarily depress either of the CONTAINMENT ISOLATION Pushbuttons. b. IF any CONTAINMENT VENTILATION ISOLATION Valve fails to close, THEN locally isolate the affected penetration.
3.	Check CV Pressure - HAS INCREASED TO GREATER THAN 10 PSIG	Return to procedure and step in effect.
4.	Determine Availability Of CV Spray As Follows:	
	a. Check CV Spray - BEING CONTROLLED BY EPP-15, LOSS OF EMERGENCY COOLANT RECIRCULATION	a. Go To Step 5.
	b. Go To Step 6	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Establish CV Spray As Follows:	
	a. Verify OPEN CV Spray Pump Inlet Valves: • SI-844A • SI-844B	
	b. Verify both CV Spray Pumps - RUNNING	
	c. Verify OPEN the following Containment Spray Valves: • SI-845A, SAT DISCH • SI-845B, SAT DISCH • SI-880A, PUMP A DISCH • SI-880B, PUMP A DISCH • SI-880C, PUMP B DISCH • SI-880D, PUMP B DISCH	
	d. Check Spray Additive Tank flow - APPROXIMATELY 12 GPM	d. Adjust SI-845C, SAT THROTTLING to obtain approximately 12 gpm Spray Additive Tank flow.
6.	Verify CONTAINMENT ISOLATION PHASE B Valves - CLOSED	
7.	Verify All RCPs - STOPPED	
8.	Verify CV AIR RECIRC COOLERs - RUNNING • HVH-1 • HVH-2 • HVH-3 • HVH-4	

Question: 48

Given the following conditions:

- A recovery from a small break LOCA is in progress.
- **NO** RCPs are running.
- EPP-008, "Post-LOCA Cooldown and Depressurization," is being implemented.
- Depressurization of the RCS has commenced.
- Pressurizer level has just risen rapidly from off-scale low to 50%.

The depressurization of the RCS has ...

a. increased RHR and SI flow, which is rapidly refilling the pressurizer.

b. caused voiding to occur in the reactor vessel head, which is rapidly refilling the pressurizer.

c. increased pressurizer spray flow, which is rapidly refilling the pressurizer.

d. caused voiding in the pressurizer level reference leg, which is providing an indication of rapidly increasing pressurizer level.

Answer:

b. caused voiding to occur in the reactor vessel head, which is rapidly refilling the pressurizer.

QUESTION NUMBER: 48
TIER/GROUP: *RO* 1/2 *SRO* 1/2
K/A: WE03EA1.2

Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization) Operating behavior characteristics of the facility.

K/A IMPORTANCE:	***RO***	3.7	***SRO***	3.9
10CFR55 CONTENT:	***55.41(b) RO***	5	***55.43(b) SRO***	

OBJECTIVE: EPP-008-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in EPP-8 by explaining the basis of each.

REFERENCES: EPP-008

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number EPP-008-03 014

JUSTIFICATION:

a. Plausible since flow from centrifugal ECCS pumps increases as RCS pressure decreases and increased ECCS flow will fill the PZR, but level increases rapidly due to voiding in the head.

b. **CORRECT** The upper head region may void during RCS depressurization if RCPs are not running. This may result in a rapidly increasing PZR level.

c. Plausible since increased spray would cause RCS depressurization and inject more water into the PZR via the spray line, but level increases rapidly due to the voiding in the head.

d. Plausible since voiding in the reference leg would increase PZR level indication, but level increases rapidly due to voiding in the head.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the effects of a natural circulation cooldown on RCS head voiding

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
18.	Determine PZR Heater Status:	
	a. Check RCS Leak Location - KNOWN TO BE PZR	a. Go To Step 19.
	b. Place all PZR Heaters in OFF	
	c. Observe the CAUTION prior to Step 21 and Go To Step 21.	
*19.	Check PZR Level - GREATER THAN 71% [60%]	Place all PZR Heaters in OFF. IF PZR level increases above 71% [60%], THEN energize PZR heaters to maintain steam bubble. Observe CAUTION prior to Step 21 and Go To Step 21.
20.	Energize PZR Heaters To Maintain Steam Bubble	

****

CAUTION

The upper head region may void during RCS depressurization if RCPs are not running. This may result in a rapidly increasing PZR level.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*21.	Depressurize RCS To Refill PZR As Follows:	
	a. Check PZR level - LESS THAN 24% [45%]	a. Go To Step 22.
	b. Use normal PZR Spray to depressurize the RCS	b. Use one PZR PORV. IF no PZR PORV is available, THEN use Auxiliary Spray.
	c. Check PZR level - GREATER THAN 24% [45%]	c. WHEN PZR level greater than 24% [45%], THEN stop RCS depressurization. Go To Step 22.
	d. Stop RCS depressurization	

Question: 49

Given the following conditions:

- The unit is operating at 100% power.
- Rod Control is in Manual.
- A safety valve fails open on SG 'B'.

Which ONE (1) of the following describes the effect on indicated power and RCS Tavg?

	INDICATED NIS POWER	**RCS T-AVG**
a.	Increases	Remains Relatively Constant
b.	Increases	Decreases
c.	Remains Relatively Constant	Remains Relatively Constant
d.	Remains Relatively Constant	Decreases

Answer:

b.	Increases	Decreases

QUESTION NUMBER: 49
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 039K5.08

Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity

K/A IMPORTANCE: ***RO*** 3.6 ***SRO*** 3.6
10CFR55 CONTENT: ***55.41(b) RO*** 5 ***55.43(b) SRO***

OBJECTIVE: MSS-14

EXPLAIN the effect on the Main Steam System due to selected failures.

REFERENCES: Main Steam Lesson Plan

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number MSS-14 003

JUSTIFICATION:

a. Plausible since the expected response to a power increase (controlled) is to withdraw rods to maintain temperature, but no rod motion is given.

b. **CORRECT** The increased heat removal due to increased steam demand will cause the RCS to cool down. This will add negative reactivity which will cause an increase in power.

c. Plausible since the expected response to a power increase (controlled) is to withdraw rods to maintain temperature, but no rod motion is given.

d. Plausible since the increased heat removal due to increased steam demand will cause the RCS to cool down, but this will add negative reactivity which will cause an increase in power.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the effect of increased steam flow on RCS parameters

REFERENCES SUPPLIED:

MAIN STEAM

NORMAL OPERATION

All three S/Gs supplying 100% steam flow to the main turbine.

INFREQUENT OPERATION

Plant heatup

The system is heated up by using heat developed by the RCPs and the PZR heaters

Above atmospheric pressure the system can be vented/blown down to remove non-condensable and condensate

At various RCS temperatures below 543°F the MSIV before and after seat drain valves and S/G blowdown flow will be adjusted to maintain or increase temperatures of the RCS

At > 543°F the MSIV bypasses will be used to warm up the main steam system. At 50 psid across the MSIVs the valves may be opened and the MSIV bypasses closed.

Power/load increase

After the reactor is taken to power the steam dumps are used to control RCS temperature

OBJ. #14

10% load - MSR purge valves opened

35% load - MSR shutoff valves opened and pneumatic time pattern transmitter started to open the timer valves (LP turbine inlet steam temperature limit 100°F/hour)

ABNORMAL OPERATION

Accidental opening of a S/G PORV or safety valve while at power will cause an increase in steam flow for that loop which in turn could:

Cause excessive cooldown of the RCS

Cause an increase in reactor power

Lead to turbine runback, reactor trip, safeguards actuation (steam line Δp)

Accidental closing of an MSIV while at power will cause an increase in steam pressures for that steam line which in turn could:

Cause lifting of the PORV and/or safeties in that steam line

OBJ. #12, 13

Cause shrinkage in the S/G level in the failed loop and increased level (swell) in the non-failed loops

USE LATEST REVISION OF IMPROVED TECH SPECS TO DISCUSS LCO ACTION STATEMENTS, AND BASIS

Lead to reactor trip and safeguards actuation by high steam flow (high stm flow w/low press)

TECHNICAL SPECIFICATIONS

LIMITING CONDITIONS FOR OPERATION

I.T.S. 3.6.3, Containment Systems

OBJ. #16

Actions

Basis

I.T.S. 3.7.1 and 3.7.2, Plant Systems

LER-94-020

Actions

Basis

OPERATING EXPERIENCE

COMMITMENTS

NONE

PLANT SPECIFIC EVENTS (NON-

Question: 50

Given the following conditions:

- The unit is operating at 85% power.
- Control Rod Bank 'D' Demand is at 195 steps.
- IRPI indication for Bank D Control Rods are as follows:

ROD	POSITION
D-8	123"
M-8	121"
H-4	120"
H-8	110"
H-12	122"

Design power peaking and Shutdown Margin Limits ...

a. are met under these conditions.

b. will be met if Control Rod H-8 is withdrawn to 115".

c. will be met if power is reduced below 80%.

d. will be met if Control Rod D-8 is inserted to 120".

Answer:

b. will be met if Control Rod H-8 is withdrawn to 115".

QUESTION NUMBER: 50

TIER/GROUP: *RO* 1/1 *SRO* 1/1

K/A: 005AK3.03

Knowledge of the reasons for the following responses as they apply to the Inoperable / Stuck Control Rod: Tech-Spec limits for rod mismatch

K/A IMPORTANCE: *RO* 3.6 *SRO* 4.1

10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: RDCNT-12

State the Technical Specification Limitations and explain the bases for the Rod Control System.

REFERENCES: Tech Spec 3.1.4

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number RNP-RO-2000 07

JUSTIFICATION:

a. Plausible since rods would be considered aligned if bank position was above 200 steps (within 15 inches). With rods below 200 steps, requirement is within 7.5 inches.

b. **CORRECT** Below 200 steps, rods must be aligned within 7.5 inches of average IRPI indication for the rods in the bank. If rod H-8 is included in this calculation, the average rod height is 119.4". If rod H-8 is not included, the average rod height is 121.5".

c. Plausible since actions are taken to lower power if a misaligned rod cannot be aligned within a time period. Although rod is misaligned, required power level is 70%, not 80%.

d. Plausible since this rod is higher than the average of the rods. Lowering rod D-8 to 120 inches would lower the average rod height to 120.75" if rod H-8 is not included and 118.8" if rod H-8 is included. Both values would still leave rod H-8 misaligned.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Comprehension of rod alignment limits and determination of rod misalignment.

REFERENCES SUPPLIED:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be as follows:

a. For bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position, and

b. For bank demand positions < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank.

APPLICABILITY: MODES 1 and 2.

ACTIONS

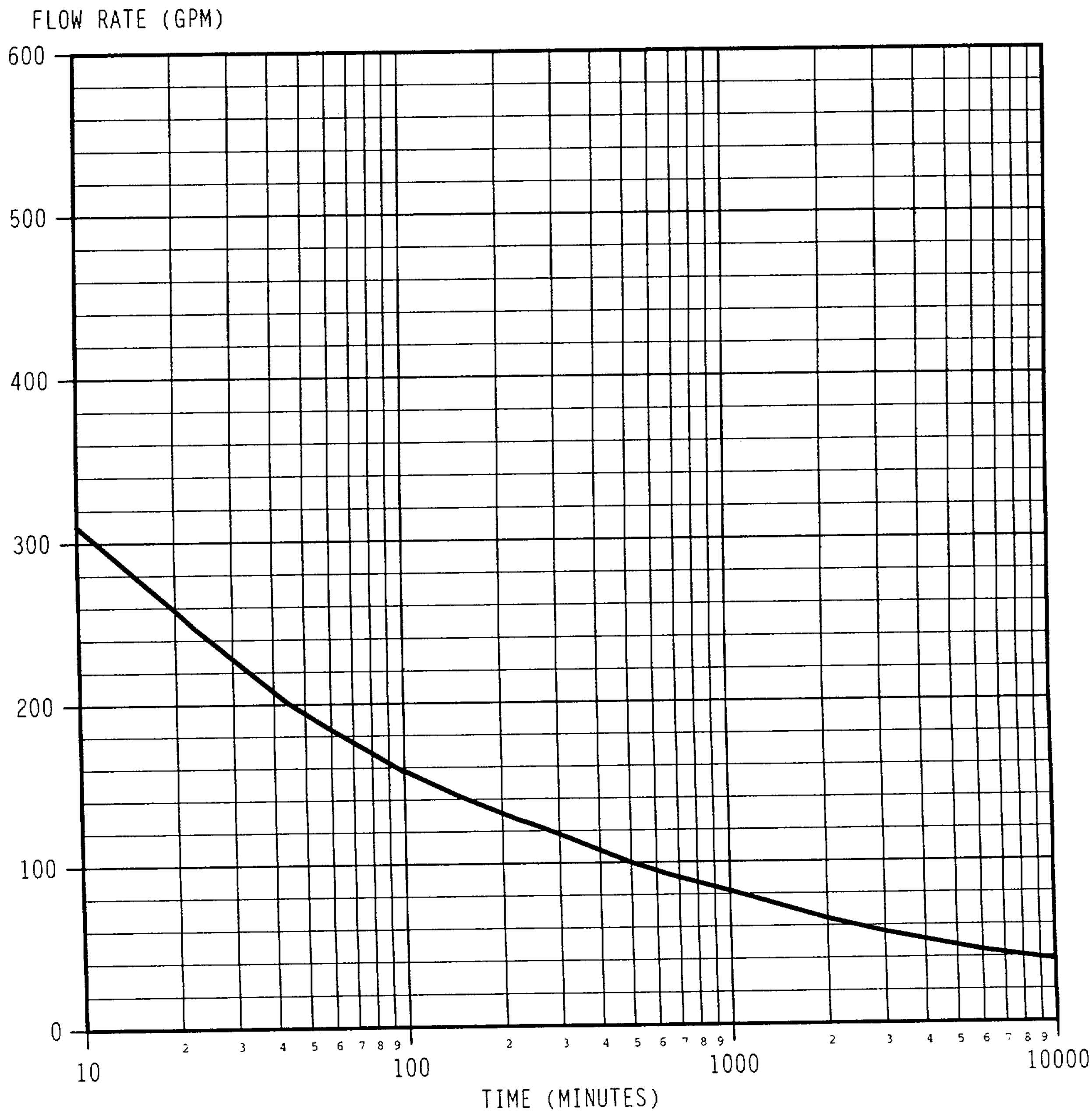
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

(continued)

SUPPLIED REFERENCE MATERIALS FOR RNP NRC REACTOR OPERATOR EXAMINATION

REFERENCE NUMBER	REFERENCE TITLE
NA	Steam Tables
EPP-15, Attachment 1	Required Flow Rate Versus Time After Reactor Trip
GP-005, Attachment 10.1	Reactor Power Ascension Indicator Log
Plant Curve 5.3	Boron Addition – Coolant Hot - Gallons
Plant Curve 5.4	Boron Addition – Coolant Cold - Gallons
Plant Curve 5.7	Dilution – Coolant Hot - Gallons
Plant Curve 5.8	Dilution – Coolant Cold - Gallons
Plant Curve 7.6	System Resistance Curve, Post Accident Containment Venting System
Plant Curve 7.16	Total Hydrogen Generation Rate from All Sources
Plant Curve 7.19	Loss of Residual Heat Removal Cooling Water Level Between 0" to –10" Below Vessel Flange
Plant Curve 7.20	Loss of Residual Heat Removal Cooling Water Level Between –10" to –36" Below Vessel Flange
Plant Curve 7.21	Loss of Residual Heat Removal Cooling Water Level Between –36" to –72" Below Vessel Flange

ATTACHMENT 1
REQUIRED FLOW RATE VERSUS TIME AFTER REACTOR TRIP
Page 1 of 1



REACTOR POWER ASCENSION INDICATOR LOG

AVG PWR % (1)	NI-35 amps	NI-36 amps	NI-41A %	NI-42A %	NI-43A %	NI-44A %	LOOP ΔT °F (1)	LOOP 1 ΔT °F	LOOP 2 ΔT °F	LOOP 3 ΔT °F	1st STAGE PRESS psig (1)	PI-446 **OR** 447 psig (2)	NET MWe MAX (1)	NET MWe	CCP % PWR (3)	NR-45 (4)	SSO (1)
15-20							9-11.5				68-90		73				
25-30							14.5-17				113-135		153				
35-40							20-23				158-180		235				
45-50							26-28.5				207-230		316				
55-60							32-34.5				261-285		398				
65-70							37-40				320-345		480				
75-80							43-46				384-410		562				
85-90							49-51.5				449-475		643				
95-100							55-57.5				513-540		725				

(1) Listed ranges and Net MWe maximums are predicted based on past plant performance. The maximum value of each indication is the maximum target value for each power increase. The SSO shall initial if plant management has determined that indications are acceptable to continue with the power escalation.

(2) Use indicator that corresponds to the channel selected on the 1st STAGE PRESSURE selector switch.

(3) Record Continuous Calorimetric Program % Power.

(4) Verify NR-45 is selected to the highest reading channel.

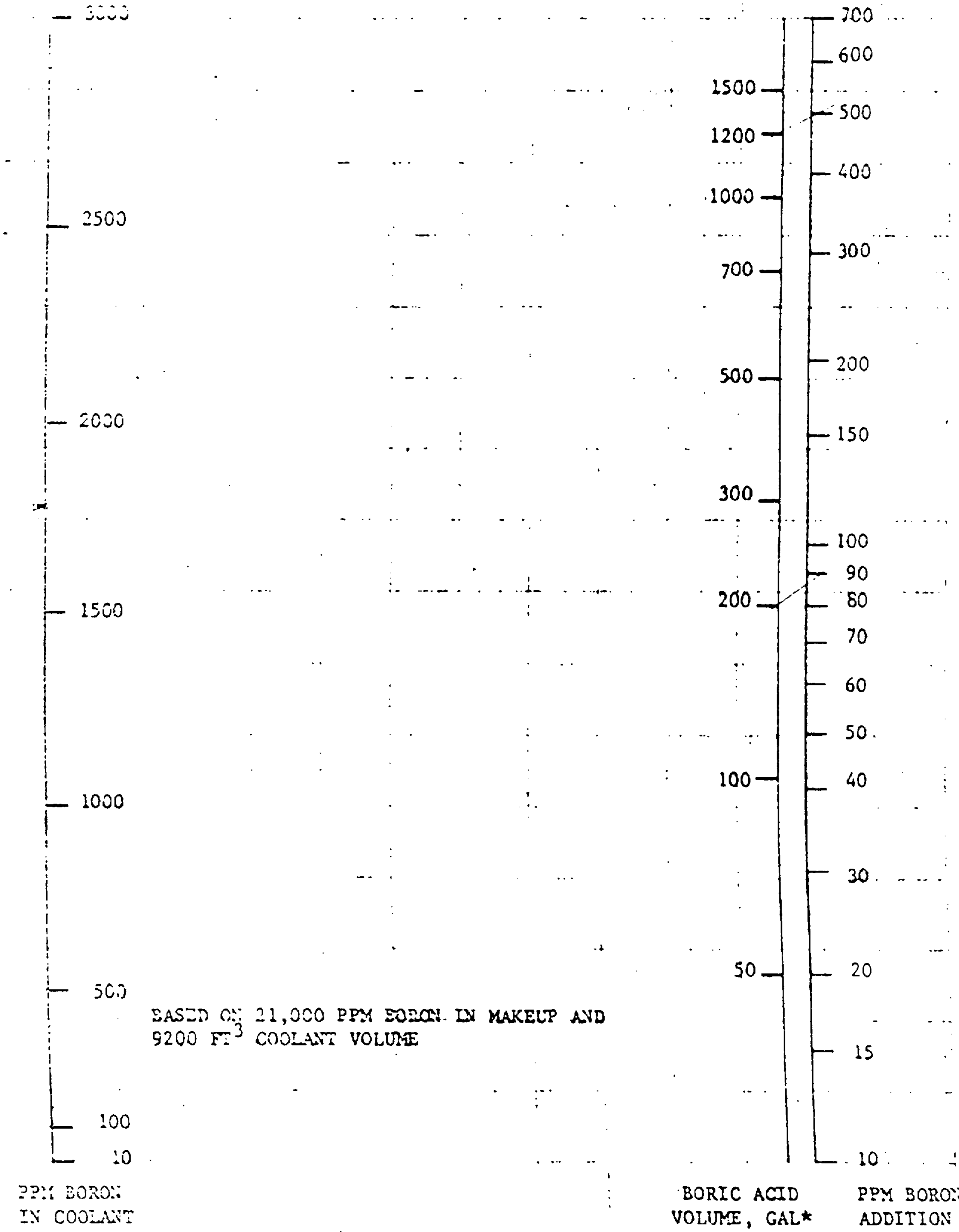


FIGURE S-3.1-3 BORON ADDITION - COOLANT HOT (~580°F)

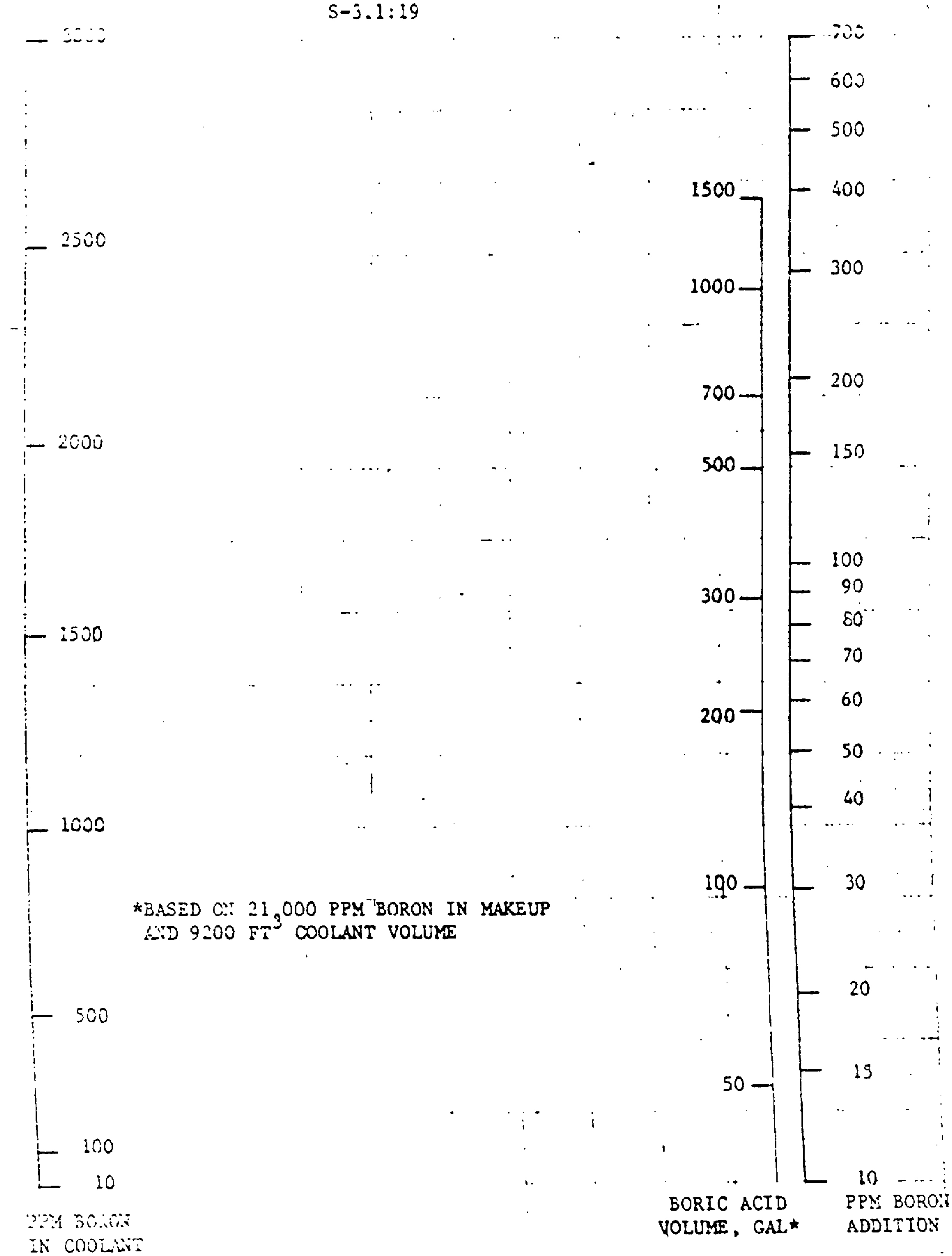


FIGURE S-3.1-4 BORON ADDITION - COOLANT COLD (-100°F)

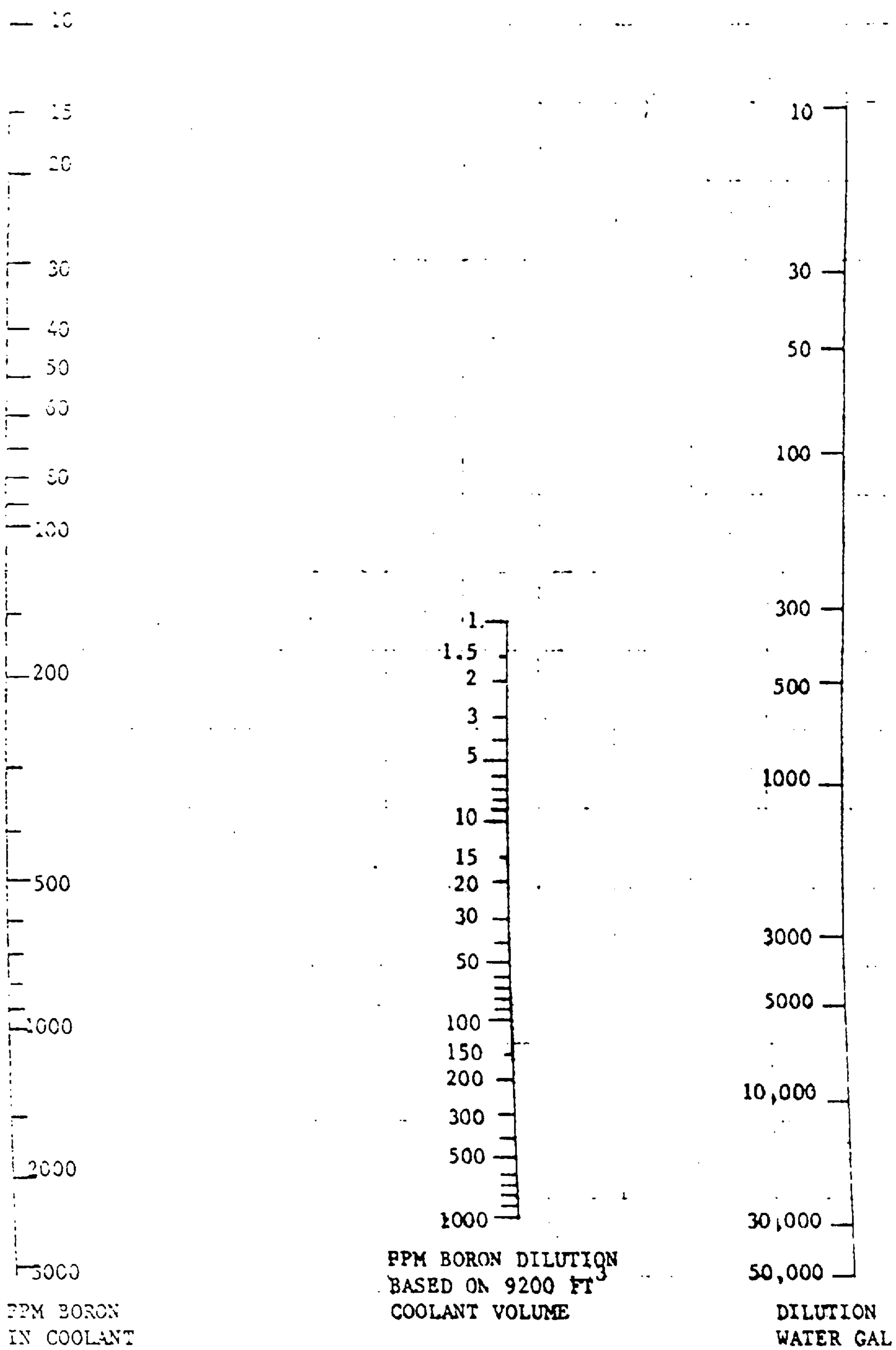


FIGURE S-3.1-7 DILUTION NOMOGRAPH - COOLANT HOT (-580°F)

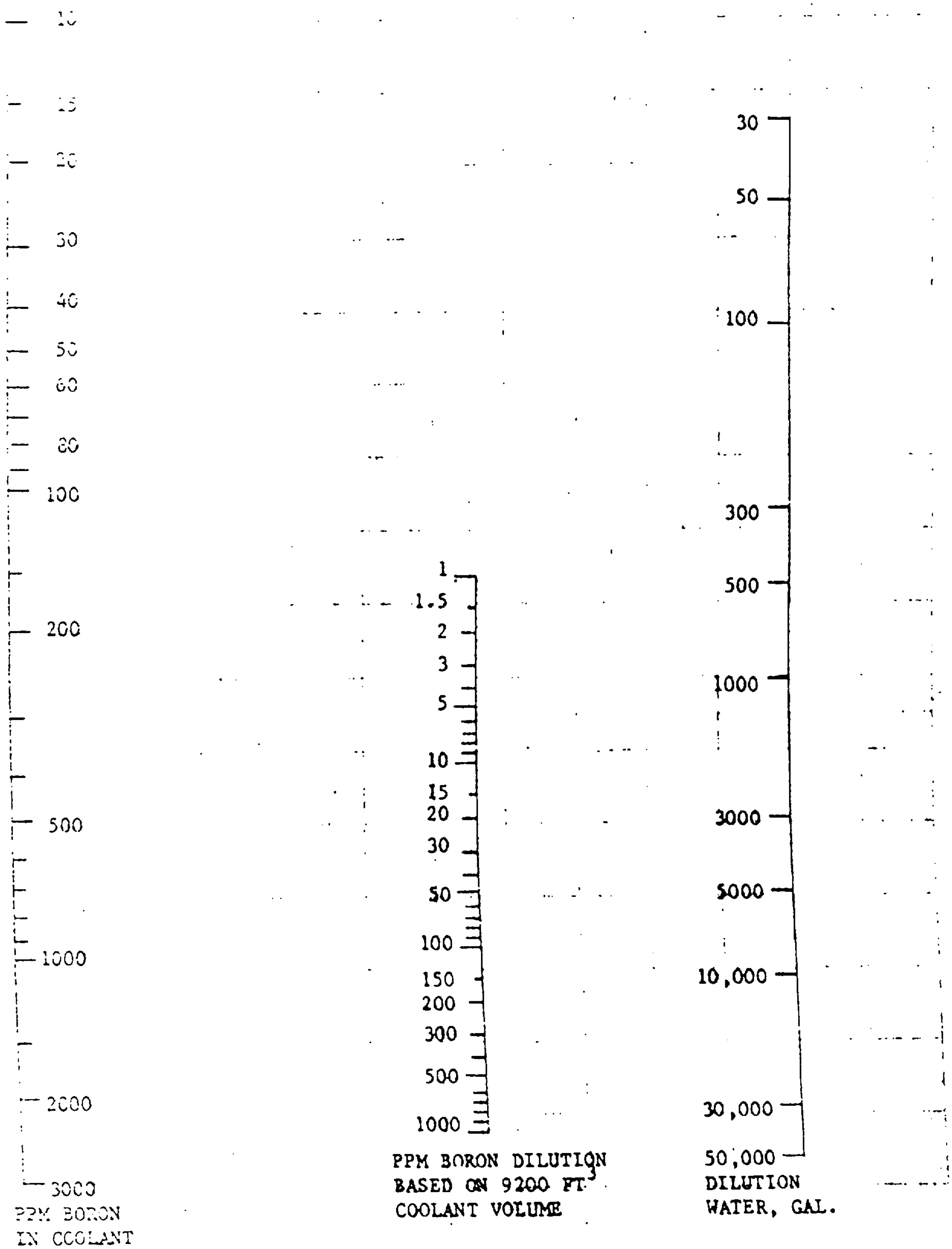


FIGURE S-3.1-5 DILUTION NOMOGRAPH - COOLANT COLD (-100°F)

46 1240

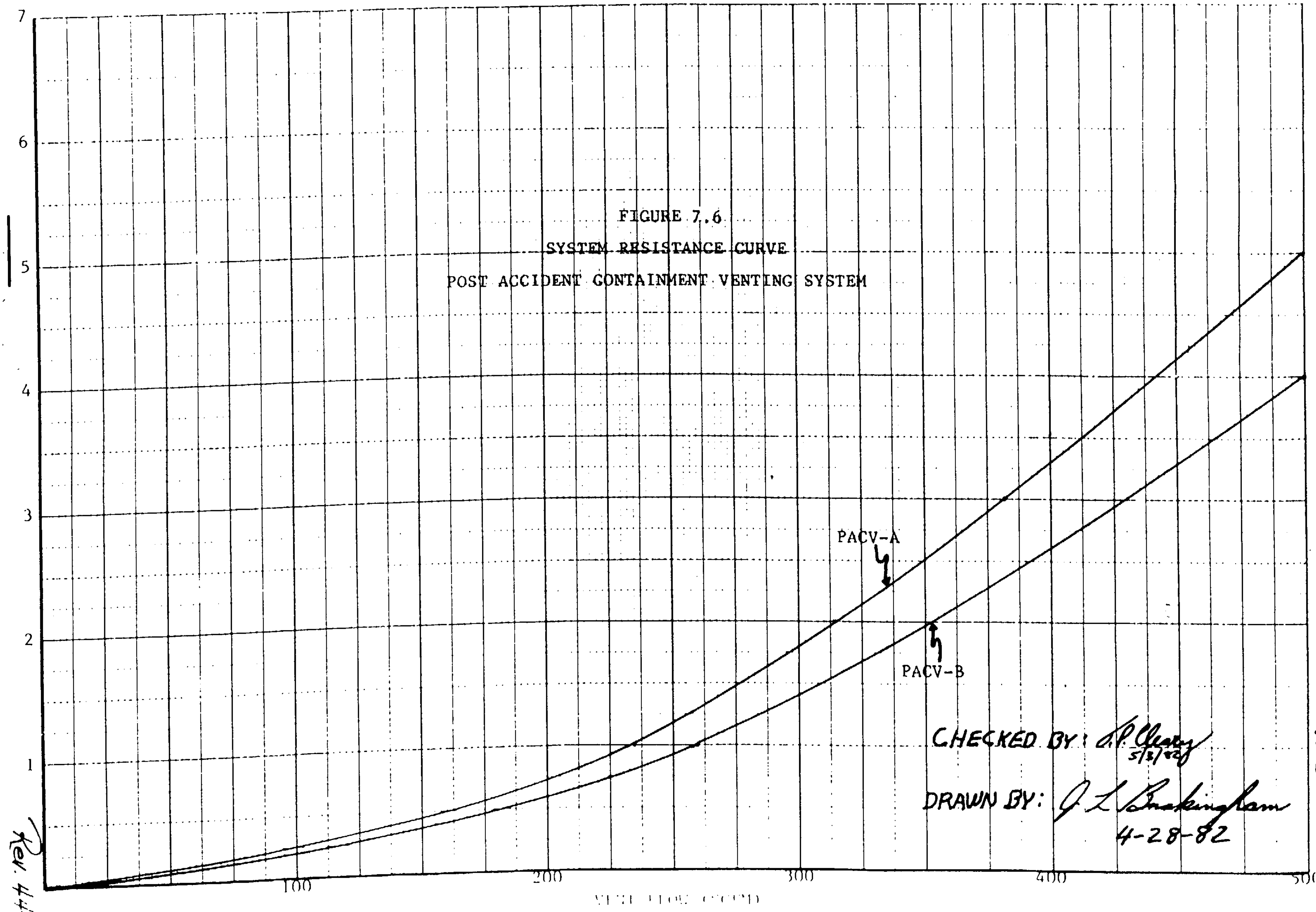
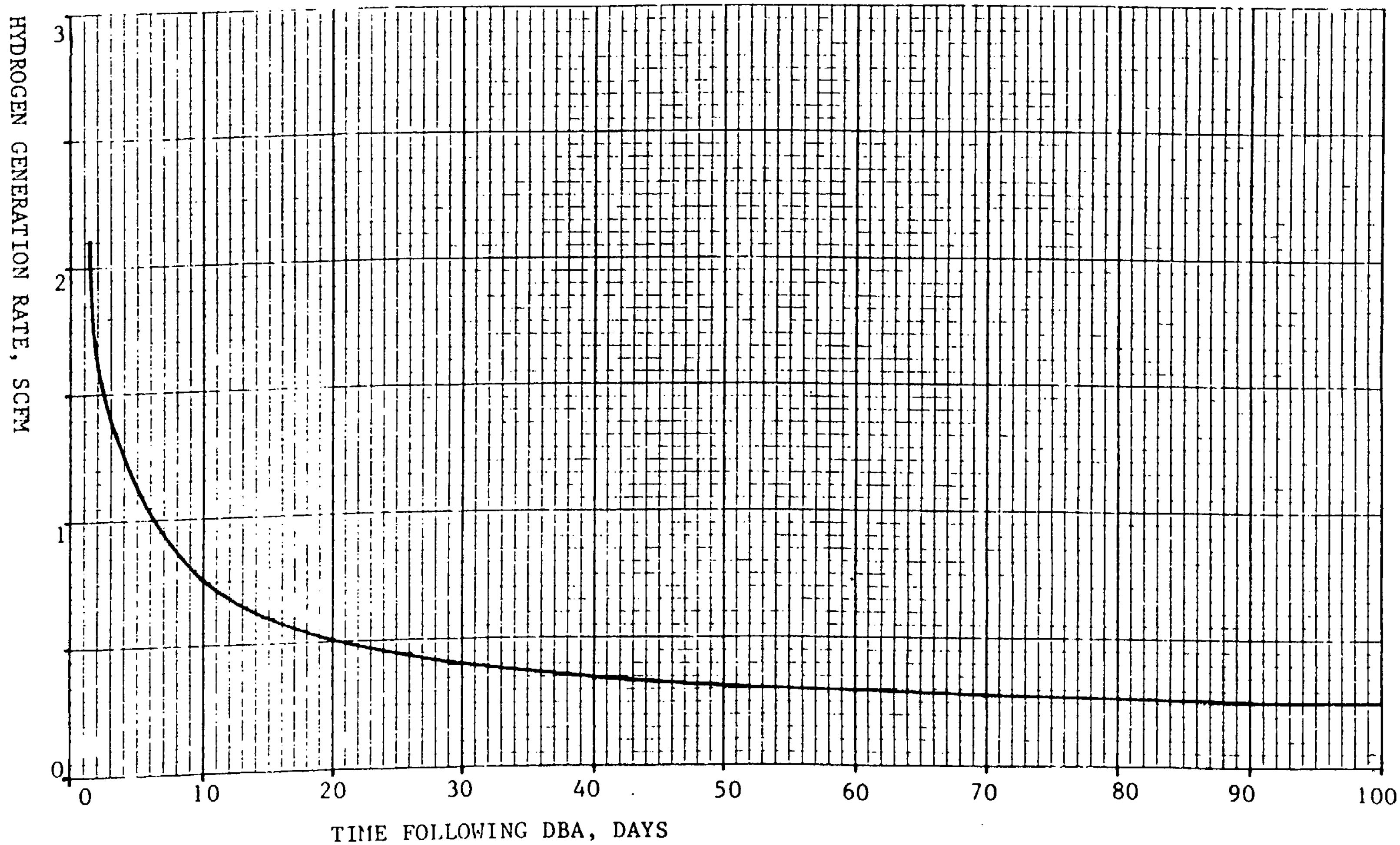


FIGURE 7.6

SYSTEM RESISTANCE CURVE

POST ACCIDENT CONTAINMENT VENTING SYSTEM

CURVE 7.16 TOTAL HYDROGEN GENERATION RATE FROM ALL SOURCES.



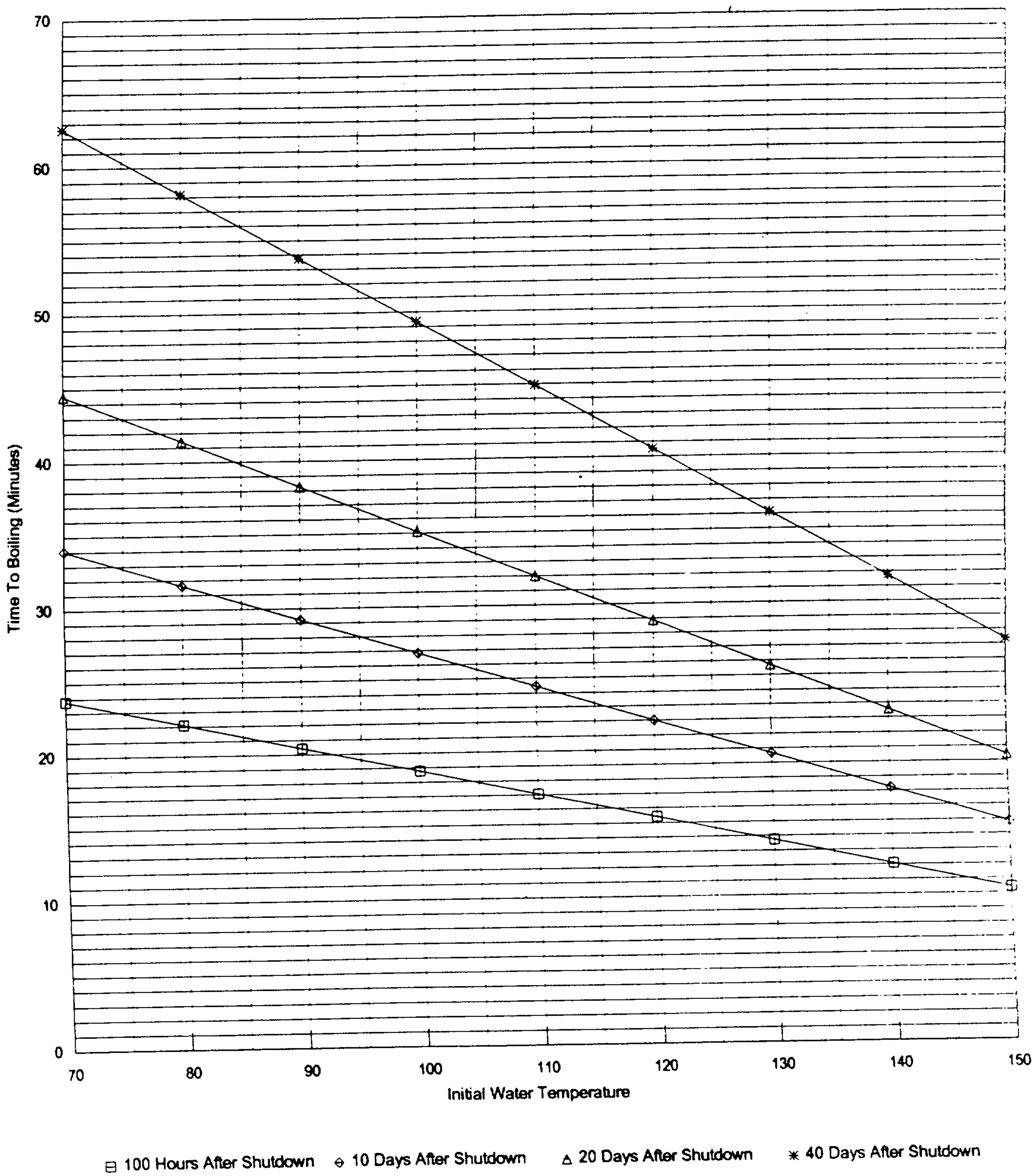
830 Day Full Power TID Core
DBA Conditions
Hydrogen Sources:

- Zirconium-Water Reaction
- Aluminum Corrosion
- Core Solution Radiolysis
- Sump Solution Radiolysis

Drawn By: 10-19-84

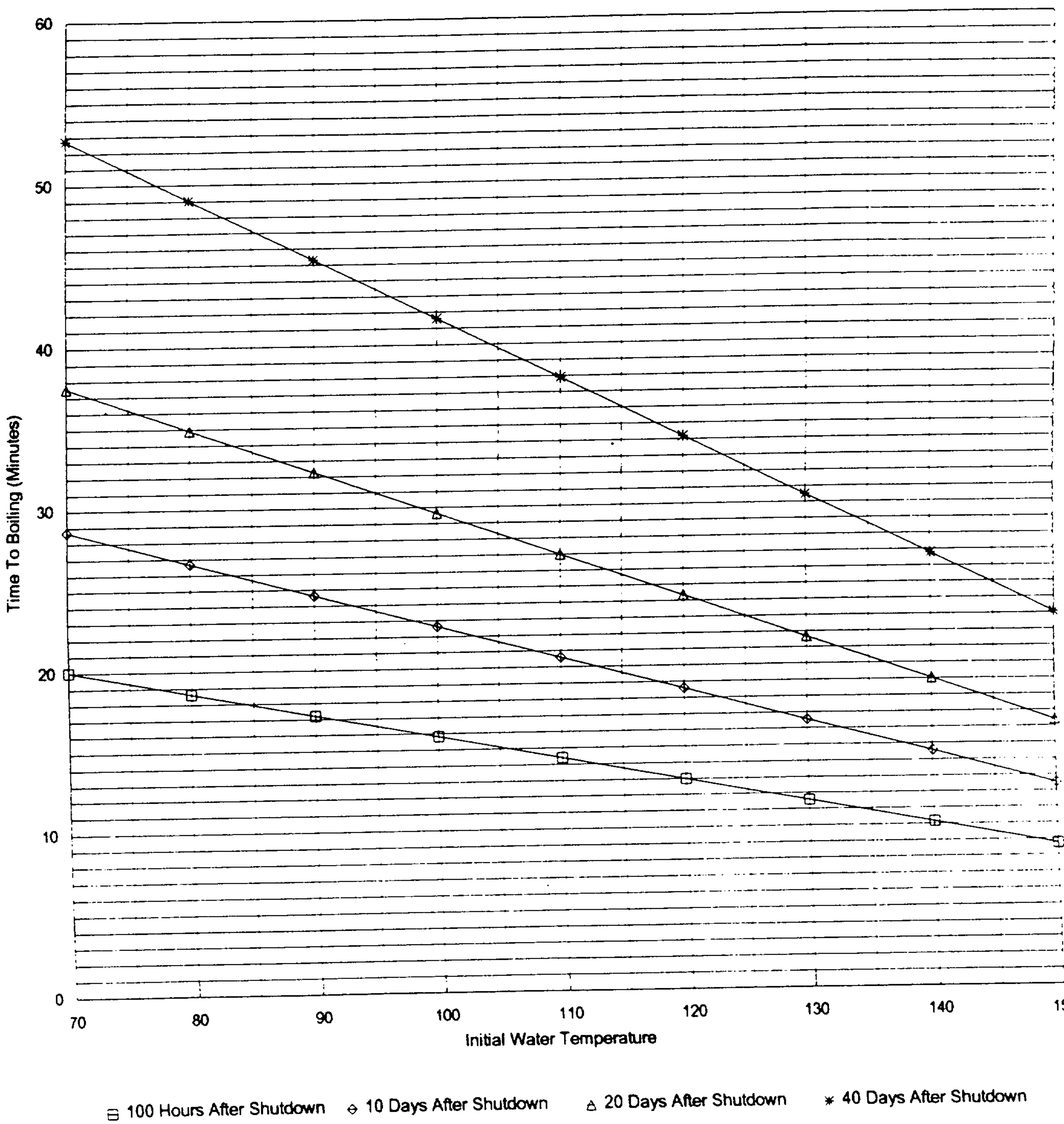
Checked By: 10/19/84

Curve 7.19 - Loss of Residual Heat Removal Cooling
Water Level Between 0" to -10" Below Vessel Flange



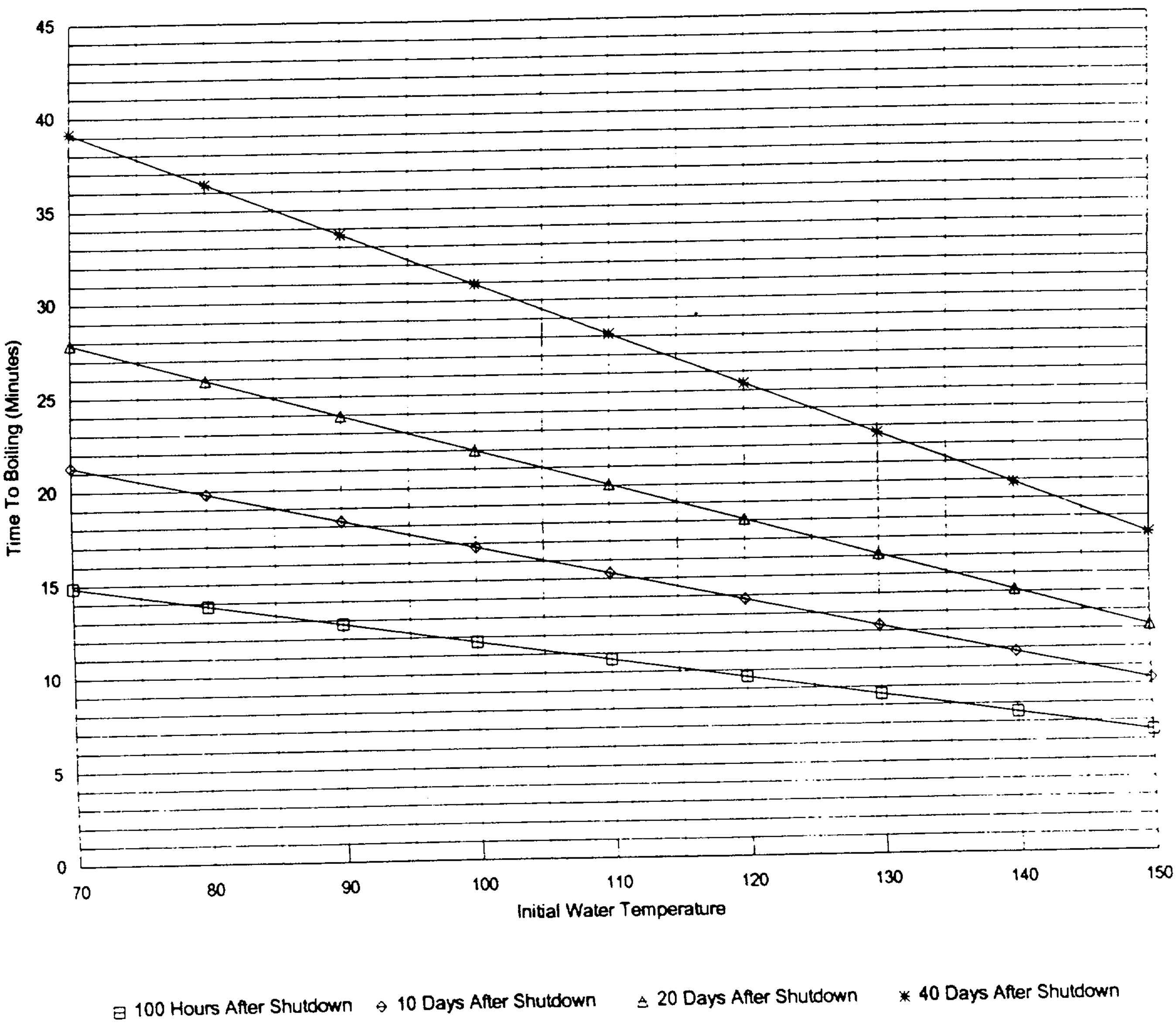
Based on calculation RNP-M/MECH-1590

Curve 7.20 - Loss of Residual Heat Removal Cooling Water Level Between -10" to -36" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

Curve 7.21 - Loss of Residual Heat Removal Cooling
Water Level Between -36" to -72" Below Vessel Flange

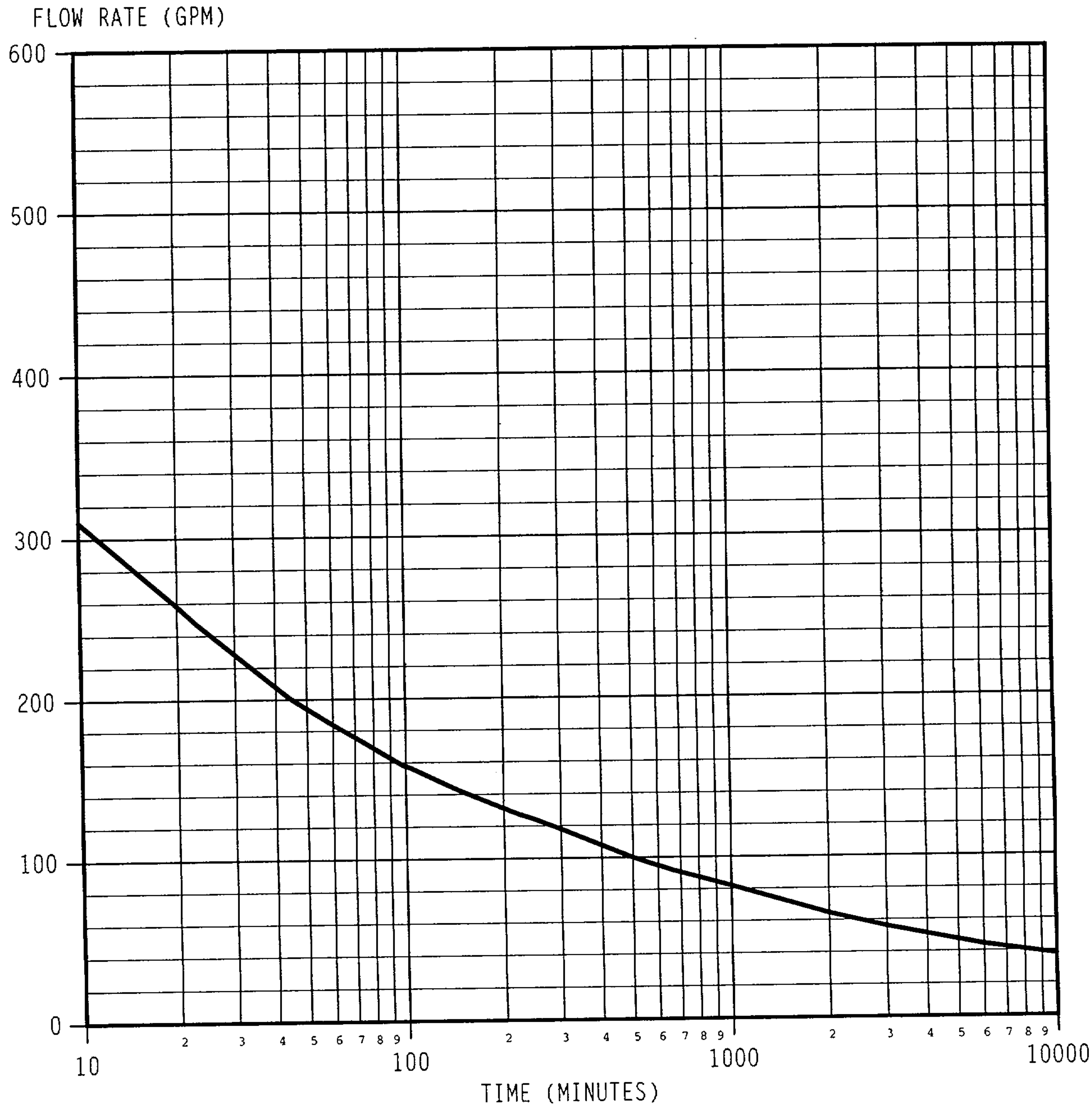


Based on calculation RNP-M/MECH-1590

SUPPLIED REFERENCE MATERIALS FOR RNP NRC REACTOR OPERATOR EXAMINATION

REFERENCE NUMBER	REFERENCE TITLE
NA	Steam Tables
EPP-15, Attachment 1	Required Flow Rate Versus Time After Reactor Trip
GP-005, Attachment 10.1	Reactor Power Ascension Indicator Log
Plant Curve 5.3	Boron Addition – Coolant Hot - Gallons
Plant Curve 5.4	Boron Addition – Coolant Cold - Gallons
Plant Curve 5.7	Dilution – Coolant Hot - Gallons
Plant Curve 5.8	Dilution – Coolant Cold - Gallons
Plant Curve 7.6	System Resistance Curve, Post Accident Containment Venting System
Plant Curve 7.16	Total Hydrogen Generation Rate from All Sources
Plant Curve 7.19	Loss of Residual Heat Removal Cooling Water Level Between 0" to –10" Below Vessel Flange
Plant Curve 7.20	Loss of Residual Heat Removal Cooling Water Level Between –10" to –36" Below Vessel Flange
Plant Curve 7.21	Loss of Residual Heat Removal Cooling Water Level Between –36" to –72" Below Vessel Flange

ATTACHMENT 1
REQUIRED FLOW RATE VERSUS TIME AFTER REACTOR TRIP
Page 1 of 1



REACTOR POWER ASCENSION INDICATOR LOG

AVG PWR % (1)	NI-35 amps	NI-36 amps	NI-41A %	NI-42A %	NI-43A %	NI-44A %	LOOP ΔT °F (1)	LOOP 1 ΔT °F	LOOP 2 ΔT °F	LOOP 3 ΔT °F	1st STAGE PRESS psig (1)	PI-446 **OR** 447 psig (2)	NET MWe MAX (1)	NET MWe	CCP % PWR (3)	NR-45 (4)	SSO (1)
15-20							9-11.5				68-90		73				
25-30							14.5-17				113-135		153				
35-40							20-23				158-180		235				
45-50							26-28.5				207-230		316				
55-60							32-34.5				261-285		398				
65-70							37-40				320-345		480				
75-80							43-46				384-410		562				
85-90							49-51.5				449-475		643				
95-100							55-57.5				513-540		725				

(1) Listed ranges and Net MWe maximums are predicted based on past plant performance. The maximum value of each indication is the maximum target value for each power increase. The SSO shall initial if plant management has determined that indications are acceptable to continue with the power escalation.

(2) Use indicator that corresponds to the channel selected on the 1st STAGE PRESSURE selector switch.

(3) Record Continuous Calorimetric Program % Power.

(4) Verify NR-45 is selected to the highest reading channel.

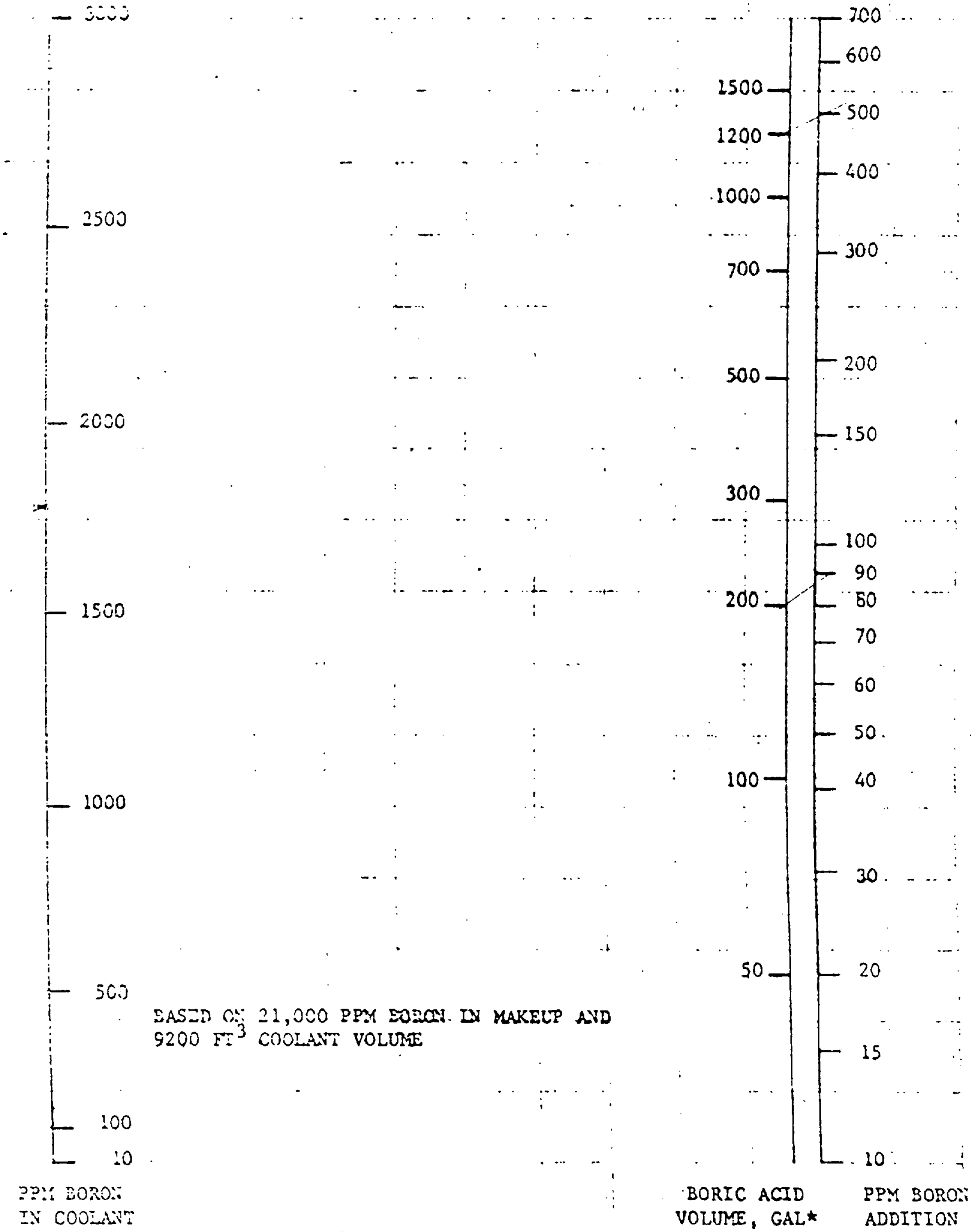


FIGURE S-3.1-3 BORON ADDITION - COOLANT HOT (~580°F)

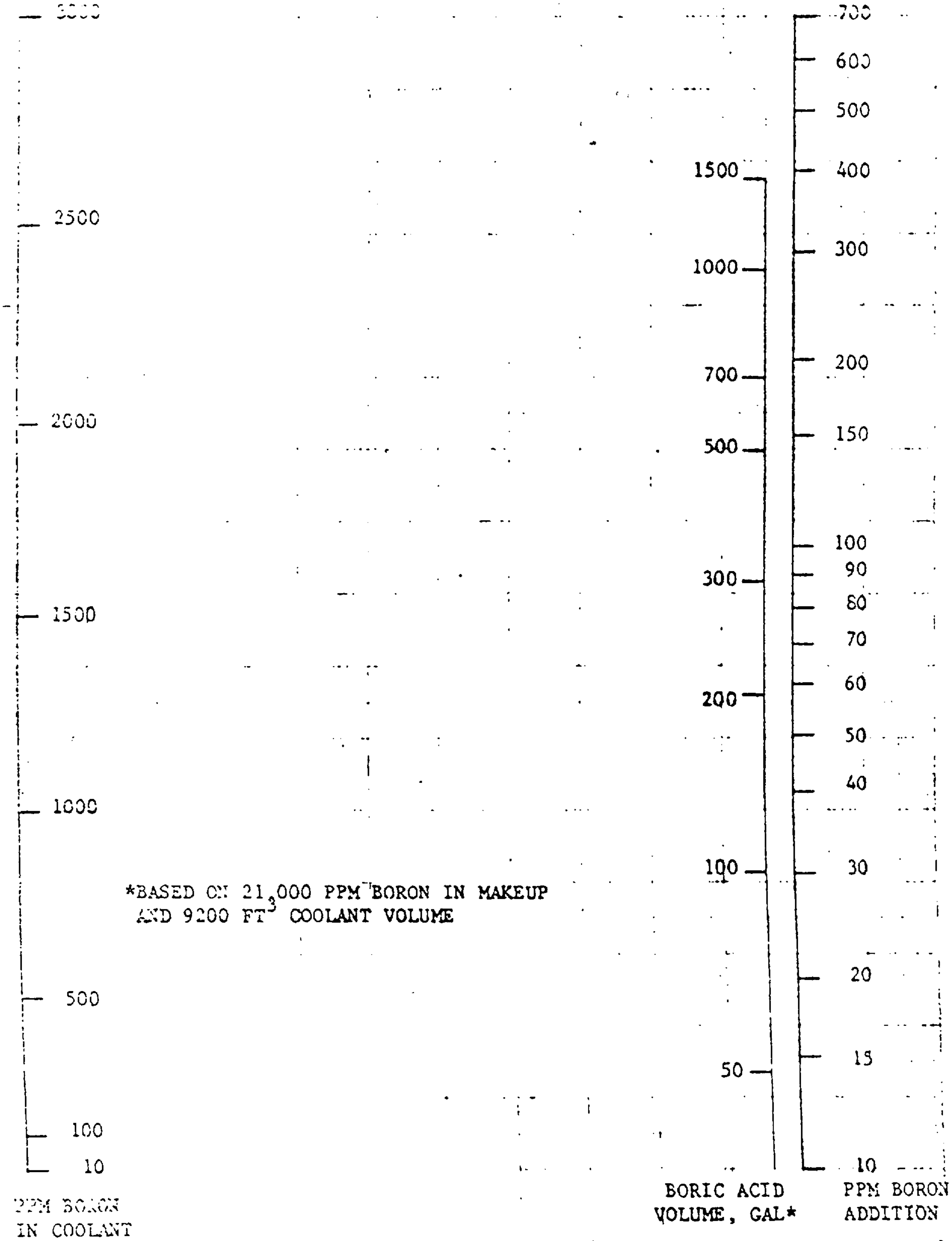


FIGURE S-3.1-4 BORON ADDITION - COOLANT COLD (-100°F)

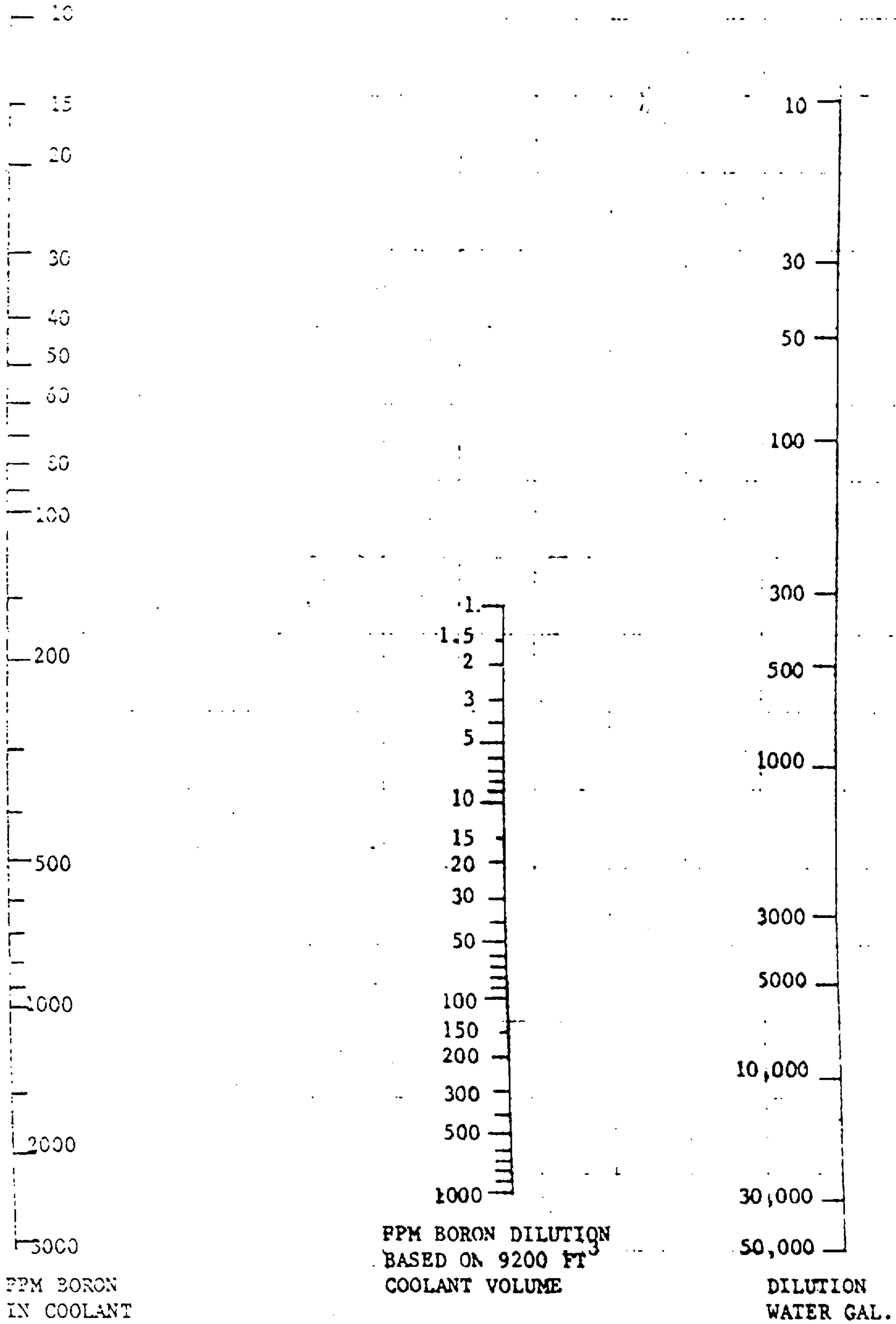


FIGURE S-3.1-7 DILUTION NOMOGRAPH - COOLANT HOT (~580°F)

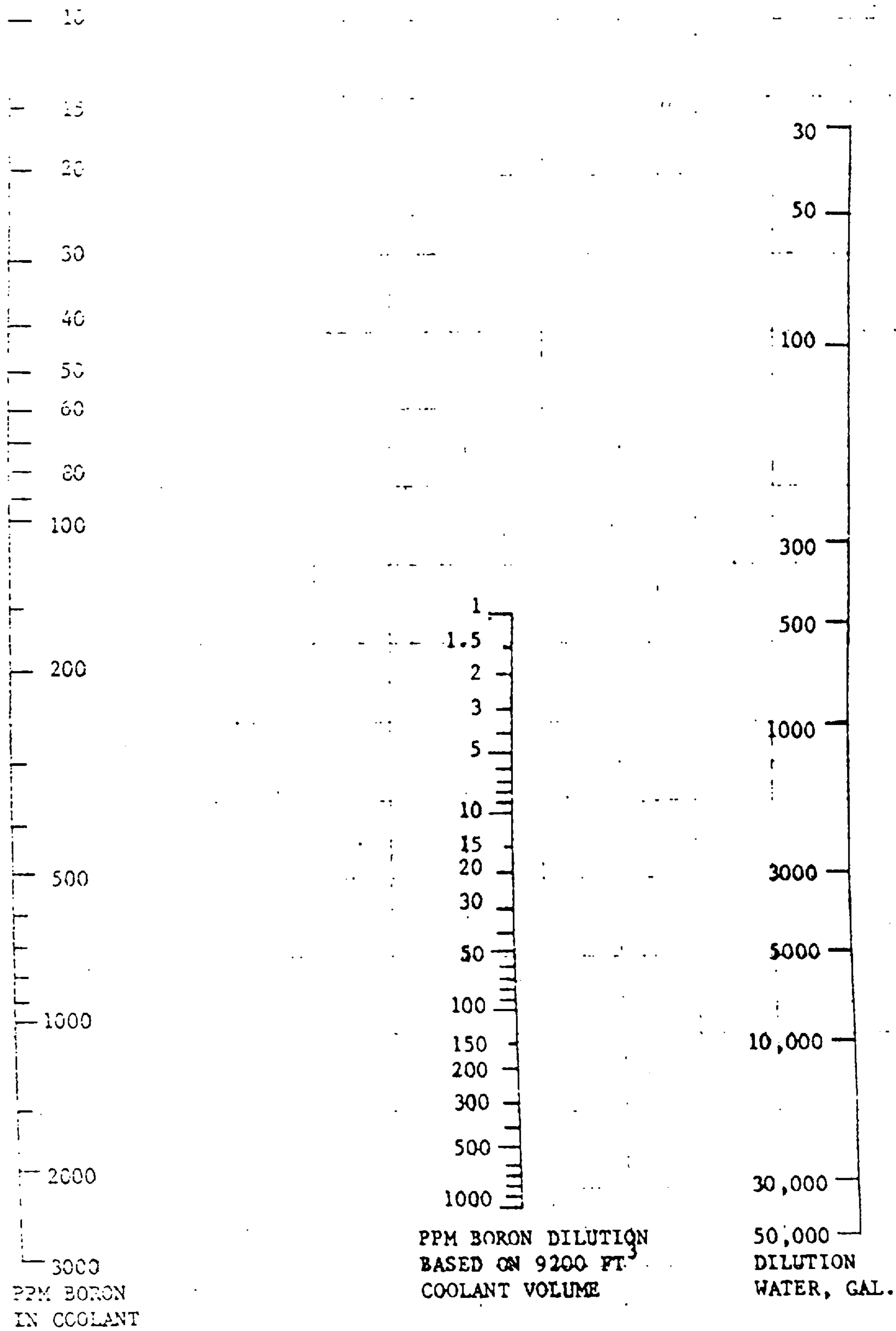


FIGURE S-3.1-8 DILUTION NOMOGRAPH - COOLANT COLD (~100°F)

46 1240

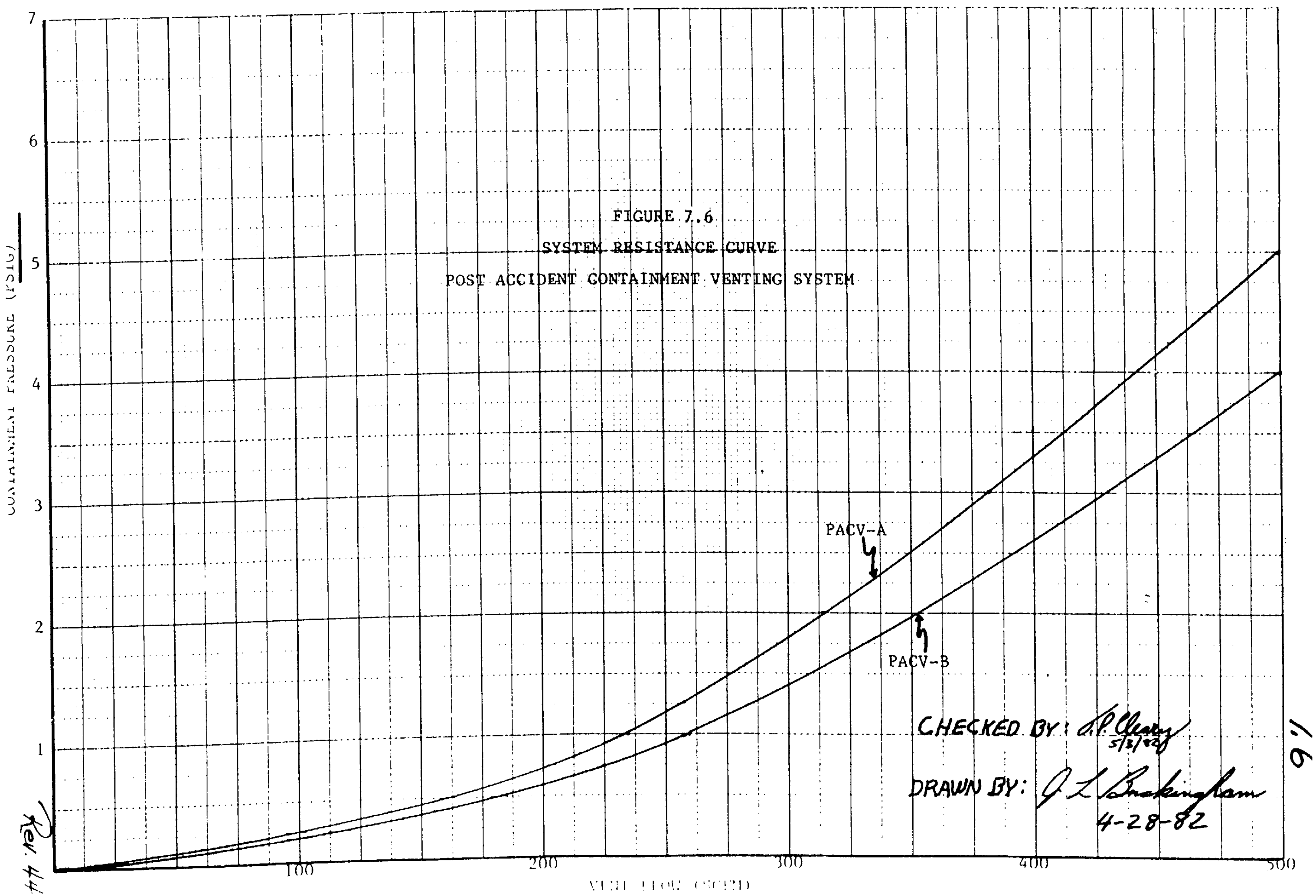
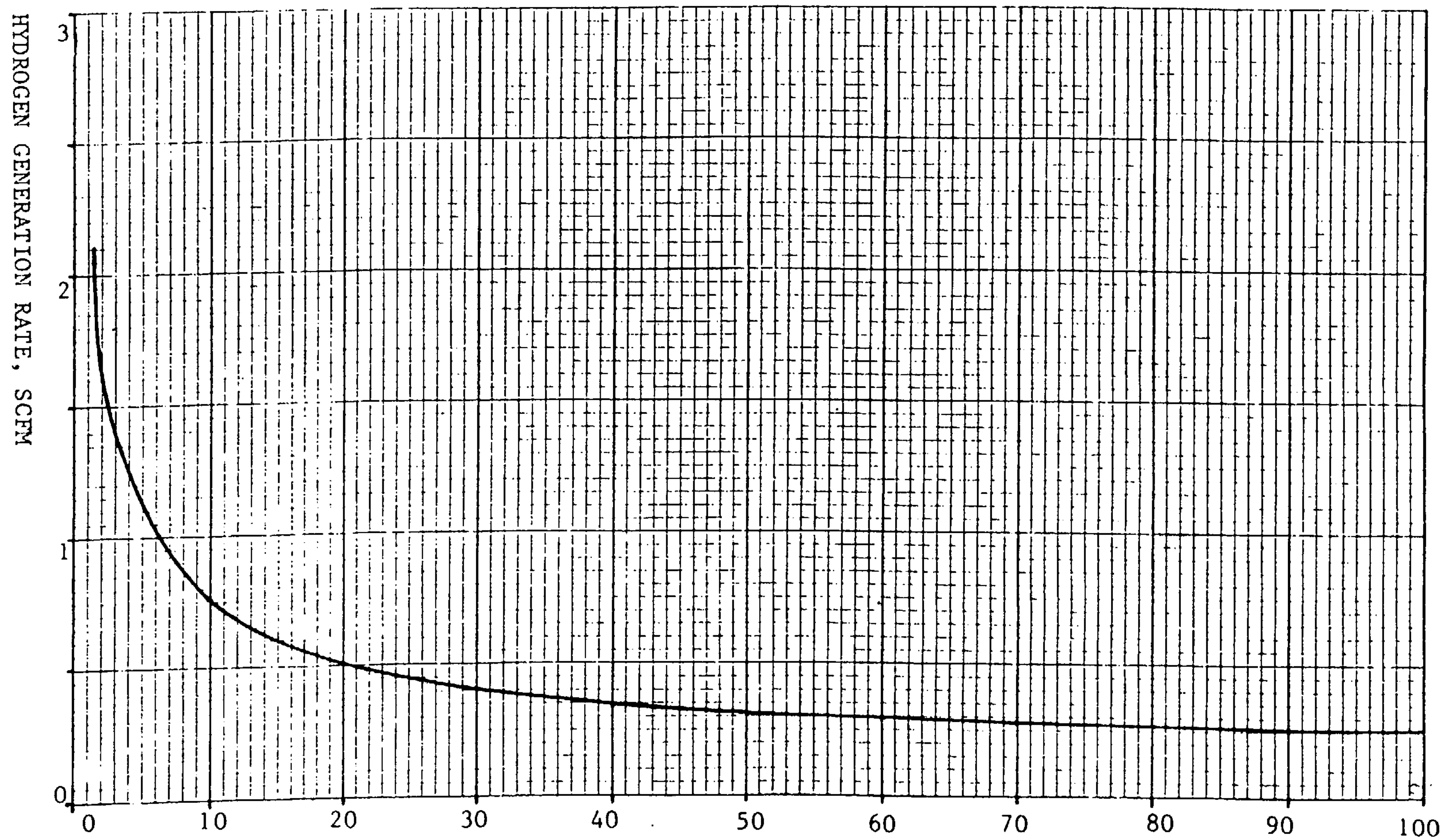


FIGURE 7.6
SYSTEM RESISTANCE CURVE
POST ACCIDENT CONTAINMENT VENTING SYSTEM
CONTAINMENT PRESSURE (PSIG)
7
6
5
4
3
2
1
100
200
300
400
500
PACV-A
PACV-B
CHECKED BY:
5/3/82
DRAWN BY:
4-28-82
Rev. 44
9.1

CURVE 7.16 TOTAL HYDROGEN GENERATION RATE FROM ALL SOURCES.



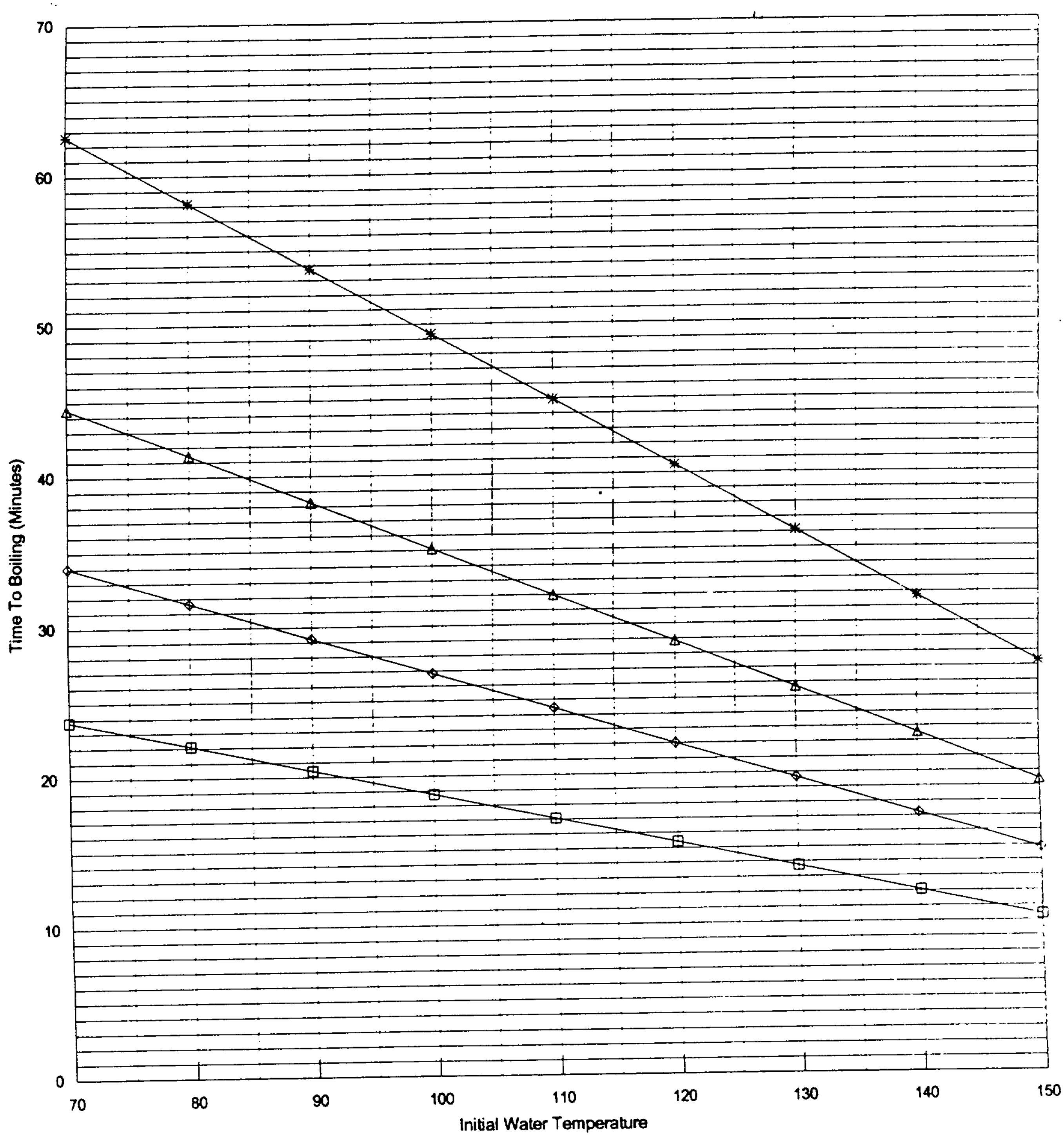
Drawn By: James M. [signature] 10-19-84

Checked By: [signature] 10/19/84

830 Day Full Power TID Core
DBA Conditions
Hydrogen Sources:

- Zirconium-Water Reaction
- Aluminum Corrosion
- Core Solution Radiolysis
- Sump Solution Radiolysis

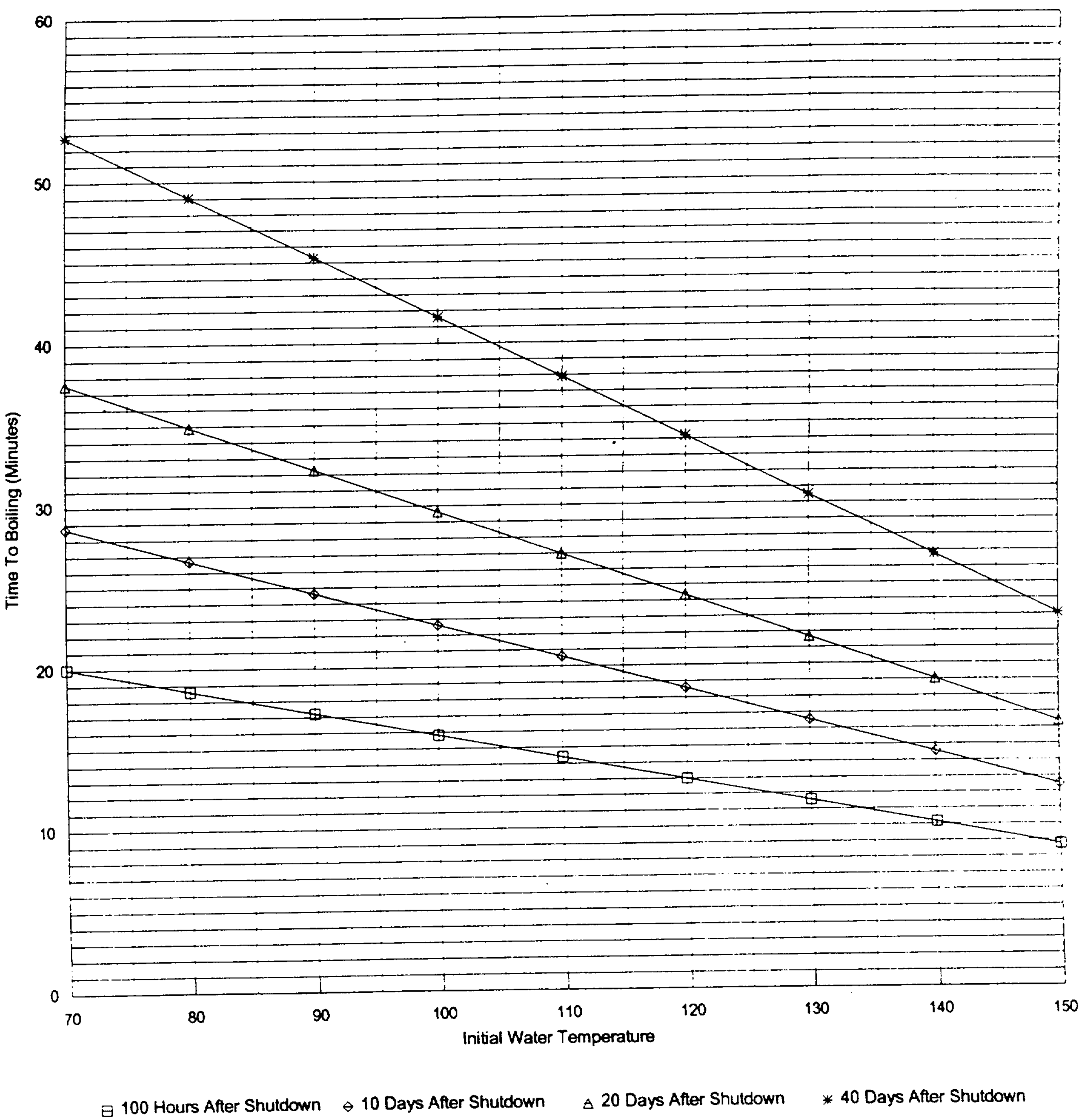
Curve 7.19 - Loss of Residual Heat Removal Cooling
Water Level Between 0" to -10" Below Vessel Flange



⊟ 100 Hours After Shutdown ◇ 10 Days After Shutdown △ 20 Days After Shutdown ✱ 40 Days After Shutdown

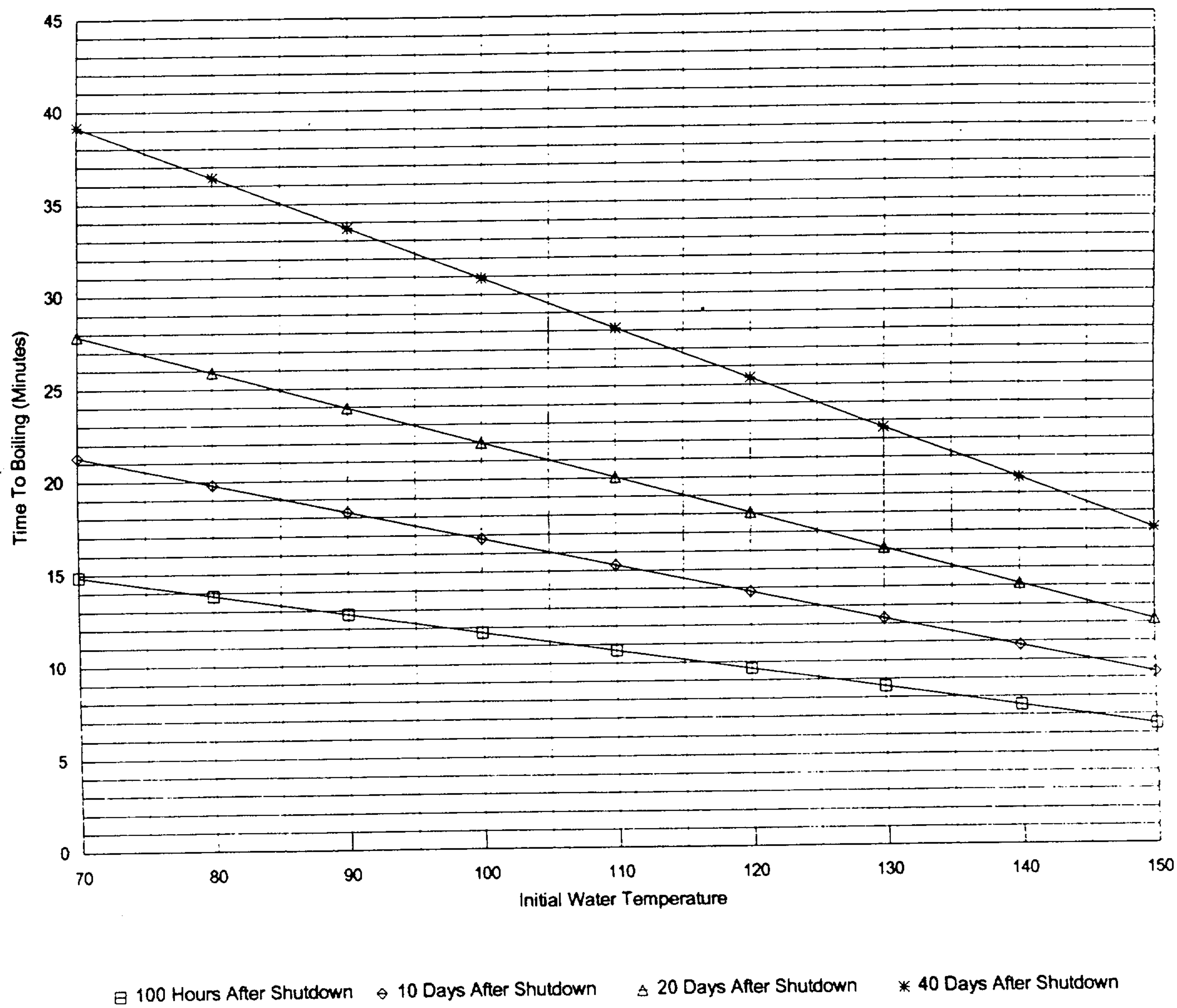
Based on calculation RNP-M/MECH-1590

Curve 7.20 - Loss of Residual Heat Removal Cooling
Water Level Between -10" to -36" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

Curve 7.21 - Loss of Residual Heat Removal Cooling Water Level Between -36" to -72" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

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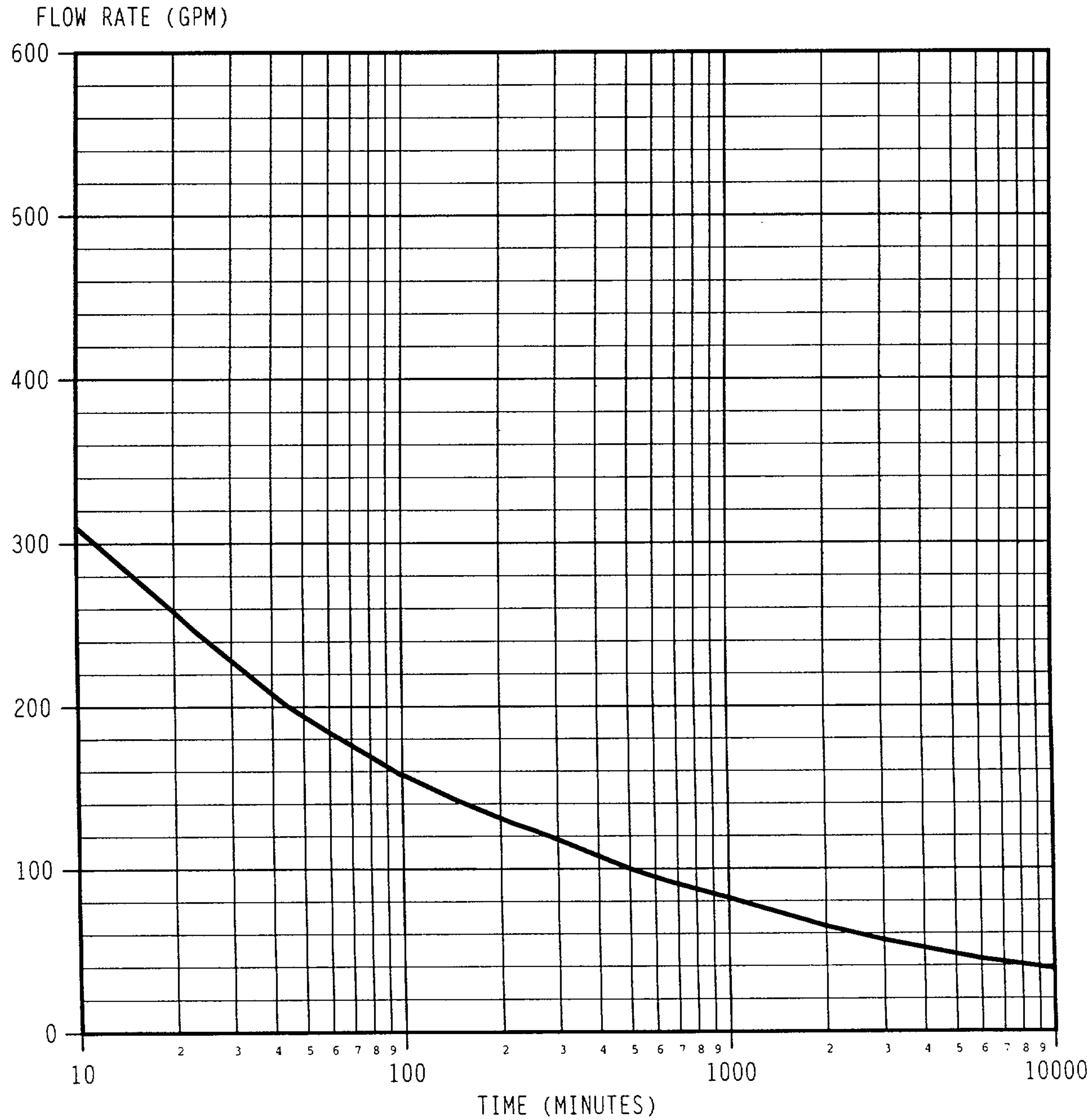
SUPPLIED REFERENCE MATERIALS FOR RNP NRC REACTOR OPERATOR EXAMINATION

REFERENCE NUMBER	REFERENCE TITLE
NA	Steam Tables
EPP-15, Attachment 1	Required Flow Rate Versus Time After Reactor Trip
GP-005, Attachment 10.1	Reactor Power Ascension Indicator Log
Plant Curve 5.3	Boron Addition – Coolant Hot - Gallons
Plant Curve 5.4	Boron Addition – Coolant Cold - Gallons
Plant Curve 5.7	Dilution – Coolant Hot - Gallons
Plant Curve 5.8	Dilution – Coolant Cold - Gallons
Plant Curve 7.6	System Resistance Curve, Post Accident Containment Venting System
Plant Curve 7.16	Total Hydrogen Generation Rate from All Sources
Plant Curve 7.19	Loss of Residual Heat Removal Cooling Water Level Between 0" to –10" Below Vessel Flange
Plant Curve 7.20	Loss of Residual Heat Removal Cooling Water Level Between –10" to –36" Below Vessel Flange
Plant Curve 7.21	Loss of Residual Heat Removal Cooling Water Level Between –36" to –72" Below Vessel Flange

ATTACHMENT 1
REQUIRED FLOW RATE VERSUS TIME AFTER REACTOR TRIP
Page 1 of 1



REACTOR POWER ASCENSION INDICATOR LOG

AVG PWR % (1)	NI-35 amps	NI-36 amps	NI-41A %	NI-42A %	NI-43A %	NI-44A %	LOOP ΔT °F (1)	LOOP 1 ΔT °F	LOOP 2 ΔT °F	LOOP 3 ΔT °F	1st STAGE PRESS psig (1)	PI-446 **OR** 447 psig (2)	NET MWe MAX (1)	NET MWe	CCP % PWR (3)	NR-45 (4)	SSO (1)
15-20							9-11.5				68-90		73				
25-30							14.5-17				113-135		153				
35-40							20-23				158-180		235				
45-50							26-28.5				207-230		316				
55-60							32-34.5				261-285		398				
65-70							37-40				320-345		480				
75-80							43-46				384-410		562				
85-90							49-51.5				449-475		643				
95-100							55-57.5				513-540		725				

(1) Listed ranges and Net MWe maximums are predicted based on past plant performance. The maximum value of each indication is the maximum target value for each power increase. The SSO shall initial if plant management has determined that indications are acceptable to continue with the power escalation.

(2) Use indicator that corresponds to the channel selected on the 1st STAGE PRESSURE selector switch.

(3) Record Continuous Calorimetric Program % Power.

(4) Verify NR-45 is selected to the highest reading channel.

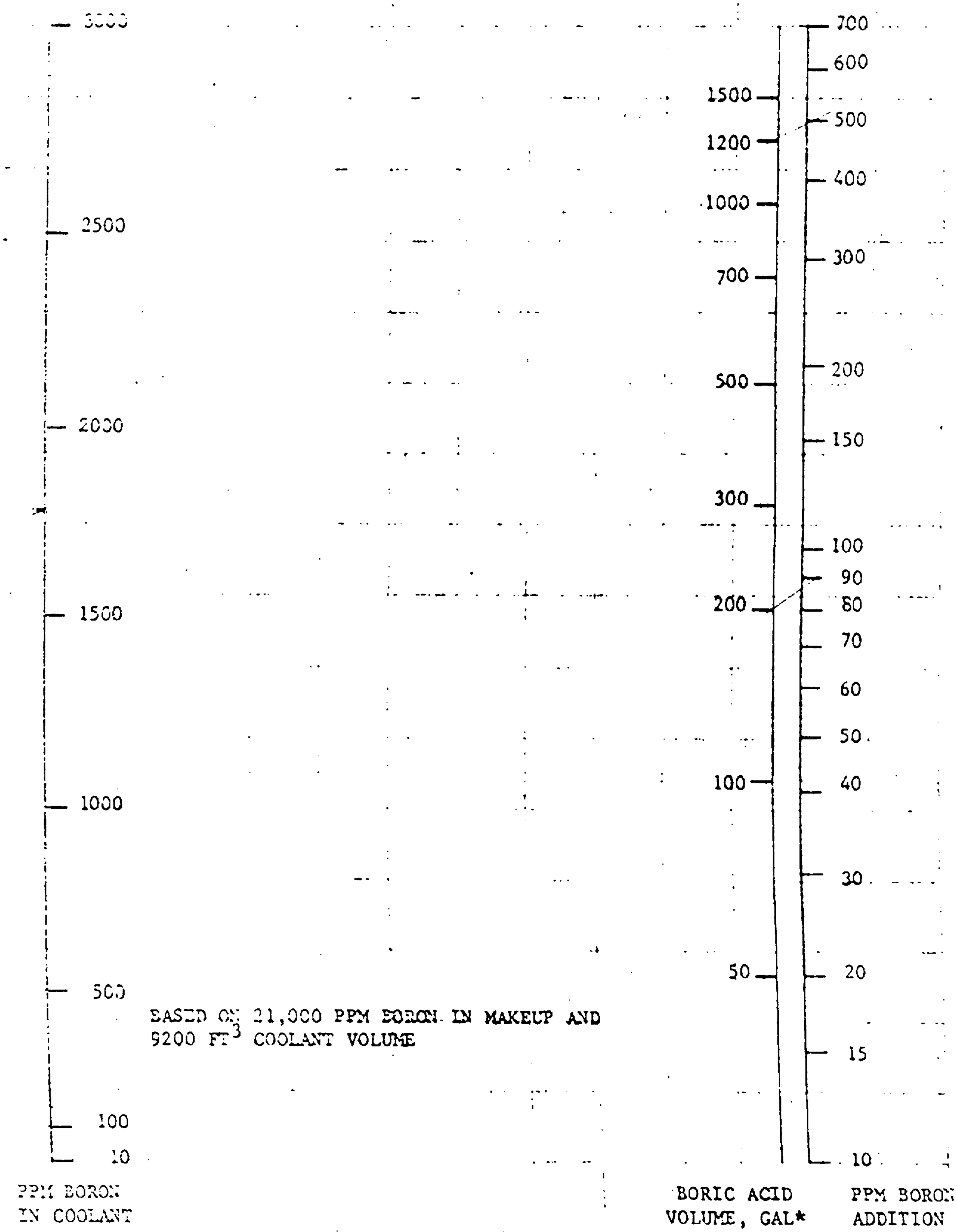


FIGURE S-3.1-3 BORON ADDITION - COOLANT HOT (~580°F)

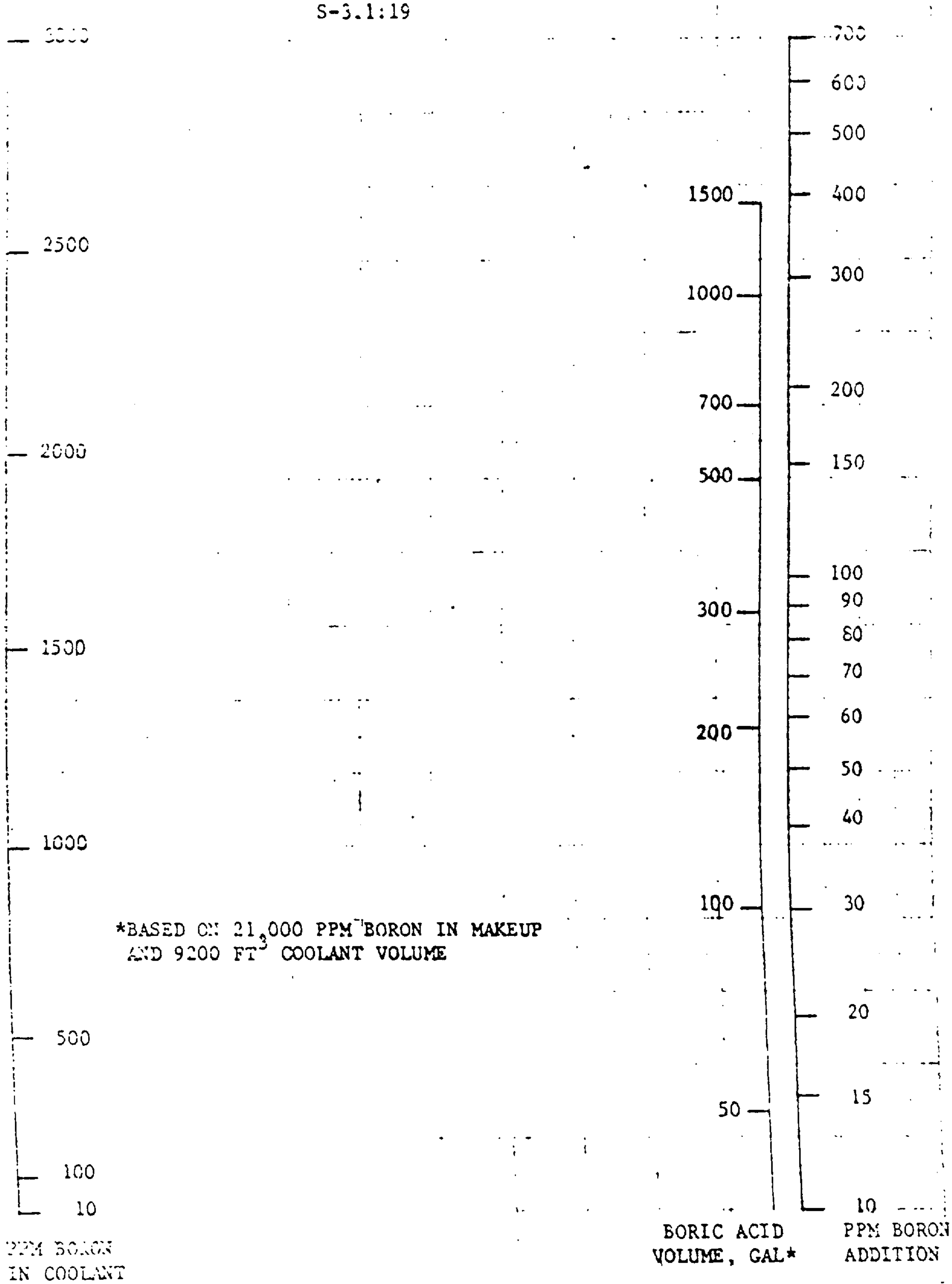


FIGURE S-3.1-4 BORON ADDITION - COOLANT COLD (-100°F)

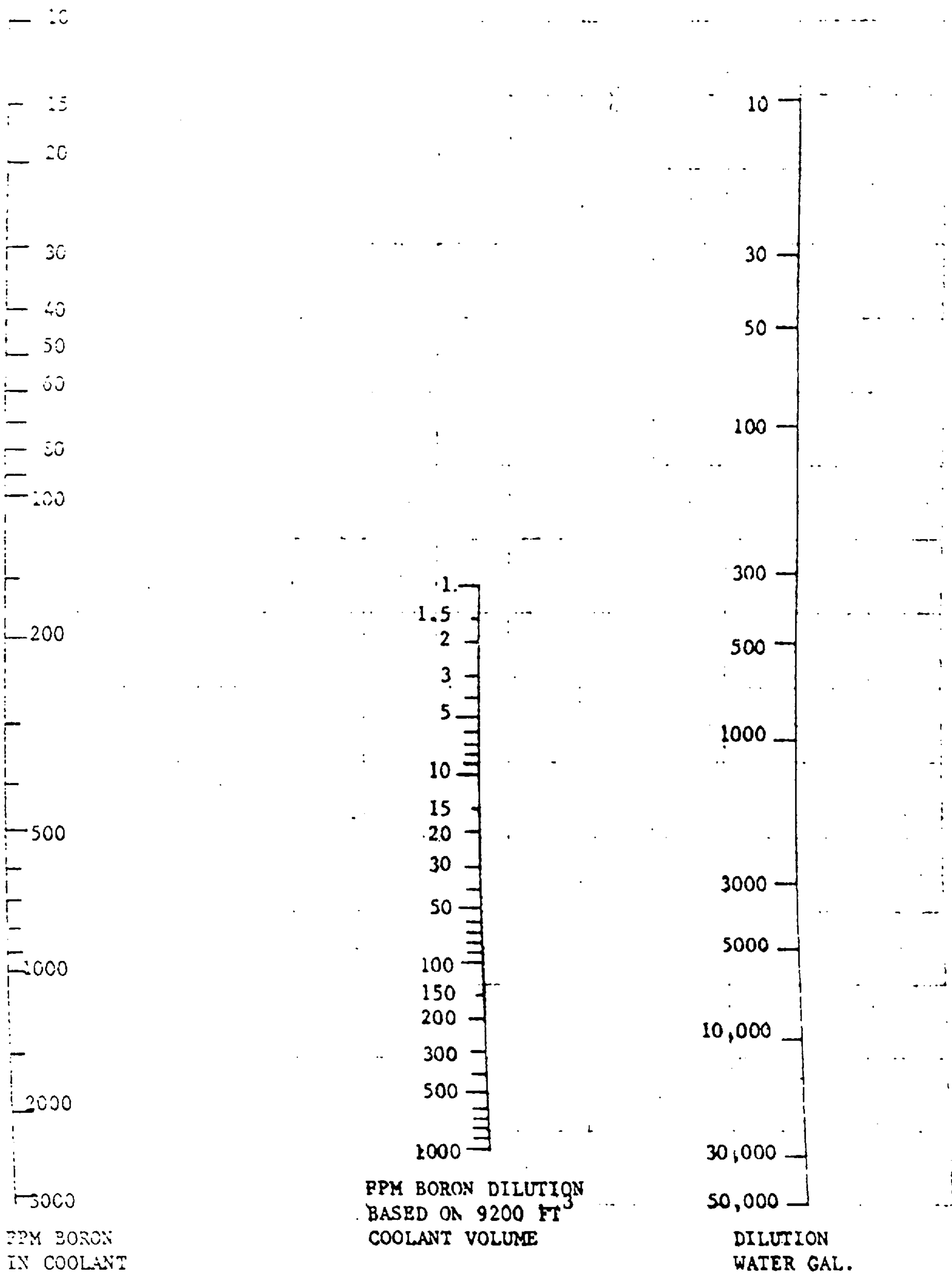


FIGURE S-3.1-7 DILUTION NOMOGRAPH - COOLANT HOT (-580°F)

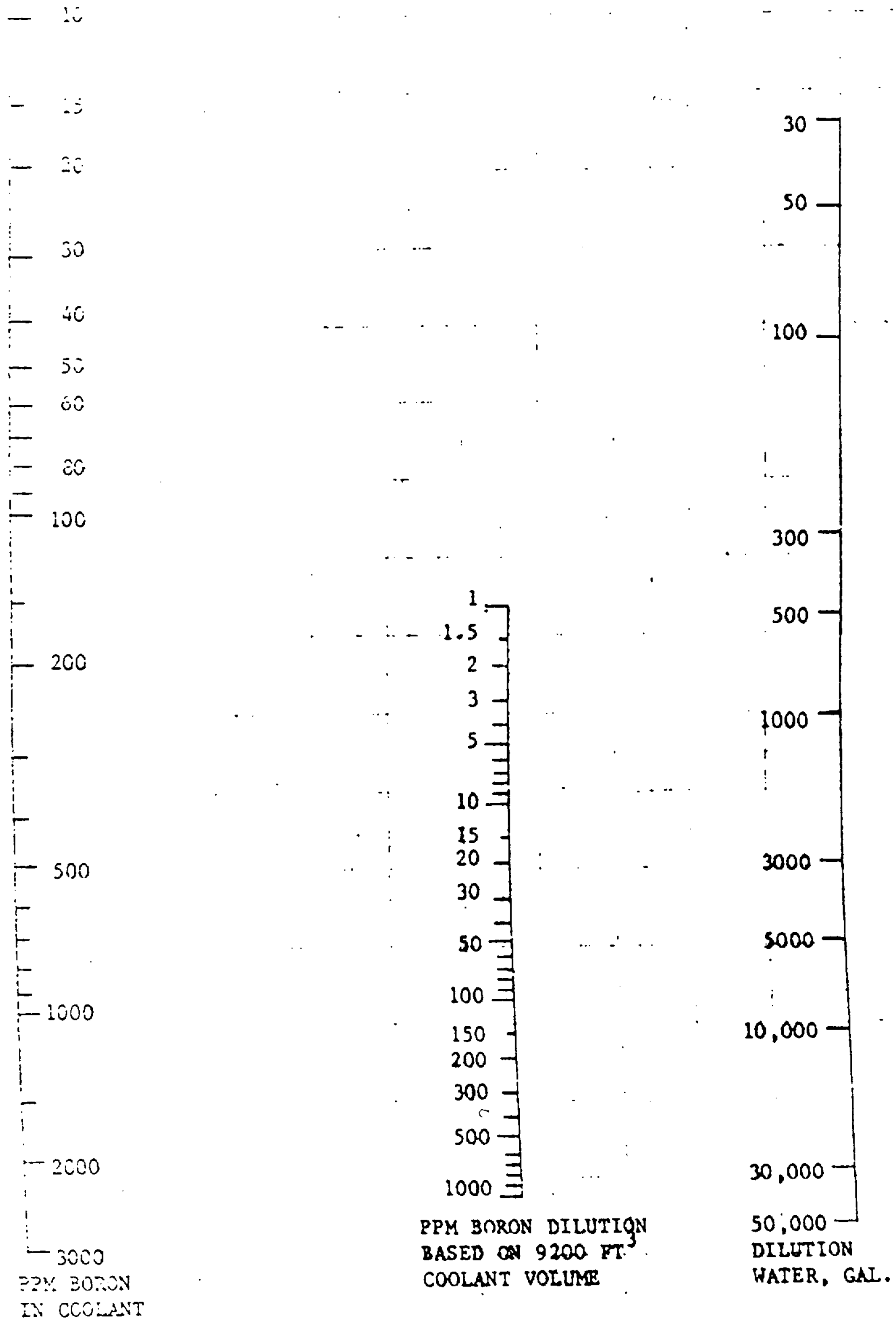


FIGURE S-3.1-8 DILUTION NOMOGRAPH - COOLANT COLD (-100°F)

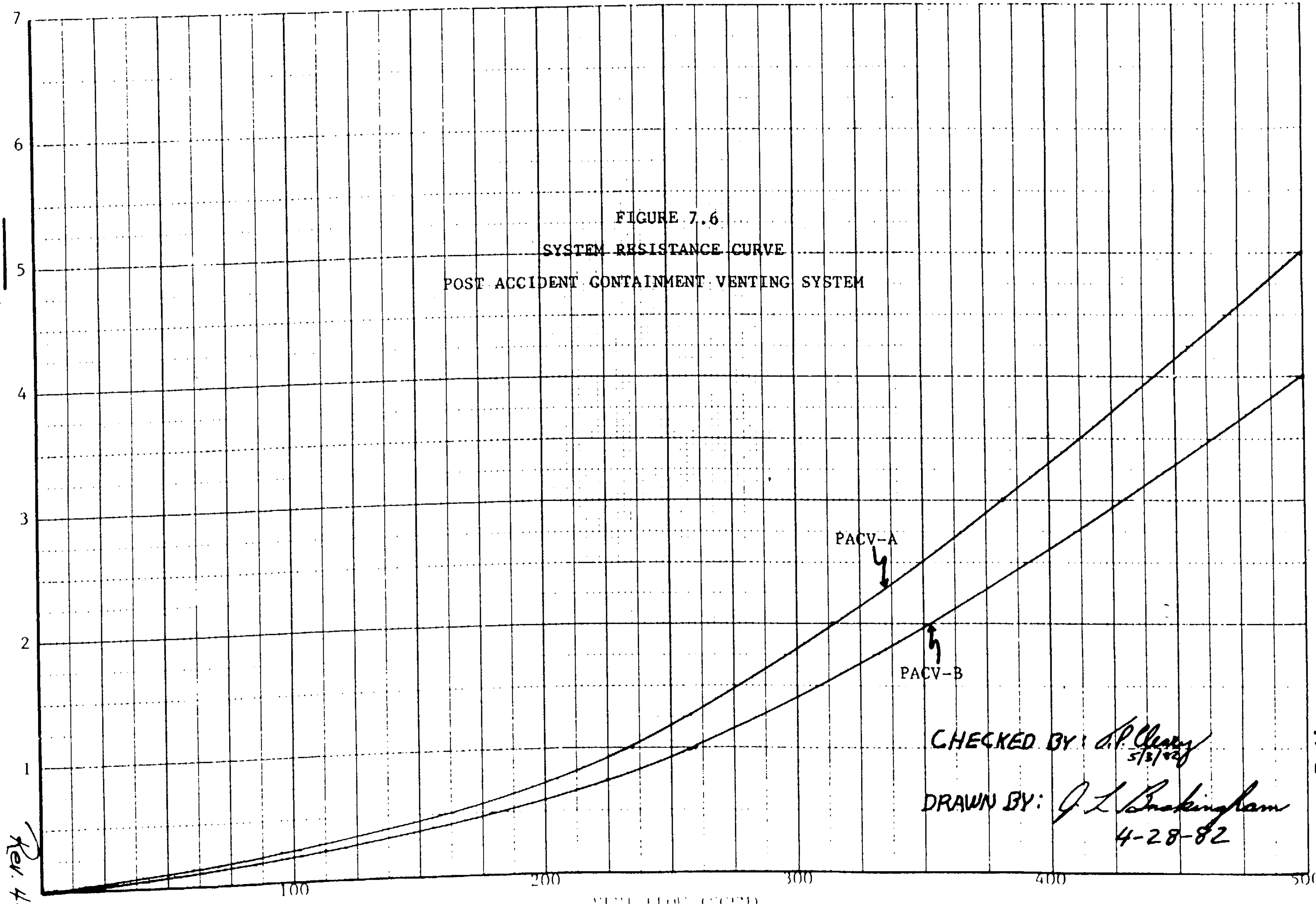
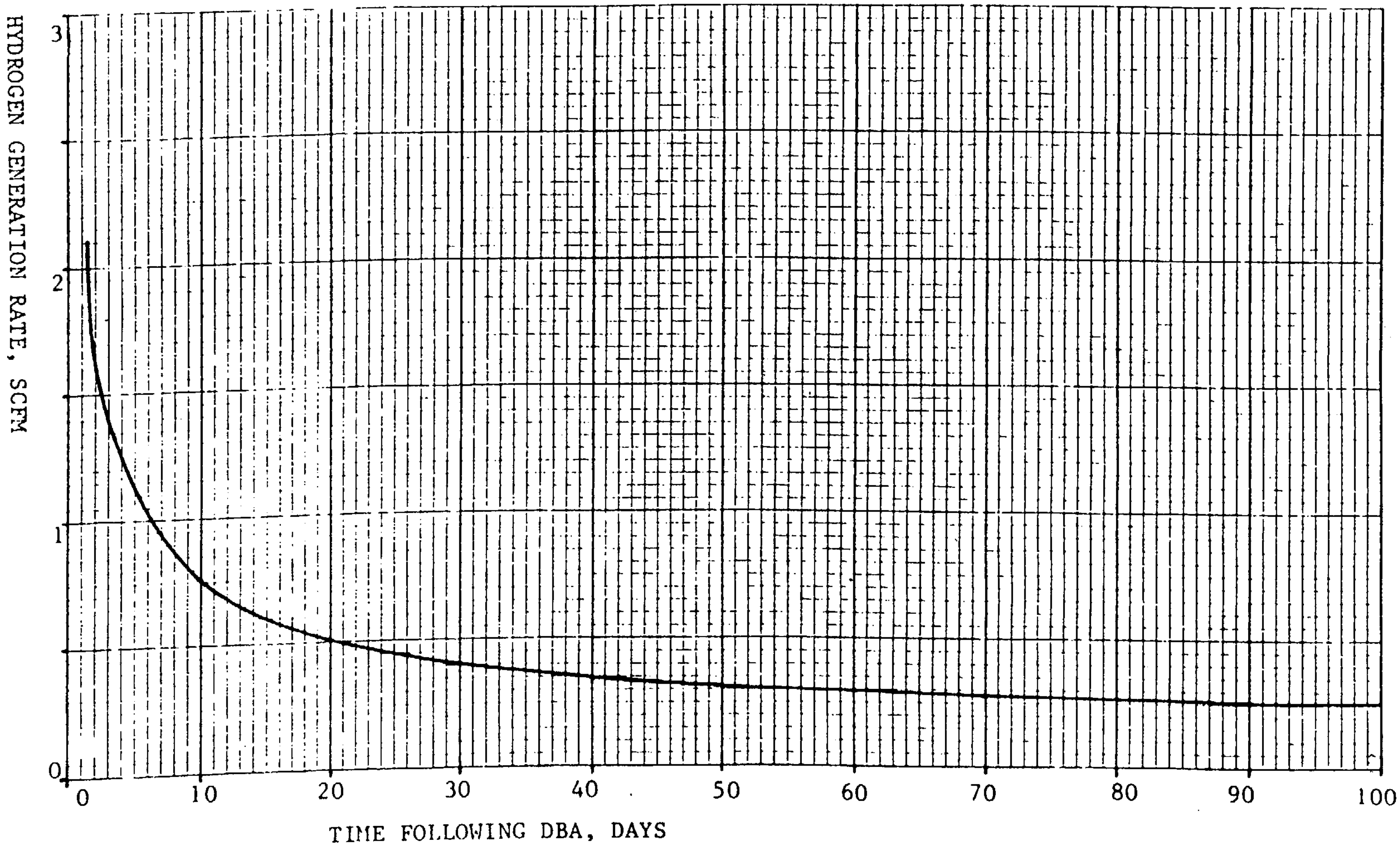


FIGURE 7.6
SYSTEM RESISTANCE CURVE
POST ACCIDENT CONTAINMENT VENTING SYSTEM
PACV-A
PACV-B
7
6
5
4
3
2
1
100
200
300
400
500
CHECKED BY:
5/3/82
DRAWN BY:
4-28-82

Rev. 44



CURVE 7.16 TOTAL HYDROGEN GENERATION RATE FROM ALL SOURCES.

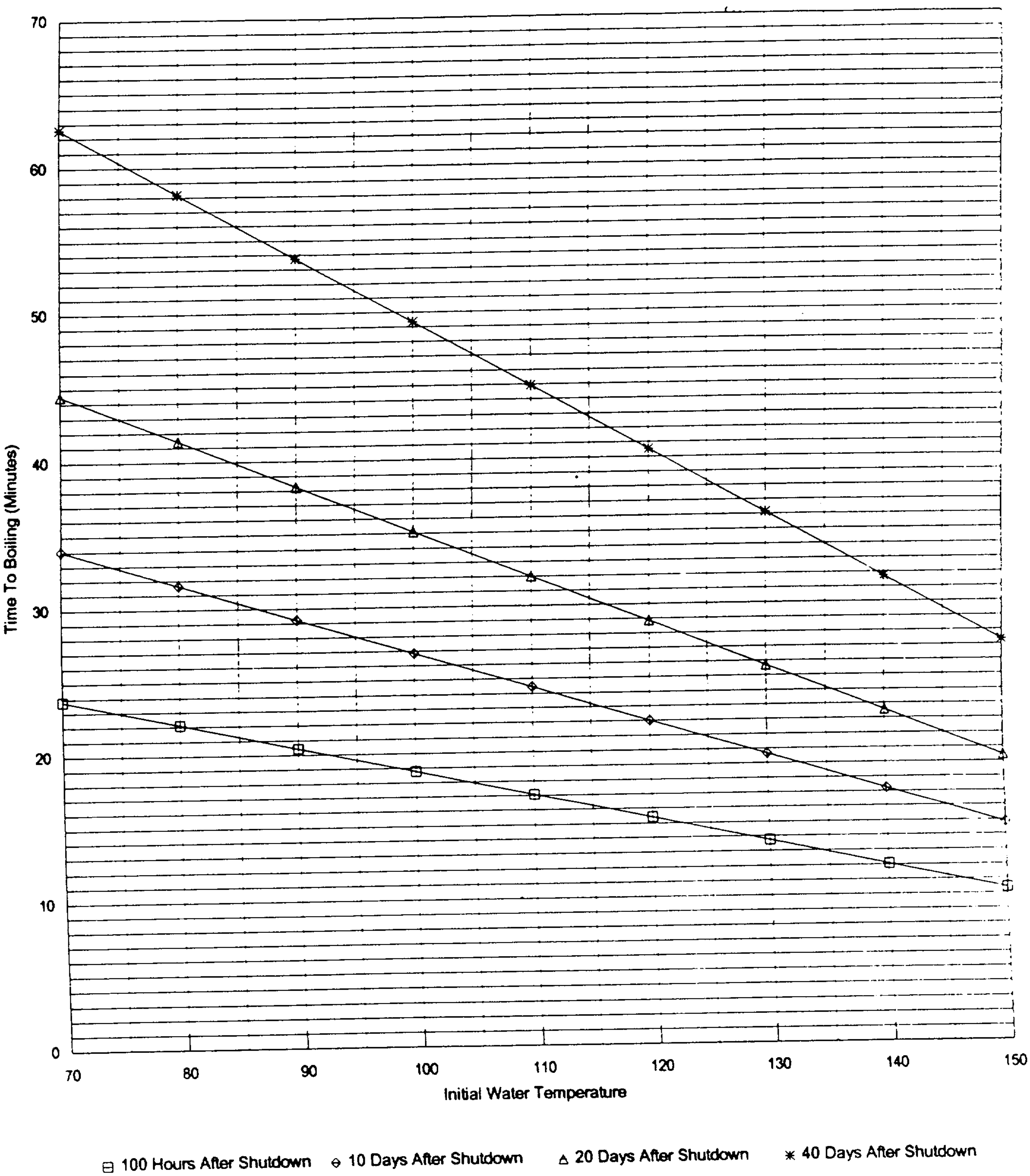
830 Day Full Power TID Core
DBA Conditions
Hydrogen Sources:

- Zirconium-Water Reaction
- Aluminum Corrosion
- Core Solution Radiolysis
- Sump Solution Radiolysis

Drawn By: 10-19-84

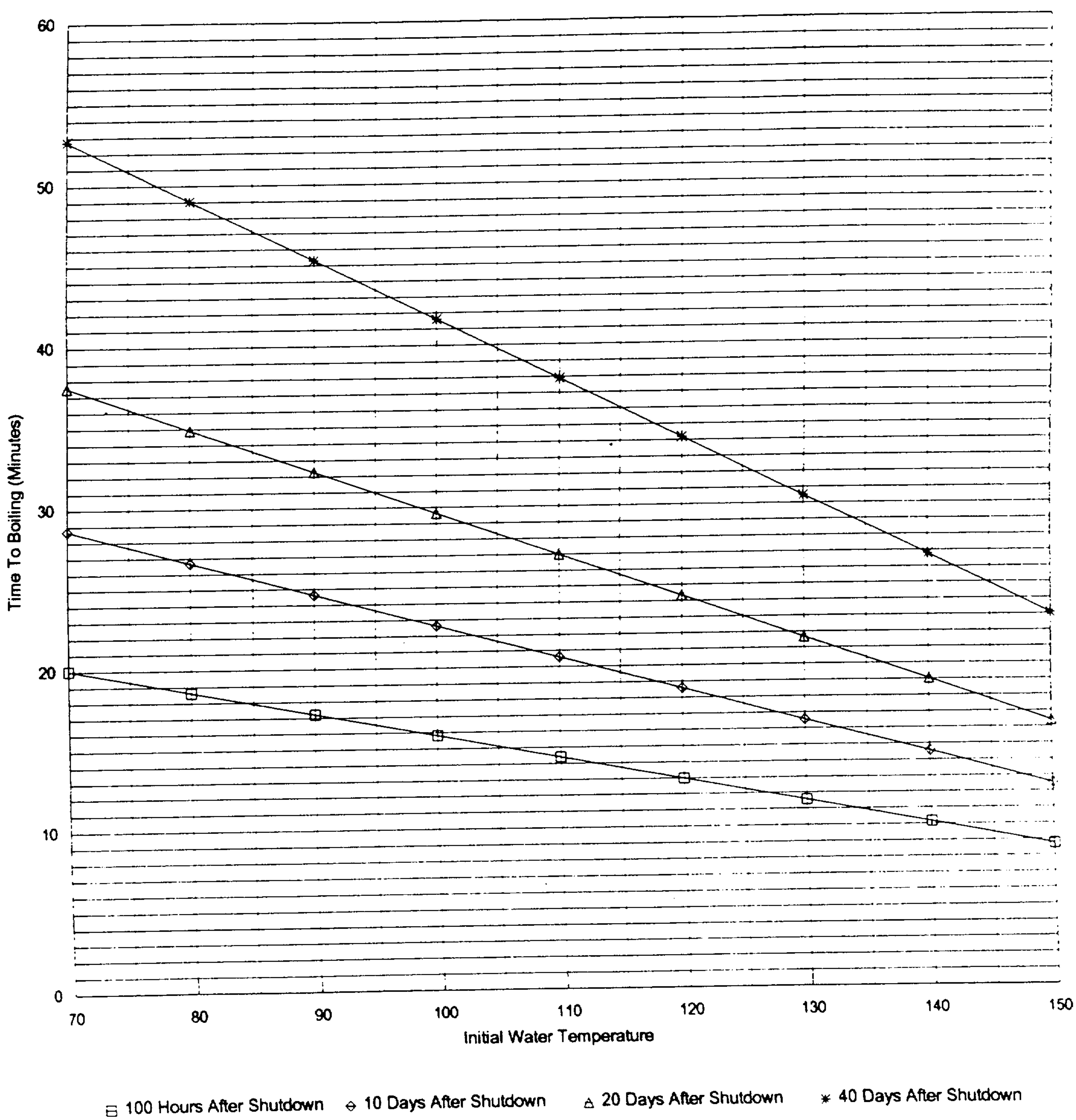
Checked By: 10/19/84

Curve 7.19 - Loss of Residual Heat Removal Cooling
Water Level Between 0" to -10" Below Vessel Flange



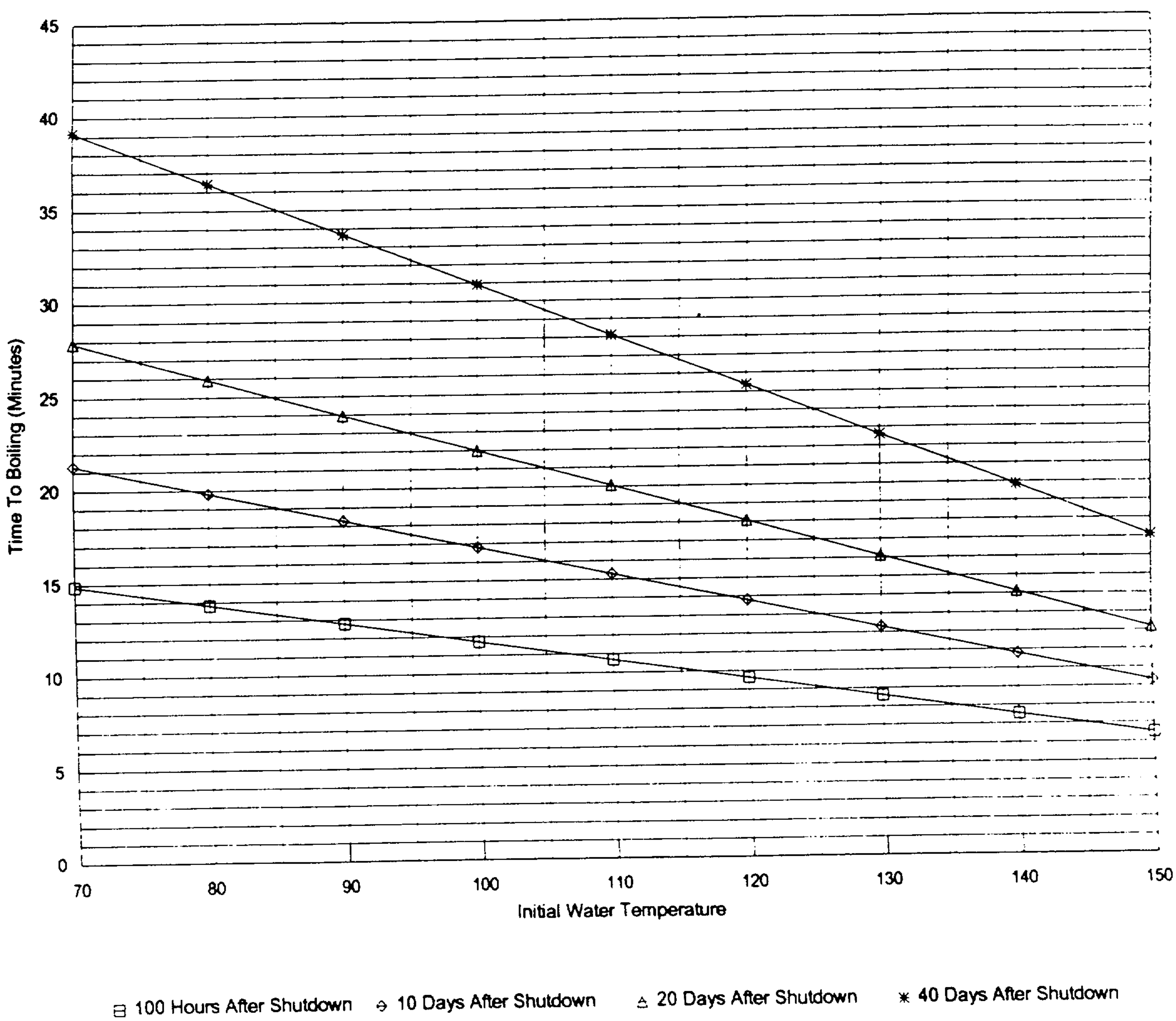
Based on calculation RNP-M/MECH-1590

Curve 7.20 - Loss of Residual Heat Removal Cooling
Water Level Between -10" to -36" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

Curve 7.21 - Loss of Residual Heat Removal Cooling Water Level Between -36" to -72" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

SUPPLIED REFERENCE MATERIALS FOR RNP NRC SENIOR REACTOR OPERATOR EXAMINATION

REFERENCE NUMBER	REFERENCE TITLE
NA	Steam Tables
AP-030, Attachment 7.1	Immediate (One Hour) Notifications to the NRC
AP-030, Attachment 7.2	Four Hour Notifications to the NRC
EPP-15, Attachment 1	Required Flow Rate Versus Time After Reactor Trip
EPP-17, Attachment 1	Containment Sump Level Vs. RWST Level
GP-005, Attachment 10.1	Reactor Power Ascension Indicator Log
OMM-046, Attachment 10.3	Available Contingency Actions
OMM-048, Attachment 10.2	PSA of On-Line Maintenance for H.B. Robinson Steam Electric Plant Unit 2
Plant Curve 3.5	Time to CV Closure
Plant Curve 5.3	Boron Addition – Coolant Hot - Gallons
Plant Curve 5.4	Boron Addition – Coolant Cold - Gallons
Plant Curve 5.7	Dilution – Coolant Hot - Gallons
Plant Curve 5.8	Dilution – Coolant Cold - Gallons
Plant Curve 7.6	System Resistance Curve, Post Accident Containment Venting System
Plant Curve 7.16	Total Hydrogen Generation Rate from All Sources
Plant Curve 7.19	Loss of Residual Heat Removal Cooling Water Level Between 0" to –10" Below Vessel Flange
Plant Curve 7.20	Loss of Residual Heat Removal Cooling Water Level Between –10" to –36" Below Vessel Flange
Plant Curve 7.21	Loss of Residual Heat Removal Cooling Water Level Between –36" to –72" Below Vessel Flange
TS 3.4.16	RCS Specific Activity
TS 3.7.4	Auxiliary Feedwater (AFW) System
TS 3.7.6	Component Cooling Water (CCW) System

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

10 CFR 50.72 states that immediate reports shall be made to the NRC Operations Center of these Emergency Events via the NRC Emergency Telecommunications System (ETS) as specified in the Emergency Plan. 10 CFR 50.72 additionally identifies Non-Emergency Events which are to be reported within One-Hour or Four-Hours to the NRC. ETS Telephones, which are identified, are located in the Control Room, the TSC, and the EOF.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
NOTE: 10 CFR 50.72 recognizes the Emergency Plan and its four Emergency Classes of Unusual Event, Alert, Site Area Emergency and General Emergency.			
EMERGENCIES 10 CFR 50.72(a)(i) 10 CFR 30.32(i)(3)(viii) 10 CFR 40.31(i)(3)(viii)	Emergency Unusual Event Alert Site Area Emergency General Emergency	HBRSEP shall notify the NRC of the declaration of any of the Emergency Classes specified in the Emergency Plan.	– Declaration of an Unusual Event, Alert, Site Area Emergency, or General Emergency – Discovery of an event that should have resulted in an Emergency Classification, but no emergency was declared – Discovery that a declared emergency exceeded the Emergency Action Levels for a higher emergency declaration, but the higher classification was not declared
ERDS ACTIVATION 10 CFR 50.72(a)(4)	ERDS Emergency	HBRSEP shall activate the ERDS as soon as possible but not later than one hour after declaring an Alert, Site Area Emergency, or General Emergency.	– An Alert, Site Area Emergency, or General Emergency is declared.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via NRC Emergency Telecommunications System (ETS) as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SHUTDOWN REQUIRED BY TS 10 CFR 50.72(b)(1)(i)(A)	Shutdown TS Shutdown Power Reduction	The initiation of any shutdown required by the TS.	– Unplanned Shutdown initiated due to maximum specific activity of the Reactor Coolant Water (plant shutdown required by TS) – Reactor Coolant System Leakage in excess of 10 GPM for greater than 24 hours (plant shutdown required by TS) – Component Cooling Water Heat Exchanger inoperable (if not corrected prior to expiration of Required Action Completion Time)
DEVIATION FROM TS (10 CFR 50.54(X)) 10 CFR 50.72(b)(1)(i)(B)	Deviation Departure License Condition	Any deviation from the TS authorized pursuant to 10 CFR 50.54(x).	– Intentional deviation from an approved plant procedure in order to preserve plant safety 10 CFR 50.54(x) (See PRO-NGGC-0200)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRINCIPAL SAFETY BARRIERS SERIOUSLY DEGRADED 10 CFR 50.72(b)(1)(ii)	Degraded Safety Barriers Fission Product Barrier	Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;	– Fuel cladding failures in the reactor, or in the storage pool, that exceed expected values, or that are unique or widespread, or that are caused by unexpected factors, and would involve a release of significant quantities of fission products – Cracks and breaks in the piping or reactor vessel, or major components in the reactor coolant system, that have safety relevance (steam generators, reactor coolant pumps, valves, etc.) – Significant welding or material defects in the RCS – Serious temperature or pressure transients – Loss of relief and/or safety valve functions during operation – Loss of Containment function or integrity - Complete loss of containment integrity function including (1) containment leakage rate greater than allowed value per SR 3.6.1.1 (i.e., entry into LCO 3.6.1 Condition A), (2) loss of containment penetration isolation functional capability (i.e., both barriers), or loss of containment spray capability
UNANALYZED PLANT CONDITION 10 CFR 50.72(b)(1)(ii)(A)	Safety Function Unanalyzed Condition	[or that resulted in the nuclear power plant being:] In an unanalyzed condition that significantly compromises plant safety;	– OTΔT changes are declared inoperable due to summator module lag constants. The channel response time exceeded the value assumed in the accident analysis. – Accumulation of voids in systems designed to remove heat from the reactor, that could inhibit the ability to adequately remove heat from the core, particularly under natural circulation conditions

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
CONDITION OUTSIDE DESIGN BASIS OF PLANT 10 CFR 50.72(b)(1)(ii)(B)	Design Bases Loss of Safety Function	[or that resulted in the nuclear power plant being:] In a condition that is outside the design basis of the plant;	– Discovery of design errors that renders a safety system inoperable – Discovery that a single train of a safety system has been incapable of performing its design function for an extended time (well beyond surveillance intervals or Required Action Completion Times) – Safety related piping found not to be seismically qualified in accordance with design bases requirements
CONDITION NOT COVERED BY OPERATING/EMERGENCY PROCEDURES 10 CFR 50.72(b)(1)(ii)(C)	OP AOP EOP PATH CSFST	[or that resulted in the nuclear power plant being:] In a condition not covered by the operating and emergency procedures.	– An event is occurring having significant implications for the health and safety of the public and no AOP or EOP is applicable to the condition.
NATURAL PHENOMENON OR CONDITION THREATENING PLANT SAFETY 10 CFR 50.72(b)(1)(iii)	Earthquake Hurricane Tornado Weather Explosion Railroad	Any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant.	– Natural phenomenon (ice storm that significantly hampers personnel in the conduct of activities necessary for safe operation of the plant). – External hazards (railroad tank car explosion that poses an actual threat to Plant safety)
ECCS DISCHARGE INTO RCS 10 CFR 50.72(b)(1)(iv)	ECCS Actuation Safety Injection	Any event that results or should have resulted in ECCS discharge into the reactor coolant system as a result of a valid signal.	– Manual or automatic Safety Injection System actuation in response to a valid signal (Section 4.5 of this procedure)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OF EMERGENCY ASSESSMENT, OFF-SITE RESPONSE, OR COMMUNICATIONS CAPABILITY 10 CFR 50.72(b)(1)(v)	Selective Signaling System Sirens ETS	Any event that results in a major loss of emergency assessment capability, off-site response capability, or communications capability (e.g., significant portion of control room indication, ETS, or off-site notification system).	– Loss of 23 or more of 45 Public Warning Sirens ($\geq$50%) as indicated on the siren activation system for a period of at least 30 minutes at any one time. – Loss of greater than 50% of communications capability (i.e., offsite communications systems which include the Selective Signaling System, the Essex System and the Local Government Radio System). – Loss of greater than 50% of the ability of the TSC or EOF to function. – Loss of instrumentation indication capability to the extent that an Emergency Action Level cannot be determined to exceed an emergency classification. – Loss of ETS if identified by the plant (Not reportable if identified by NRC) – Loss of commercial telephone system to the extent that required communications could not be made to official offsite locations (e.g., EOCs, Warning Points)
INTERNAL THREAT TO PLANT SAFETY (FIRES, TOXIC GAS, RADIOLOGICAL RELEASE) 10 CFR 50.72(b)(1)(vi)	Fire Toxic Explosive Release Personnel Safety	Any event that poses an actual threat to the safety of the nuclear power plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.	– Fire confirmed inside Protected Area (if fire poses an actual threat to plant safety or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant). – Unplanned release of radioactive gases or toxic gas inside Protected Area (if release significantly hampered site personnel in the performance of duties necessary for safe operation of the plant).

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

HBRSEP shall immediately notify the NRC Operations Center via ETS as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SAFETY LIMIT, LIMITING SAFETY SYSTEM SETTING EXCEEDED 10 CFR 50.36(c)(1)(i)(A)	Safety Limit Limiting Safety System Setting	If any safety limit is exceeded, shut down the reactor. HBRSEP shall notify the [NRC within 1 hour via ETS per 10 CFR 50.72(a)(1), See Emergency Plan Procedures]. Operation must not be resumed until authorized by the NRC.	– Reactor pressure exceeds 2735 psig while at power – The limits of TS Table 2.1.1-1 are exceeded – Limiting Safety System Settings in TS Table 3.3.1-1 are exceeded
SAFETY SYSTEM DOES NOT FUNCTION AS REQUIRED 10 CFR 50.36(c)(1)(ii)(A)	ESF RPS Limiting Safety System Setting	HBRSEP shall notify the NRC if the automatic safety system [to correct an abnormal situation before a safety limit is exceeded] has been determined not to function as required.	– A failure mechanism is discovered that indicates that the RPS will not function to trip the reactor under certain required conditions.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the ETS within one hour after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
THEFT/UNLAWFUL DIVERSION OF SNM OR SPENT FUEL SHIPMENT 10 CFR 73.71(a)(1)	SNM Spent Fuel Security Safeguards	Any discovery of the loss of any shipment of SNM or spent fuel, and within one hour after recovery of or accounting for such lost shipment	– Shipment Emergency Event
THEFT/UNLAWFUL DIVERSION OF SNM 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(1)	Theft of SNM Diversion Security Safeguards	Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause: (1) A theft or unlawful diversion of SNM	– Shipment Emergency Event
SABOTAGE OF PLANT EQUIPMENT 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(2)	Sabotage Damage to Plant SNM Spent Fuel Security Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (2) Significant physical damage to a power reactor...or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent fuel a facility or carrier possesses.	– Shipment Emergency Event – Security Event (Reference 2.11)

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the ETS within one hour after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
UNAUTHORIZED TAMPERING WITH PLANT EQUIPMENT 10 CFR 73, Appendix G, I(a)(3)	Unauthorized Use Tampering Security System Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (3) Interruption of normal operation of HBRSEP through the unauthorized use of or tampering with its machinery, components, or controls including the security system.	– Security Event (Reference 2.11)
ENTRY OF UNAUTHORIZED PERSON INTO PROTECTED OR VITAL AREA 10 CFR 73, Appendix G, I(b)	Unauthorized Entry Security Safeguards	An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.	– Security Event (Reference 2.11)
FAILURE, DEGRADATION, OR DISCOVERED VULNERABILITY OF SAFEGUARD SYSTEM 10 CFR 73, Appendix G, I(c) Procedure SEC-NGGC-2147	Degradation Vulnerability Safeguards Unauthorized Undetected Access Security	Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area or transport for which compensatory measures have not been employed.	
INTRODUCTION OF CONTRABAND INTO VITAL OR PROTECTED AREA 10 CFR 73, Appendix G, I(d)	Contraband Unauthorized Security Safeguards	The actual or attempted introduction of contraband into a protected area, material process area, vital area, or transport.	Contraband applies to items that could be used to commit radiological sabotage as defined in 10 CFR 73.2.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM

HBRSEP shall immediately notify the NRC Operations Center via ETS, when:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OR THEFT OF LICENSED MATERIAL (>1000X 10 CFR 20 LIMITS) 10 CFR 20.2201(a)(i)	Loss Theft Missing Licensed Radioactive Material	Immediately notify the NRC, after its occurrence becomes known, any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in [10 CFR 20] Appendix C under such circumstances that it appears to HBRSEP that an exposure could result to persons in unrestricted areas.	– A radiography source is discovered missing. The source is licensed to the radiography contractor. If the contractor does not make the required notification, HBRSEP should notify the NRC Operations Center via ETS.
EXTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, OR SNM (5X ANNUAL LIMIT) 10 CFR 20.2202(a)(1)	Byproduct Source SNM Exposure Dose Release Occupational	Notwithstanding any other requirements for notification, immediately notify the NRC of any event involving byproduct, source, or SNM possessed by HBRSEP that may have caused or threatens to cause any of the following conditions: 1. An individual to receive: (i) A total effective dose equivalent of 25 rems or more; or (ii) An eye dose equivalent of 75 rems or more; or (iii) A shallow dose equivalent to the skin or extremities of 250 rads or more; or 2. The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM

HBRSEP shall immediately notify the NRC Operations Center via ETS, when:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
INTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, SNM (>5X OCCUPATIONAL LIMIT) 10 CFR 20.2201(a)(i)	Intake Ingestion Release Source Byproduct SNM	The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - ISFSI

HBRSEP shall immediately notify the NRC Operations Center via ETS, when:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - ACCIDENTAL CRITICALITY OR LOSS OF SNM 10 CFR 72.74	ISFSI Criticality SNM Loss	The licensee shall notify the NRC Operations Center via ETS within one hour of discovery of accidental criticality or any loss of SNM.	– Unusually high radiation readings discovered in the vicinity of the ISFSI that could indicate possibility of a criticality event

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SNM SHIPMENTS

HBRSEP shall notify the NRC Operations Center via the ETS within one hour of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOST OR UNACCOUNTED SHIPMENT OF SNM 10 CFR 70.52(b) 10 CFR 73.71(a)(1)	Shipment Loss SNM Spent Fuel Theft Diversion Safeguards Security	HBRSEP shall notify the NRC Operations Center via the ETS within one hour after discovery of any loss of any shipment of SNM or spent fuel or any incident in which an attempt has been made, or is believed to have been made, to commit a theft or unlawful diversion of SNM.	– Shipment Emergency Event – Security Event (Reference 2.11)
LOST OR UNACCOUNTED SHIPMENT OF SNM - RECOVERY 10 CFR 73.71(a)(1)	Recovery Accounting Shipment SNM Security Safeguards	HBRSEP shall notify the NRC Operations Center via the ETS within one hour after recovery of, or accounting for, any lost shipment of SNM.	

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FOLLOW-UP

With respect to the telephone notifications made under paragraphs (a) and (b) of 10 CFR 50.72, in addition to making the required initial notification, HBRSEP shall during the course of the event immediately report:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(1)	Degradation Emergency Class Change Update Termination	(i) any further degradation in the level of safety of the plant or other worsening plant conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.	– Refer to Reference 2.27
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(2)	Result Evaluation Effectiveness Unknown	(i) the results of ensuing evaluations or assessments of plant conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant behavior that is not understood.	
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(3)	Open Continuous Communication	Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.	– Refer to Reference 2.27

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS - NRC REGION II OFFICE

HBRSEP shall immediately notify the final delivery carrier and, by telephone and telegram, mailgram, or facsimile, the NRC Region II Office when:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
THEFT/UNLAWFUL DIVERSION OF TRITIUM 10 CFR 30.55(c)	Incident Theft Tritium Attempt Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft of more than 10 curies of tritium (outside of spent fuel) at any one time or more than 100 curies of tritium in one calendar year.	– 10 Curies of tritium discovered missing from the Chemistry Laboratory, and reason exists to suspect that the tritium was stolen
THEFT/UNLAWFUL DIVERSION OF SOURCE MATERIAL 10 CFR 40.64(c)	Incident Attempt Theft Diversion Source Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of more than 15 pounds of Source Material at any one time or 150 pounds of Source Material in any one calendar year.	– A source assembly is discovered missing from a new fuel shipment.
SHIPPING PACKAGE RADIOACTIVELY CONTAMINATED 10 CFR 20.1906(d)(1)	Contamination Shipment	Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87;	– New or Spent Fuel Shipment Cask arrives with surface contamination in excess of limits.
SHIPPING PACKAGE EXCEEDING EXTERNAL DOSE RATE LIMITS 10 CFR 20.1906(d)(2)	Radiation Dose Rate Shipment	External radiation levels exceeds of the limits of 10 CFR 71.47.	– New or Spent Fuel Shipment Cask arrives with external radiation levels in excess of limits.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FFD			
The NRC Region II Administrator must be notified immediately by telephone of the following:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
NRC EMPLOYEE NOT FIT FOR DUTY 10 CFR 26.27(d)	Alcohol Influence Substance NRC employee FFD Fitness for Duty	If HBRSEP has a reasonable belief that an NRC employee may be under the influence of any substance, or unfit for duty...the Region II Administrator must be notified immediately by telephone. During other than normal working hours, the NRC Operations Center via ETS must be notified.	
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FFD			
The NRC Operations Center via ETS must be notified immediately by telephone of the following:			
FALSE POSITIVE ERROR ON FFD SPECIMEN 10 CFR 26, Appendix A, Subpart B, 2.8(e)(5)	FFD Fitness for Duty False Positive Specimen Laboratory	Should a false positive error occur on a blind performance test specimen and the error is determined to be an administrative error, HBRSEP shall promptly notify the NRC.	
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - IAEA			
The NRC Director, NRR or Director, NMSS must be notified immediately by telephone of the following:			
SURPRISE VISIT OF IAEA OFFICIAL 10 CFR 75.7	IAEA International Atomic Energy Agency Credential	HBRSEP shall immediately communicate by telephone, with respect to the credentials of any other person who claims to be an IAEA representative and shall accept telephone confirmation of such credentials by the Commission.	– Person arrives on site bearing IAEA credentials, who is not accompanied by an NRC employee, and has had no prior confirmation in writing of credentials.

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
DEGRADED SAFETY BARRIERS DISCOVERED WHILE SHUT DOWN 10 CFR 50.72(b)(2)(i)	Shutdown Safety Barrier Fission Product Barriers Degrade Unanalyzed	Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.	– Corrosion of Reactor Coolant System piping found while shutdown (indicative of a material problem that caused abnormal degradation of the RCS pressure boundary). – Significant degradation of Reactor Fuel Rod Cladding identified during testing of fuel assemblies (Reference 2.19).

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ESF OR RPS INITIATION (MANUAL/AUTOMATIC) 10 CFR 50.72(b)(2)(ii)	Manual Automatic Actuation Engineered Safety Feature ESF Valid Clearance Ventilation System Reactor Protection System RPS Reactor Trip	Any event or condition that results in a manual or automatic actuation of any ESF, including the RPS, except when: (A) The actuation results from and is part of a pre-planned sequence during testing or reactor operation; (B) The actuation is invalid and: (1) Occurs while the system is properly removed from service; (2) Occurs after the safety function has been already completed; or (3) Involves only the following specific ESFs or their equivalent systems: (i) Not Applicable (ii) Control Room emergency ventilation system; (iii) Reactor building ventilation system; (iv) Fuel building ventilation system; or (v) Auxiliary building ventilation system.	– Safety Injection System actuation (also see Emergency Plan Procedures) – Reactor Trip (Manual or Automatic). – EDG start due to a valid undervoltage trip signal on emergency bus E1 or E2 – A single train of Containment Isolation actuates. – A valid signal for Containment Ventilation Isolation occurs. ---------- All ESF actuations are reportable except the following three categories. 1) An invalid ESF or RPS actuation occurs when the system is already properly removed from service if all requirements of plant procedures for removing equipment from service have been met. This includes required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers. 2) An invalid ESF or RPS actuation occurs after the safety function has already been completed (e.g., an invalid containment isolation signal while the containment isolation valves are already closed, or an invalid actuation of the RPS when all rods are fully inserted). 3) ESF actuations that are caused by non-ESF systems may be excluded because these are not considered ESF actuations of safety significance. (Reference 2.19) 4) Invalid actuations of the listed ventilation systems.

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
CONDITION THAT COULD PREVENT FULFILLMENT OF SAFETY FUNCTIONS 10 CFR 50.72(b)(2)(iii)	Loss of Safety Function Residual Heat Mitigation Shutdown Generic Setpoint Drift Engineering Evaluation Operability Determination Common Mode Failure	Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition. (B) Remove residual heat, (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident.	– Loss (inoperability) of both Trains, e.g., ECCS, Low Temperature Overpressure Protection System, or Lake Robinson water level below LCO 3.7.8 limit. 1) Overpressurization of the RCS (if Overpressure Protection System fails to perform its intended function) - Loss of one Train of required equipment, and the cause of the failure could fail the other train, and there is a reasonable expectation that the other train would not fulfill its safety function if required. 1) Contaminated lubrication fluid degrades SI Pump operation (a single condition could prevent fulfillment of a safety function if both trains could be reasonably expected to be inoperable). 2) EDG Air Start Solenoids (if it demonstrates a design, procedural, or equipment deficiency that could prevent the fulfillment of a safety function, i.e., if both diesels are susceptible to same problem) - Multiple equipment inoperability or unavailability. 1) Generic setpoint drift (if indicative of a generic and/or repetitive problem with switches used in safety systems) 2) Oversized breaker wiring lugs (incompatible pigtails and lugs could cause one or more safety systems to fail to perform their intended functions) – Control Rod failure (if failure prevented the fulfillment of a safety function) – Operator action to inhibit the RPS (actions would prevent fulfillment of a safety function)

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
AIRBORNE RELEASE TO UNRESTRICTED AREA (>20X 10 CFR 20 LIMITS) 10 CFR 50.72(b)(2)(iv)(A)	Airborne Release Unrestricted Public Radioactive Effluent	Any airborne radioactive release that, when averaged over a time period of 1 hour, results in concentrations in unrestricted area that exceeds 20 times the applicable concentration specified in Appendix B to 10 CFR 20, Table 2, Column 1.	– Unplanned gaseous release (if release exceeded 20 times the applicable concentrations specified in Appendix B, Table 2, Column 1 of 10 CFR 20 averaged over a time period of one hour)
LIQUID EFFLUENT RELEASE TO UNRESTRICTED AREA (>20X 10 CFR 20 LIMITS) 10 CFR 50.72(b)(2)(iv)(B)	Liquid Release Unrestricted Public Radioactive Effluent Concentration Discharge	Any liquid effluent release that, when averaged over a time period of 1 hour, exceeds 20 times the applicable concentration specified in Appendix B to 10 CFR 20 , Table 2, Column 2, at the point of entry into the receiving waters (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases.	– Radioactive release exceeding TS (if release exceeds 20 times the applicable limit of Appendix B, Table 2, Column 2 of 10 CFR 20 when averaged over one hour)
TRANSPORT OF CONTAMINATED INJURED PATIENT 10 CFR 50.72(b)(2)(v)	Contaminate Injured Person Medical Transport Rescue Hospital	Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment.	– Any event requiring the transport of a radioactively contaminated or potentially contaminated (Reference 2.19) person to an off-site medical facility for treatment

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRESS RELEASES AND GOVERNMENT NOTIFICATIONS 10 CFR 50.72(b)(2)(vi)	News Release Press Radio Television Fatality Environment Public Health and Safety Release	Any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials.	– Any News release concerning - A fatality, - Inadvertent release of radioactively contaminated materials to public areas - unusual or abnormal releases of radioactive effluents, or - Information associated with an Emergency Event except when the ERO is activated (Reference 2.27) – Notification to other government agencies concerning - A fatality on site, - Health and safety of the public or site personnel, - Inadvertent release of radioactively contaminated materials to public areas, - Discovered endangered species kill.

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via ETS as soon as possible but not later than 4 hours after the discovery of any of the following events or conditions involving spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - EXPOSURES TO RADIATION OR RADIOACTIVE MATERIALS IN EXCESS OF LIMITS, OR RELEASES IN EXCESS OF LIMITS 10 CFR 72.75(b)(1)	ISFSI Release Exposure Fire Explosion Toxic	Any event that prevents immediate actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits, or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases).	– Explosion or fire involves ISFSI resulting in radiological releases
ISFSI - DEFECT IMPORTANT TO SAFETY 10 CFR 50.72(b)(2)(vii)(A) 10 CFR 72.75(b)(2)	ISFSI Defect Safety	A defect in any spent fuel storage structure, system, or component which is important to safety.	– A defect discovered in the design or construction of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits
ISFSI - REDUCTION IN EFFECTIVENESS 10 CFR 50.72(b)(2)(vii)(B) 10 CFR 72.75(b)(3)	ISFSI Confinement Reduction Effectiveness	A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use.	– Wear or degradation of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits
ISFSI - DEPARTURE FROM LICENSE CONDITION 10 CFR 72.75(b)(4)	ISFSI Emergency Departure Deviation Health and Safety License Condition	An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under 10 CFR 72 when the action is immediately needed to protect the public health and safety and no action consistent with license conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.	– Action taken in an emergency that departs from procedure that is deemed necessary to prevent releases or radiation doses to the public in excess of 10 CFR 20 limits (See PRO-NGGC-0200)

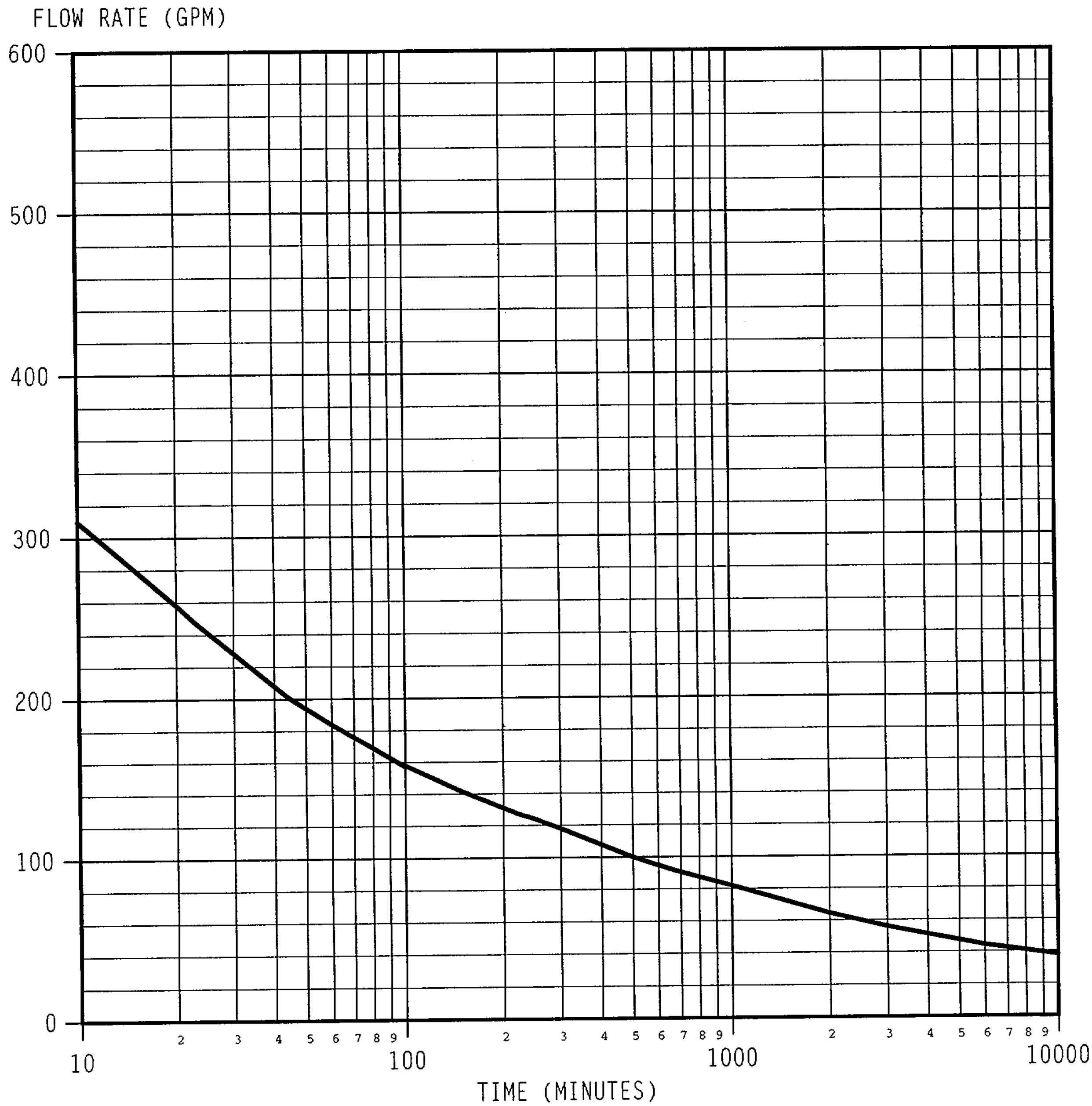
FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

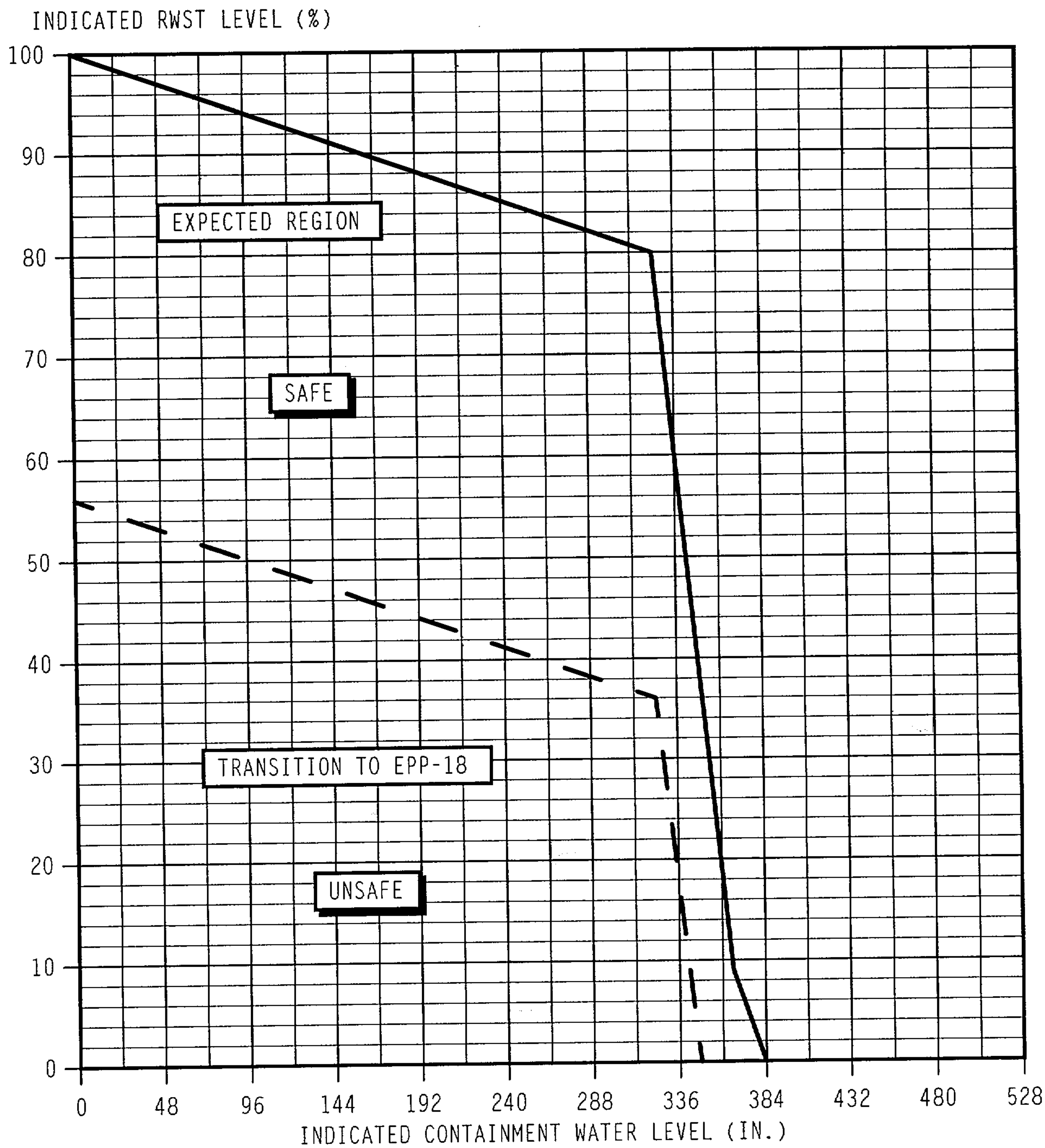
HBRSEP shall notify the NRC Operations Center via ETS as soon as possible but not later than 4 hours after the discovery of any of the following events or conditions involving spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - TREATMENT OF CONTAMINATED PERSON AT OFFSITE MEDICAL FACILITY 10 CFR 72.75(b)(5)	ISFSI Contaminate Injured Person Medical Transport Rescue Hospital	An event that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination.	– An individual is injured requiring offsite medical treatment and receives contamination from ISFSI(s) that cannot be removed prior to transport
ISFSI - FIRE OR EXPLOSION 10 CFR 72.75(b)(6)	ISFSI Fire Explosion Damage Integrity	An unplanned fire or explosion damaging any spent fuel, or any device, container, or equipment containing spent fuel when the damage affects the integrity of the material or its container	– ISFSI unit is damaged by an external explosion and the integrity of the ISFSI unit is potentially affected

ATTACHMENT 1
REQUIRED FLOW RATE VERSUS TIME AFTER REACTOR TRIP
Page 1 of 1



ATTACHMENT 1
CONTAINMENT SUMP LEVEL VS. RWST LEVEL
Page 1 of 1



REACTOR POWER ASCENSION INDICATOR LOG

AVG PWR % (1)	NI-35 amps	NI-36 amps	NI-41A %	NI-42A %	NI-43A %	NI-44A %	LOOP ΔT °F (1)	LOOP 1 ΔT °F	LOOP 2 ΔT °F	LOOP 3 ΔT °F	1st STAGE PRESS psig (1)	PI-446 **OR** 447 psig (2)	NET MWe MAX (1)	NET MWe	CCP % PWR (3)	NR-45 (4)	SSO (1)
15-20							9-11.5				68-90		73				
25-30							14.5-17				113-135		153				
35-40							20-23				158-180		235				
45-50							26-28.5				207-230		316				
55-60							32-34.5				261-285		398				
65-70							37-40				320-345		480				
75-80							43-46				384-410		562				
85-90							49-51.5				449-475		643				
95-100							55-57.5				513-540		725				

(1) Listed ranges and Net MWe maximums are predicted based on past plant performance. The maximum value of each indication is the maximum target value for each power increase. The SSO shall initial if plant management has determined that indications are acceptable to continue with the power escalation.

(2) Use indicator that corresponds to the channel selected on the 1st STAGE PRESSURE selector switch.

(3) Record Continuous Calorimetric Program % Power.

(4) Verify NR-45 is selected to the highest reading channel.

AVAILABLE CONTINGENCY ACTIONS

INFORMATION USE

1.0 Decay Heat Removal:

1) In the case of a loss of the normal decay heat removal equipment while the Residual Heat Removal System is aligned for shutdown cooling, AOP-020 should be followed.

NOTE: In order to supply power to the RHR pump in accordance with the referenced procedure (EPP-025) in the following step, electrical terminations are required.

2) If a loss of station power is the cause of the loss of normal decay heat removal equipment, the backup diesel power that is required by OMP-003 should be placed in service automatically or, manually if necessary, by the normal operating procedures listed on the appropriate attachment to this procedure. (OP's 601,603,604). If the normal diesel backup power is not available, or fails to operate, then the contingency actions necessary to provide alternate power to the decay heat removal equipment provided in EPP-025 should be performed, and heat removal capability restored.

3) In the event that the Reactor is completely defueled and the normal supplies of cooling water to the SFP heat exchanger are lost, the engine driven fire pump in conjunction with the alignment of the fire water system to the SFP heat exchanger, will provide an available backup to all other supply pumps that are powered from the onsite or offsite power supplies in the event that all onsite and offsite power is lost

4) The steps necessary to connect the fire water system to the SFP heat exchanger as a temporary cooling water supply can be found in OP-306, Component Cooling Water System, Section 8.3 Spent Fuel Pit Heat Exchanger Emergency Cooling.

5) Spent Fuel Pit Cooling Pump "A" is powered form 480v Bus No.3, and 480v Bus No. 3 may be powered from the Dedicated Shutdown Diesel Generator (DSDG) via the Dedicated Shutdown (DS) Bus in the event that offsite and onsite backup power is lost. Alignment of the DS Bus to 480v Bus No. 3 is contained in EPP-025.

AVAILABLE CONTINGENCY ACTIONS

NOTE: This attachment provides the available contingency actions for the operations personnel to restore the "Shutdown Safety Functions" under conditions of either fuel in the Containment or with the Reactor completely "defueled".

2.0 Electrical Power:

1) IF the normal 115KV switch yard supply to the Start Up Transformer has been lost due to relay action, and the normal "Backfeed" method is not available (downstream equipment unavailable). The dispatcher should be contacted to determine if switching instructions may be issued to reenergize one section of the 115KV bus. This section of 115KV bus may then supply the Start Up Transformer, via the auto transformer, from the 230KV switch yard. Under these conditions the fault that caused the original relay action must be verified not to be on the section of 115KV bus to be used.

NOTE: In order to supply power to the RHR pump in accordance with the referenced procedure (EPP-025) in the following step, electrical terminations are required.

2) With fuel in the Reactor, or with the Reactor completely defueled, the contingency actions associated with EPP-025 will supply power to the minimum equipment necessary to maintain the Decay Heat Removal, and Inventory Control "Shutdown Safety Functions"

AVAILABLE CONTINGENCY ACTIONS

3.0 Inventory Control:

1) Normal inventory maintenance is controlled by the Operating Procedures. In the event that excessive leakage occurs with the refueling cavity full, AOP-020 - Loss of Residual Heat Removal (Shutdown Cooling), should be followed to isolate the leak, establish makeup to the cavity at the maximum available rate, and place the RHR system in the recirculation mode if the leakage cannot be isolated and the CV sump level rises to the minimum required to operate the RHR pumps in the recirculation mode.

2) In the event that leakage from the Spent Fuel Pit occurs while the Reactor has been offloaded to the Spent Fuel Pit, OP-910 - Spent Fuel Pit Cooling and Purification System, or OP-913, Refueling Water Purification Pump Operation, are used to initiate make up to the spent fuel pit.

3) The following Procedures are also available to establish alternative means to make-up to the Spent Fuel Pit:

a. OP-301 Chemical And Volume Control System, may be used to initiate blended make-up to the RWST, and OP-913, Refueling Water Purification Pump Operation used to subsequently make-up to the SFP.

CAUTION

The flow path aligned in the following step is non-borated water and may lead to a dilution accident in the SFP if used to make up for a large loss of SFP inventory.

b. The demineralized water system may be connected directly to the SFP clean up loop for make-up through the valves listed below:

- DW-215 - DEMINERALIZED WATER TO PLANT COMPONENTS
- SFPC-808 - DEMIN WATER INLET

AVAILABLE CONTINGENCY ACTIONS

4.0 Reactivity Control:

1) Borated makeup sources, and all components necessary to inject the borated water are required to be operable in accordance with OMP-003 when fuel is in the vessel. Other means of borated makeup when the RCS is intact include the flow path through the RCP seals, however this should only be used as a last resort. Normal letdown if available when fuel is in the vessel, may be used to divert displaced inventory to the CVCS Hold Up Tank (HUT). As an alternate means of increasing the Boron Concentration in the Refueling cavity when the vessel head has been removed, 100 lb. bags of Granulated Boric Acid may be added to the cavity. One 100 lb. bag of Granulated Boric Acid will increase the Cavity Boron Concentration approximately 6 ppm. Contact the Reactor Engineer to provide guidance IAW the Reactivity Management Program. (SOER 94-2)

2) When the core is offloaded to the SFP, borated make-up is available from the RWST in accordance with the procedure listed on Attachment 10.2 of this procedure, however if the SFP is at the full level and no more inventory can be added, Boron Concentration may be increased by adding Granulated Boric Acid to SFP locally. One 100 lb. bag of Granulated Boric Acid will increase the Boron Concentration of the SFP approximately 6 ppm. Contact the Reactor Engineer to provide guidance IAW the Reactivity Management Program. (SOER 94-2)

AVAILABLE CONTINGENCY ACTIONS

5.0 Containment:

1) Containment Closure is controlled by the Improved Technical Specifications and plant procedures based on the current plant status, the procedures listed below are intended to maintain the applicable degree of isolation at the plant conditions indicated in the procedures and are either successful or are performed until the proper degree of isolation is achieved, therefore there are no contingency actions applicable that are not contained in the controlling procedures:

   a. GP-002 - COLD SOLID TO HOT SUBCRITICAL AT NO LOAD T_{-avg}
   (establishes "Containment Integrity" in accordance with OP-923 when RCS is at 200°F)

   b. OMM-033 - IMPLEMENTATION OF CV CLOSURE
   (controls closure of CV penetrations when RCS temperature is less than 200°F)

   c. GP-010 - REFUELING
   (establishes "Containment Closure" for refueling when the Reactor Vessel head is removed and core components are being moved)

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

1.0 INTRODUCTION

This document identifies the risk impact of various combinations of equipment safety functions being unavailable due to maintenance during reactor critical and power operation ("on-line maintenance"). The risk impact measure for this analysis is core damage frequency (CDF), as calculated using the current probabilistic safety analysis (PSA) model of the Robinson plant. While this analysis provides risk insights that can be obtained in no other way, it is intended that the information contained in this document be used in conjunction with design basis information, operational experience, and engineering judgment to determine the extent and scope of any planned on-line maintenance activity. Because the PSA only measures risk impact, and not defense in depth, the allowed out of service times presented in this document may be different than those of the plant's technical specifications. This document shall not be used as a basis for extending a Tech. Spec. Action Statement but should be observed when the recommended limits of this document are more restrictive than the limits imposed by technical specifications.

2.0 METHODOLOGY

2.1 Determination of Train Combinations for On-line Maintenance

Systems identified as safety significant by the maintenance rule expert panel were evaluated for on-line maintenance impact on core damage risk. These systems were broken down into two major trains and separated on the 12-week on-line schedule. This schedule was used to determine the presentation of results.

Note that the 12-week on-line schedule contains some systems or trains that are maintained on-line but whose function is not impaired by the maintenance action. These systems were not included in the PSA analysis, since the PSA considers the impact of unavailable functions when determining risk impact. However, some maintenance actions, even if they do not render the system incapable of performing its accident mitigation function, may increase the likelihood of a transient or other initiating events. Systems or trains in this category were included in the analysis.

2.2 Calculation of Core Damage Frequencies

In order to determine the risk impact of planned maintenance, a "baseline" core damage frequency was required. This baseline core damage frequency served as the basis for determining whether the calculated risk increase for a given equipment configuration was safety significant or non-safety significant. This baseline CDF was determined by setting all unavailability events in the PSA model to the "in service" value of zero. The model was then quantified to obtain the baseline CDF.

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

In order to assess the relative impact of performing on-line maintenance on a single system or a pair of systems, the system train function was made unavailable. All other system functions in the PSA model except those being taken out for on-line maintenance were made available. The PSA model was then solved for this combination of equipment out of service to determine the new core damage frequency for that condition.

If maintenance could increase the likelihood of a transient or other initiating event, this impact had to be considered in the analysis. This was addressed by assuming that the maintenance would cause the appropriate initiating event in the model to increase in frequency by a factor of ten. An example of this "environmental event" would be work on Reactor Protection Logic. While planned logic testing at power would not remove the reactor trip function, the likelihood of a reactor trip initiating event is considered greater than during periods when no testing is conducted. Another example is switchyard work. Switchyard work is not considered safety significant in itself and is not included in the matrices. However, switchyard work in combination with EDG or AFW steam driven pump maintenance is a higher risk impact evolution due to the increased potential for a station blackout, and should be avoided.

2.3 *Determination of Significant Risk Increase due to On-line Maintenance*

There are several criteria for determining whether a given risk increase is safety significant or non-safety significant. The criteria utilized in this analysis were based on the EPRI PSA Applications Guide. Three thresholds for safety significance were applied in the present analysis:

- The instantaneous value of CDF calculated for the given condition should not be above 1E-3 per year.
- The change in core damage probability for the condition, which is the product of the instantaneous CDF increase (over the baseline) for the given condition and the length of time the condition would exist, should not be allowed to exceed 1E-6 without consideration of additional, non-quantifiable factors.
- The change in core damage probability for the condition may exceed 1E-6 provided: 1) the change in core damage probability does not exceed 1E-5; 2) additional, non-quantifiable factors (possibly including contingency measures) are considered; 3) an appropriate level of management approval is obtained.

These three thresholds were applied, using the calculated CDF for each combination of equipment functions unavailable, the baseline CDF with no equipment in test or maintenance, and an assumed equipment unavailable time of 72 hours.

3.0 RESULTS

3.1 Assumptions and Considerations

This analysis does not consider all Safety Significant Systems identified by the maintenance rule expert panel, but rather is limited to those systems whose maintenance activities may contribute to core damage through unavailability of system train functions. Some electrical systems whose functions are not made unavailable while on-line are not included in the list of system train functions. These systems are discussed in Section 3.2 and in the Notes on Table 1.

The EPRI PSA applications guide recommends an evaluation of Large Early Release Frequency (LERF) for applications. A review of the level 2 (containment performance) PSA analysis reveals that functional failures of containment safeguards systems (containment isolation, containment spray, containment fan coolers) do not significantly contribute to the potential for large early releases from sever accidents. LERF scenarios are dominated by interfacing-system LOCAs (RHR-750/751) and steam generator tube ruptures, which by nature create a release path. The status of the containment safeguards systems have little impact on large early releases, and would not be considered Safety Significant based on their limited impact on the PSA results.

Since the on-line maintenance matrix was quantified with core damage as the end-state, containment systems were not included on the matrix. However, if consideration is given to potential performance degradation of containment isolation, the frequency of large early releases would increase. Therefore, maintenance activities that render a containment isolation valve open (non-isolatable) or that compromise Main Steam isolation via the SRVs, PORVs or MSIVs should not be done while any core-damage mitigating system function listed in Table 1 is unavailable.

While instantaneous CDF and increase in core damage probability (delta CDF * time out of service) were considered, the cumulative safety impact associated with on-line maintenance activity over the entire cycle was not included. The impact of maintenance activity on initiating event frequencies, where applicable, was assumed to be an increase by a factor of ten.

As stated in the introduction, this analysis is intended to be used in conjunction with design basis information, operational experience, and engineering judgment to determine the extent and scope of any planned on-line maintenance activity.

3.2 *Presentation of Results*

The results of the analysis are presented in matrix format to facilitate determination of the safety significance of system train functions being unavailable. Because of the amount of information resulting from this study, a number of different views of the results are presented. The matrices and other information are contained in Tables 1 and 2.

Table 1 lists the maintenance events that were analyzed. The table lists the maintenance event description, the system or train accident mitigation function, and the assumed impact on initiating events, if applicable. The table details the safety function to be maintained for combinations that are considered safety significant. A number of power systems, that will not have planned maintenance out of service time, have been removed from the matrix: 4KV AC (5170), 480V AC (5175 non-safety related), 208/120V AC (5185), and transformers and switchyard (5120). However, testing of these systems may introduce a higher probability of an undervoltage initiator. Therefore, work on the AFW steam driven pump and the EDGs should not be performed in conjunction with maintenance or test on these systems due to the increased potential for a station blackout.

Table 2 is a matrix which shows the number of hours that a combination of equipment can be unavailable before the change in core damage probability (delta CDF * time) would exceed 1E-6. Note that the 1E-6 core damage probability threshold is only one part of the analysis. Cells marked with an X are not recommended because the instantaneous CDF would exceed the 1E-3 threshold.

It is made up of three separate matrices: One for train A equipment, one for train B equipment, and one for "cross-train" equipment. These matrices list the maintenance events across the top and down the left side

Maintenance that exceeds the allowed hours in Table 2 will place the plant in a potentially High Risk Impact configuration and is not recommended. Planning maintenance to exceed the hours in Table 2 should be accompanied with plant general manager approval per Attachment 10.4 and a review of non-quantifiable factors (e.g. reason maintenance is necessary on-line). Any maintenance, planned or emergent, which exceeds the hours in Table 2 should be accompanied with risk impact insights from PSA, and development of contingency plans.

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

3.3 Use of the On-line Maintenance Matrices

The matrices apply only for reactor critical and power operation (all trips and actuation signals in place) and only for combinations of one or two system trains at a time. If three or more system train functions need to be unavailable at the same time, then further analysis needs to be performed.

The matrices only address best estimate risk impact, and not defense in depth. The most limiting configurations must be determined through a combination of Technical Specifications, the matrix, and other design basis documents.

When using these results to determine appropriate on-line maintenance, it is important to remember that all functions listed in Table 1, which are not designated as unavailable, are assumed to be functional. The scope of this application assumes that equipment must be available to provide its safety function. If the system, structure or component (SSC) is in service providing the safety function, some components may be defeated such that the ability to maintain the function is not degraded. Existing plant procedures shall be used to determine the availability of an SSC.

In case of emergent equipment unavailability, a review of the equipment functions already unavailable must be performed. Potential high risk impact situations need to be identified and non-quantifiable factors and contingency plans must be identified. An example of a non-quantifiable factor would be the need to shutdown the plant if the repair is not expedited. Plant shutdowns introduce additional risk through challenging safety systems which in itself is not quantifiable. The potential High Risk Impact configurations need to be avoided or limited in duration as much as practical. It is not recommended to intentionally enter what are potentially high risk impact configurations.

3.4 *Matrix Limitations*

Not all high safety significant SSC are included directly in the matrices. Any maintenance activities or emergent conditions that could degrade any of these safety functions should be evaluated against other equipment that is unavailable. These SSCs fall in the following categories:

- Normally passive high safety significant SSCs:
  - Reactor coolant system boundary
  - Containment structure
- SSCs for which on-line maintenance or unavailability is not expected:
  - Pressurizer safety valves
  - Steam generator safety valves
  - Safety injection accumulators
  - Main steam isolation valves
  - Feedwater isolation valves
  - Station batteries

Note: The unavailability of these SSCs is controlled through short duration Tech Specs. Restoration of the unavailable function should be a top priority.

- SSCs that support containment integrity and environmental control
  - Containment spray
  - Service water booster pumps
  - Containment cooling

Note: When performing maintenance or removing these components from service, a qualitative assessment addressing the remaining defense in depth should be performed.

- Support systems
  - Diesel fuel oil
  - Nitrogen supply to PORVS
  - Auxiliary building HVAC

Note: Maintenance activities and unavailability of these systems should be evaluated for the impact on the supported front-line system.

- Other
  - Control room emergency filtration and pressurization

Note: Maintenance activities and unavailability or these components are adequately controlled through Tech Spec adherence.

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Table 1. Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
1080	A	RPS Channel A Logic In Test & Maintenance (Includes RX Trip Bkr 1065, And Safeguards Train A)	RPS CHANNEL A	Prevent Inadvertent RX Trip, Provide RX Trip and Safeguards Actuation Logic on Valid Transient. (See Note 2)	Conservatively assumed train A SI actuation signal Fails, Increase frequency of ATWS and RX Trip.
2005	A	RCS PZR PORV Train A Unavailable (RC-456, N2 Header, Block Valve RC-535)	RCS PZR PORV 456	Provide a Bleed Path for Feed and Bleed Cooling, and Maintain RCS Integrity.	Assumes PORV or Block Valve will not open. Stuck open Block Valve not analyzed. See Block Valve entry under train B.
2045	A	RHR Train A Unavailable	RHR PUMP A	Provide RCS Inventory Control and Decay Heat Removal	
2060	A	CVCS Charging Pump B Unavailable (Train A)	CVCS CHGP B	Provide RCP Seal Injection. If removal of function is permitted, prevent total loss of CVCS which could lead to a plant trip.	Increased frequency of total loss of CVCS initiator not included.
2080	A	SI Pump A Unavailable (Train A)	SI PUMP A	Provide RCS Inventory Control	Pump B can be swapped to the A train to maintain function.
3020	A	S/G A PORV RV-1 Unavailable (Includes Specific IA Support Manifold)	S/G A PORV RV-1	Provide Ability for Cooldown From Hot to Cold Shutdown	Open function failed: Results conservative by allowing reclose failures in cutsets.
3020	A	S/G B PORV RV-2 Unavailable (Includes Specific IA Support Manifold)	S/G B PORV RV-2	Provide Ability for Cooldown From Hot to Cold Shutdown	Open function failed: Results conservative by allowing reclose failures in cutsets.
3020	A	S/G C PORV RV-3 Unavailable (Includes Specific IA Support Manifold)	S/G C PORV RV-3	Provide Ability for Cooldown From Hot to Cold Shutdown	Open function failed: Results conservative by allowing reclose failures in cutsets.
3050	A	MFW Pump Train A Unavailable	MFWP A	Prevent Loss Causing Plant Trip, and mitigate ATWS or Loss of AFW	Assumes A train MFW or CND pumps are unavailable and increased frequency of Total Loss of MFW initiator.
3065	A	AFW MD Pump Train A Unavailable (Includes Actuation Channel)	AFW MDP A	Automatically Deliver Condensate From the CST to the S/Gs Following a Plant Trip (See Note 3)	

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 1 .Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
3065	A	AFW SD Pump Train Unavailable	AFW SDP	Automatically Deliver Condensate From the CST to the S/Gs Following a Plant Trip (See Notes 3 and 4)	Pump is unavailable when 2 or 3 S/Gs are unavailable to supply steam or receive flow via MS-V1-8A,B,C or AFW-V2-14A,B,C
4060	A	SW Pump A Unavailable	SW PUMP A	Provide Cooling for Safety Related Equipment, Prevent Loss of SW Initiator	
4060	A	SW Pump B Unavailable	SW PUMP B	Provide Cooling for Safety Related Equipment, Prevent Loss of SW Initiator	
4080	A	CCW Pump A Unavailable (Train DS)	CCW PUMP A	Provide Cooling for Safety Related Equipment, Prevent Loss of CCW Initiator	
4080	A	CCW Pump B Unavailable (Train A)	CCW PUMP B	Provide Cooling for Safety Related Equipment, Prevent Loss of CCW Initiator	
5095	A	EDG A Unavailable (Includes Room Cooling 8210, And Fuel Oil 5100)	EDG A	Provide Power to the Emergency Bus (See Note 4)	
5175	A	480V Emergency Bus E1 In Test Or Maintenance, Assumed Available	EMERGENCY BUS E1	Prevent Bus Undervoltage Initiator and Provide Power to Emergency Bus and Safety Related MCC Loads	Assumes increased frequency of Loss of Emergency Bus E1 Initiator.
5235	A	DC, One Train A Battery Charger Unavailable	DC BAT CHG A/A1	DC Bus, supplied by a Battery Charger, Must be Available to Provide Control Power.	Assumes increased frequency of Loss of DC Bus A Initiator.
6135	A	Air Compressor A Unavailable	AIR COMP A	Prevent Loss of Instrument Air, and Provide Instrument Air to S/G PORVs and CVCS	Increased frequency of loss of Instrument Air initiator not included.
6135	A	Air, Primary Air Compressor Unavailable	AIR COMP PRIM	Prevent Loss of Instrument Air, and Provide Instrument Air to S/G PORVs and CVCS	Increased frequency of loss of Instrument Air Initiator not included.
6175	A	Fire Pump, Engine Driven Unavailable	FIRE PUMP DIESEL	Provide Alternate Cooling to SI, AFW, and Charging Pumps	
6270	A	Deepwell Pump B Unavailable	DEEPWELL PUMP B	Provide Makeup to CST or Alternate AFW Supply	

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 1 .Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
1080	B	RPS Channel B Logic In Test & Maintenance (Includes RX Trip Bkr 1065, And Safeguards Train B)	RPS CHANNEL B	Prevent Inadvertent RX Trip, Provide RX Trip and Safeguards Actuation Logic on Valid Transient. (See Note 2)	Conservatively assumed train B SI actuation signal fails, Increase frequency of ATWS and RX trip.
2005	B	RCS PZR PORV Train B Unavailable (RC-455C, N2 Header, Block Valve RC-536)	RCS PZR PORV 455C	Provide a Bleed Path for Feed and Bleed Cooling, and Maintain RCS Integrity Given a Stuck Open PORV.	Assumes PORV or Block Valve will not open. Stuck open Block Valve not analyzed.
2005	B	Both RCS PZR Block Valves Closed But Available	RCS BLOCK VALVES	Provide at Least One Path to Mitigate a Pressure Challenge	Assumes PORV is operable when Block Valve is open. See Block Valve entry below.
2045	B	RHR Train B Unavailable	RHR PUMP B	Provide RCS Inventory Control and Decay Heat Removal	
2060	B	CVCS Charging Pump A Unavailable (Train DS)	CVCS CHGP A	Provide RCP Seal Injection. If removal of function is permitted, prevent total loss of CVCS which could lead to a plant trip.	
2060	B	CVCS Charging Pump C Unavailable (Train B)	CVCS CHGP C	Provide RCP Seal Injection. If removal of function is permitted, prevent total loss of CVCS which could lead to a plant trip.	
2080	B	SI Pump C Unavailable (Train B)	SI PUMP C	Provide RCS Inventory Control	Pump B can be swapped to the B train to maintain function.
3050	B	MFW Pump Train B Unavailable	MFWP B	Prevent Loss Causing Plant Trip, and mitigate ATWS or Loss of AFW	Assumes B train MFW or CND pumps are unavailable and increased frequency of total loss of MFW initiator.
3065	B	AFW MD Pump Train B Unavailable (Includes Actuation Channel)	AFW MDP B	Automatically Deliver Condensate From the CST to the S/Gs Following a Plant Trip	
4060	B	SW Pump C Unavailable	SW PUMP C	Provide Cooling for Safety Related Equipment, Prevent Loss of SW Initiator	

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Table 1 .Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
4060	B	SW Pump D Unavailable	SW PUMP D	Provide Cooling for Safety Related Equipment, Prevent Loss of SW Initiator	
4080	B	CCW Pump C Unavailable (Train B)	CCW PUMP C	Provide Cooling for Safety Related Equipment, Prevent Loss of CCW Initiator	
5095	B	EDG B Unavailable (Includes Room Cooling 8210, And Fuel Oil 5100)	EDG B	Provide Power to the Emergency Bus (See Note 4)	
5098	B	DSDG (Includes DS Fuel Oil 5100) Unavailable	DSDG	Provide Power to the DS Bus	Taking out the DSDG is not as limiting as taking out the DS Bus.
5114	B	DS Bus Unavailable	DS BUS	Provide Power to Chg Pump A, CCW Pump A (Alternate for SW Pump D, MCC5 and Deepwell Pumps)	Taking out the DS Bus takes out the DSDG, CCWA, CVCS CHGP A and can be considered one function.
5175	B	480V Emergency Bus E2 In Test Or Maintenance, Assumed Available	EMERGENCY BUS E2	Prevent Bus Undervoltage Initiator and Provide Power to Emergency Bus and Safety Related MCC Loads	Assumes increased frequency of Loss of Emergency Bus E2.
5235	B	DC, One Train B Battery Charger Unavailable	DC BAT CHG B/B1	DC Bus, supplied by a Battery Charger, Must be Available to Provide Control Power.	Assumes increased frequency of Loss of DC Bus B Initiator.
6135	B	Air Compressor B Unavailable	AIR COMP B	Prevent Loss of Instrument Air, and Provide Instrument Air to S/G PORVs and CVCS	Increased frequency of loss of Instrument Air Initiator not included.
6135	B	Air Compressor D Unavailable	AIR COMP D	Prevent Loss of Instrument Air, and Provide Instrument Air to S/G PORVs and CVCS	Increased frequency of loss of Instrument Air Initiator not included.

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Table 1 .Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
6175	B	Fire Pump, Motor-Driven Unavailable	FIRE PUMP MOTOR	Provide Alternate Cooling to SI, AFW, and Charging Pumps	
6270	B	Deepwell Pump A Unavailable	DEEPWELL PUMP A	Provide Makeup to CST or Alternate AFW Supply	
6270	B	Deepwell Pump C Unavailable	DEEPWELL PUMP C	Provide Makeup to CST or Alternate AFW Supply	

NOTES:

1. Trains as designated by 12 week on-line schedule.
2. Do not perform RPS channel logic test for combinations designated as not allowed. Matrix assumes that test does not remove RX Trip and actuation function.
3. The CST must be available to provide suction to the AFW Pumps otherwise all three pumps are considered unavailable.
4. A number of power systems that will not have planned maintenance unavailabilities, are not included on the matrix: 4KV AC (5170), 480V AC (5175 non -safety related), 208/120V AC (5185) and transformers and switchyard (5120). However, testing or maintenance activities on these systems may introduce additional risk of an undervoltage initiator. Therefore, do not perform testing or maintenance activities on these systems while performing EDG or AFW SDP maintenance due to increased risk of a station blackout.
5. This matrix considers risk only from a Core Damage.

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 1 .Matrix Event Description And Safety Function

SYS	TRAIN (Note 1)	MATRIX EVENT DESCRIPTION (Note 5)	SHORT NAME (see matrices)	TRAIN SAFETY FUNCTION To Be Maintained for Not Recommended Combinations	MODELLING NOTES
6175	B	Fire Pump, Motor-Driven Unavailable	FIRE PUMP MOTOR	Provide Alternate Cooling to SI, AFW, and Charging Pumps	
6270	B	Deepwell Pump A Unavailable	DEEPWELL PUMP A	Provide Makeup to CST or Alternate AFW Supply	
6270	B	Deepwell Pump C Unavailable	DEEPWELL PUMP C	Provide Makeup to CST or Alternate AFW Supply	

NOTES:

1. Trains as designated by 12 week on-line schedule.
2. Do not perform RPS channel logic test for combinations designated as not allowed. Matrix assumes that test does not remove RX Trip and actuation function.
3. The CST must be available to provide suction to the AFW Pumps otherwise all three pumps are considered unavailable.
4. A number of power systems that will not have planned maintenance unavailabilities, are not included on the matrix: 4KV AC (5170), 480V AC (5175 non -safety related), 208/120V AC (5185) and transformers and switchyard (5120). However, testing or maintenance activities on these systems may introduce additional risk of an undervoltage initiator. Therefore, do not perform testing or maintenance activities on these systems while performing EDG or AFW SDP maintenance due to increased risk of a station blackout.
5. This matrix considers risk only from a Core Damage.

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Table 2. Matrix Showing Allowable Hours for Plant Configurations To Remain Non-Risk Significant

(DELTA CDP<1E-06)

Train A Matrix

Exceeding these allowed hours require PGM approval, review of non-quantifiable factors, contingency planning and PSA insights. X - Safety Significant Exceeds Maximum Instantaneous CDF of 1E-3 and SHOULD BE AVOIDED		RPS CHANNEL A	RCS PZR PORV 456	RHR PUMP A	CVCS CHGP B	SI PUMP A	S/G A PORV RV-1	S/G B PORV RV-2	S/G C PORV RV-3	MFWP A	AFW MDP A	AFW SDP	SW PUMP A	SW PUMP B	CCW PUMP A	CCW PUMP B	EDG A	EMERGENCY BUS E1	DC BAT CHG A/A1	AIR COMP A	AIR COMP PRIM	FIRE PUMP DIESEL	DEEPWELL PUMP B
		1080	2005	2045	2060	2080	3020	3020	3020	3050	3065	3065	4060	4060	4080	4080	5095	5175	5235	6135	6135	6175	6270
RPS CHANNEL A	1080	804	56	136	461	326	117	117	117	296	78	71	584	584	466	471	122	718	639	804	775	617	303
RCS PZR PORV 456	2005	56	93	11	85	26	54	54	54	39	37	26	86	85	83	76	24	91	90	91	87	92	79
RHR PUMP A	2045	136	11	174	149	169	22	22	22	116	65	60	161	161	154	155	106	169	165	173	172	164	143
CVCS CHGP B	2060	461	85	149	1081	147	124	124	124	363	95	96	718	718	506	279	165	932	804	1068	1043	789	461
SI PUMP A	2080	326	26	169	147	576	83	83	83	278	78	86	456	456	404	407	188	531	489	576	569	484	337
S/G A PORV RV-1	3020	117	54	22	124	83	136	52	52	109	59	59	128	128	124	124	80	134	131	136	136	131	117
S/G B PORV RV-2	3020	117	54	22	124	83	52	136	52	109	59	59	128	128	124	124	80	134	131	136	136	131	117
S/G C PORV RV-3	3020	117	54	22	124	83	52	52	136	109	59	59	128	128	124	124	80	134	131	136	136	131	117
MFWP A	3050	296	39	116	363	278	109	109	109	548	14	45	438	438	389	393	144	506	468	548	541	463	311
AFW MDP A	3065	78	37	65	95	78	59	59	59	14	104	9	98	98	97	97	73	102	101	104	102	100	93
AFW SDP	3065	71	26	60	96	86	59	59	59	45	9	105	92	90	98	98	14	100	92	105	102	100	93
SW PUMP A	4060	584	86	161	718	456	128	128	128	438	98	92	2190	73	782	850	184	1718	1307	2190	2086	903	551
SW PUMP B	4060	584	85	161	718	456	128	128	128	438	98	90	73	2190	834	850	184	1718	1307	2190	2086	913	551
CCW PUMP A	4080	466	83	154	506	404	124	124	124	389	97	98	782	834	1348	X	161	995	834	1348	1307	932	506
CCW PUMP B	4080	471	76	155	279	407	124	124	124	393	97	98	850	850	X	1390	172	1153	963	1369	1327	942	509
EDG A	5095	122	24	106	165	188	80	80	80	144	73	14	184	184	161	172	196	190	184	196	194	131	128
EMERGENCY BUS E1	5175	718	91	169	932	531	134	134	134	506	102	100	1718	1718	995	1153	190	6738	2137	6257	5475	2037	706
DC BAT CHG A/A1	5235	639	90	165	804	489	131	131	131	468	101	92	1307	1307	834	963	184	2137	3129	3129	2920	1537	93
AIR COMP A	6135	804	91	173	1068	576	136	136	136	548	104	105	2190	2190	1348	1369	196	6257	3129	8760	8760	2920	804
AIR COMP PRIM	6135	775	87	172	1043	569	136	136	136	541	102	102	2086	2086	1307	1327	194	5475	2920	8760	8760	2738	789
FIRE PUMP DIESEL	6175	617	92	164	789	484	131	131	131	463	100	100	903	913	932	942	131	2037	1537	2920	2738	2920	635
DEEPWELL PUMP B	6270	303	79	143	461	337	117	117	117	311	93	93	551	551	506	509	128	706	93	804	789	635	804

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 2. Matrix Showing Allowable Hours for Plant Configurations To Remain Non-Risk Significant

(DELTA CDP<1E-06)

Train B Matrix

Exceeding these allowed hours require PGM approval, review of non-quantifiable factors, contingency planning and PSA insights. X - Safety Significant Exceeds Maximum Instantaneous CDF of 1E-3 and SHOULD BE AVOIDED		RPS CHANNEL B	RCS PZR PORV 455C	RCS BLOCK VALVES	RHR PUMP B	CVCS CHGP A	CVCS CHGP C	SI PUMP C	MFWP B	AFW MDP B	SW PUMP C	SW PUMP D	CCW PUMP C	EDG B	DSDG	DS BUS	EMERGENCY BUS E2	DC BAT CHG B/B1	AIR COMP B	AIR COMP D	FIRE PUMP MOTOR	DEEPWELL PUMP A	DEEPWELL PUMP C
		1080	2005	2005	2045	2060	2060	2080	3050	3065	4060	4060	4080	5095	5098	5114	5175	5235	6135	6135	6175	6270	6270
RPS CHANNEL B	1080	804	56	400	168	409	461	503	326	124	105	101	479	89	163	66	558	712	804	730	226	701	303
RCS PZR PORV 455C	2005	56	93	92	15	85	85	36	40	44	57	56	86	10	85	52	88	91	91	87	72	90	79
RCS BLOCK VALVES	2005	400	92	2738	210	654	775	932	244	148	143	137	952	76	241	90	1095	1947	2576	1825	293	1947	596
RHR PUMP B	2045	168	15	210	229	180	188	199	143	85	91	89	198	78	121	66	204	221	229	222	135	222	178
CVCS CHGP A	2060	409	85	654	180	859	153	95	370	147	124	83	466	85	207	93	576	762	859	782	238	768	417
CVCS CHGP C	2060	461	85	775	188	153	1081	97	404	153	132	126	283	107	192	38	679	922	1068	963	252	932	461
SI PUMP C	2080	503	36	932	199	95	97	1413	411	122	137	131	718	110	216	49	804	1168	1413	1217	267	1184	512
MFWP B	3050	326	40	244	143	370	404	411	644	23	122	117	447	100	184	78	476	588	644	600	217	528	173
AFW MDP B	3065	124	44	148	85	147	153	122	23	178	75	73	158	72	98	51	162	169	177	136	116	173	146
SW PUMP C	4060	105	57	143	91	124	132	137	122	75	151	27	137	67	93	56	140	148	151	148	76	145	79
SW PUMP D	4060	101	56	137	89	83	126	131	117	73	27	144	131	57	92	56	128	141	144	142	73	139	76
CCW PUMP C	4080	479	86	952	198	466	283	718	447	158	137	131	1460	110	201	X	819	1200	1436	1234	268	1200	518
EDG B	5095	89	10	76	78	85	107	110	100	72	67	57	110	119	26	22	111	116	119	117	87	117	102
DSDG	5098	163	85	241	121	207	192	216	184	98	93	92	201	26	258	93	225	248	258	250	145	249	196
DS BUS	5114	66	52	90	66	93	38	49	78	51	56	56	X	22	93	93	78	91	93	92	70	91	74
EMERGENCY BUS E2	5175	558	88	1095	204	576	679	804	476	162	140	128	819	111	225	78	1825	1436	1825	1510	278	1348	528
DC BAT CHG B/B1	5235	712	91	1947	221	762	922	1168	588	169	148	141	1200	116	248	91	1436	6738	6257	3809	313	3021	695
AIR COMP B	6135	804	91	2576	229	859	1068	1413	644	177	151	144	1436	119	258	93	1825	6257	8760	5153	328	6738	804
AIR COMP D	6135	730	87	1825	222	782	963	1217	600	136	148	142	1234	117	250	92	1510	3809	5153	8760	316	3809	701
FIRE PUMP MOTOR	6175	226	72	293	135	238	252	267	217	116	76	73	268	87	145	70	278	313	328	316	328	314	234
DEEPWELL PUMP A	6270	701	90	1947	222	768	932	1184	528	173	145	139	1200	117	249	91	1348	3021	6738	3809	314	6738	21
DEEPWELL PUMP C	6270	303	79	596	178	417	461	512	173	146	79	76	518	102	196	74	528	695	804	701	234	21	804

PSA OF ON-LINE MAINTENANCE FOR H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

Table 2. Matrix Showing Allowable Hours for Plant Configurations To Remain Non-Risk Significant

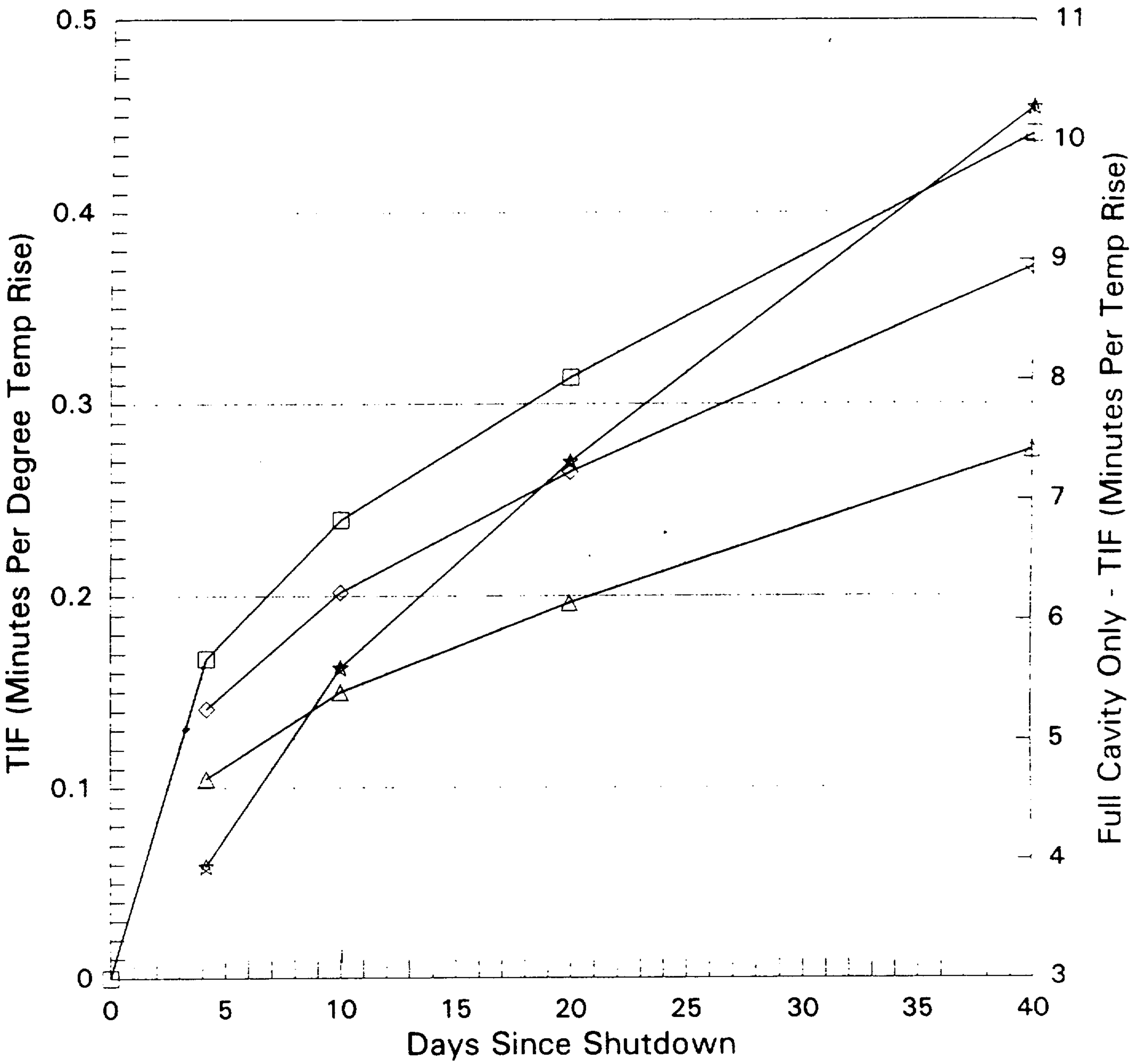
(DELTA CDP<1E-06)

Train A by Train B Matrix

Exceeding these allowed hours require PGM approval, review of non-quantifiable factors, contingency planning and PSA insights. X - Safety Significant Exceeds Maximum Instantaneous CDF of 1E-3 and SHOULD BE AVOIDED		RPS CHANNEL B	RCS PZR PORV 455C	RCS BLOCK VALVES	RHR PUMP B	CVCS CHGP A	CVCS CHGP C	SI PUMP C	MFWP B	AFW MDP B	SW PUMP C	SW PUMP D	CCW PUMP C	EDG B	DSDG	DS BUS	EMERGENCY BUS E2	DC BAT CHG B/B1	AIR COMP B	AIR COMP D	FIRE PUMP MOTOR	DEEPWELL PUMP A	DEEPWELL PUMP C
		1080	2005	2005	2045	2060	2060	2080	3050	3065	4060	4060	4080	5095	5098	5114	5175	5235	6135	6135	6175	6270	6270
RPS CHANNEL A	1080	105	56	400	168	409	461	421	326	124	105	101	479	89	163	66	558	712	804	730	226	701	303
RCS PZR PORV 456	2005	56	72	93	15	85	85	36	40	44	57	56	86	10	85	52	88	91	91	87	72	90	79
RHR PUMP A	2045	136	11	134	X	144	149	155	119	86	81	78	155	20	104	60	159	167	173	170	113	169	143
CVCS CHGP B	2060	461	85	775	188	376	487	97	404	153	132	126	521	102	192	44	679	922	1068	963	252	932	461
SI PUMP A	2080	326	26	289	164	142	147	X	301	136	119	116	413	21	178	60	436	531	576	541	209	534	337
S/G A PORV RV-1	3020	117	54	130	22	121	124	65	113	77	71	70	124	64	89	56	128	134	136	134	97	134	117
S/G B PORV RV-2	3020	117	54	130	22	121	124	65	113	77	71	70	124	64	89	56	128	134	136	134	97	134	117
S/G C PORV RV-3	3020	117	54	130	22	121	124	65	113	77	71	70	124	64	89	56	128	134	136	134	97	134	117
MFWP A	3050	296	39	229	138	336	363	370	85	106	119	114	398	98	176	79	152	348	548	515	205	461	311
AFW MDP A	3065	78	40	94	71	93	95	97	77	10	61	60	97	22	74	49	17	27	104	102	79	102	93
AFW SDP	3065	71	26	80	65	94	96	94	51	15	45	43	98	12	74	49	79	93	105	100	76	104	93
SW PUMP A	4060	584	87	1217	207	541	718	859	498	164	76	59	876	69	217	66	649	1653	2190	1752	159	1390	551
SW PUMP B	4060	584	85	1217	207	548	718	859	498	164	77	59	876	65	211	65	649	1653	2190	1752	159	1390	551
CCW PUMP A	4080	466	83	903	196	167	528	690	436	157	136	131	X	100	218	93	718	1109	1348	1168	265	1138	506
CCW PUMP B	4080	471	76	894	196	459	515	701	440	157	136	131	X	83	155	X	724	1138	1369	1184	265	1153	509
EDG A	5095	122	24	145	72	116	155	98	150	43	48	40	151	X	29	23	154	188	196	191	122	190	128
EMERGENCY BUS E1	5175	718	91	1947	221	762	932	1153	548	121	25	25	1043	116	247	83	X	3244	6257	3650	204	3021	706
DC BAT CHG A/A1	5235	639	90	1460	206	679	804	826	231	27	34	34	867	113	238	85	1068	2137	3129	2037	231	2190	93
AIR COMP A	6135	804	91	2576	229	859	1068	1413	644	177	151	144	1436	119	258	93	1825	6257	2920	5153	328	6738	804
AIR COMP PRIM	6135	775	87	2137	226	842	1043	1369	635	177	151	144	1390	119	256	93	1752	5153	8760	1436	326	5840	789
FIRE PUMP DIESEL	6175	617	92	1413	213	782	789	963	531	167	136	90	973	92	248	91	1138	2037	2920	2190	26	2086	635
DEEPWELL PUMP B	6270	303	79	596	178	417	461	512	173	146	79	76	518	102	196	74	528	695	804	701	234	21	21

Curve 3.5 - Time To CV Closure

Time = (200 - Initial Water Temp.) x Thermal Inertia Factor (TIF)



0 To -10" Below Flange

-10" To -36" Below Flange

-36" To -72" Below Flange

Refueling Cavity Full

Based on Calculation RNP-M/MECH-1590

Use Thermal Inertia Factor = 0.00167 x t(hrs) prior to 100 Hours After Shutdown

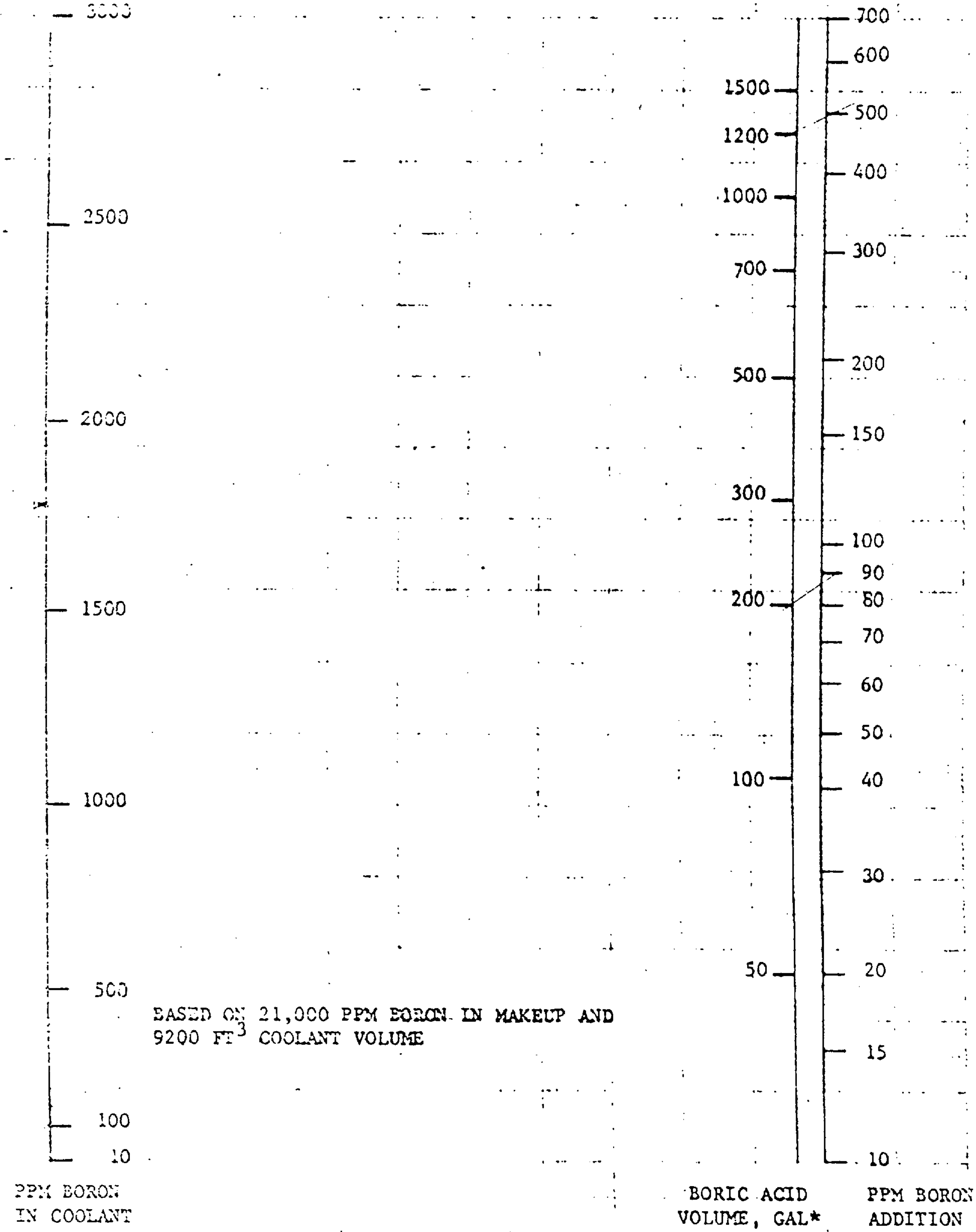


FIGURE S-3.1-3 BORON ADDITION - COOLANT HOT (~580°F)

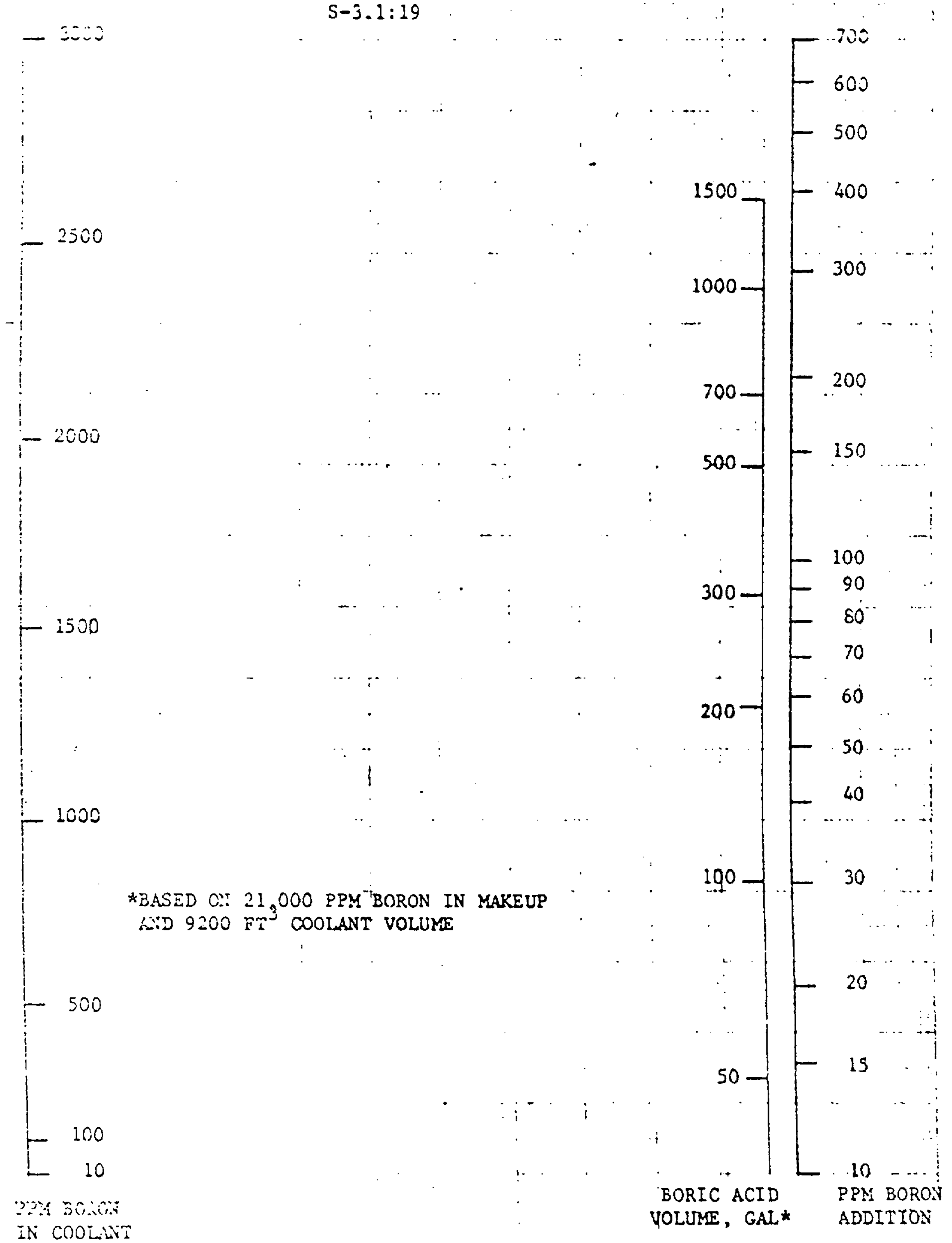


FIGURE S-3.1-4 BORON ADDITION - COOLANT COLD (-100°F)

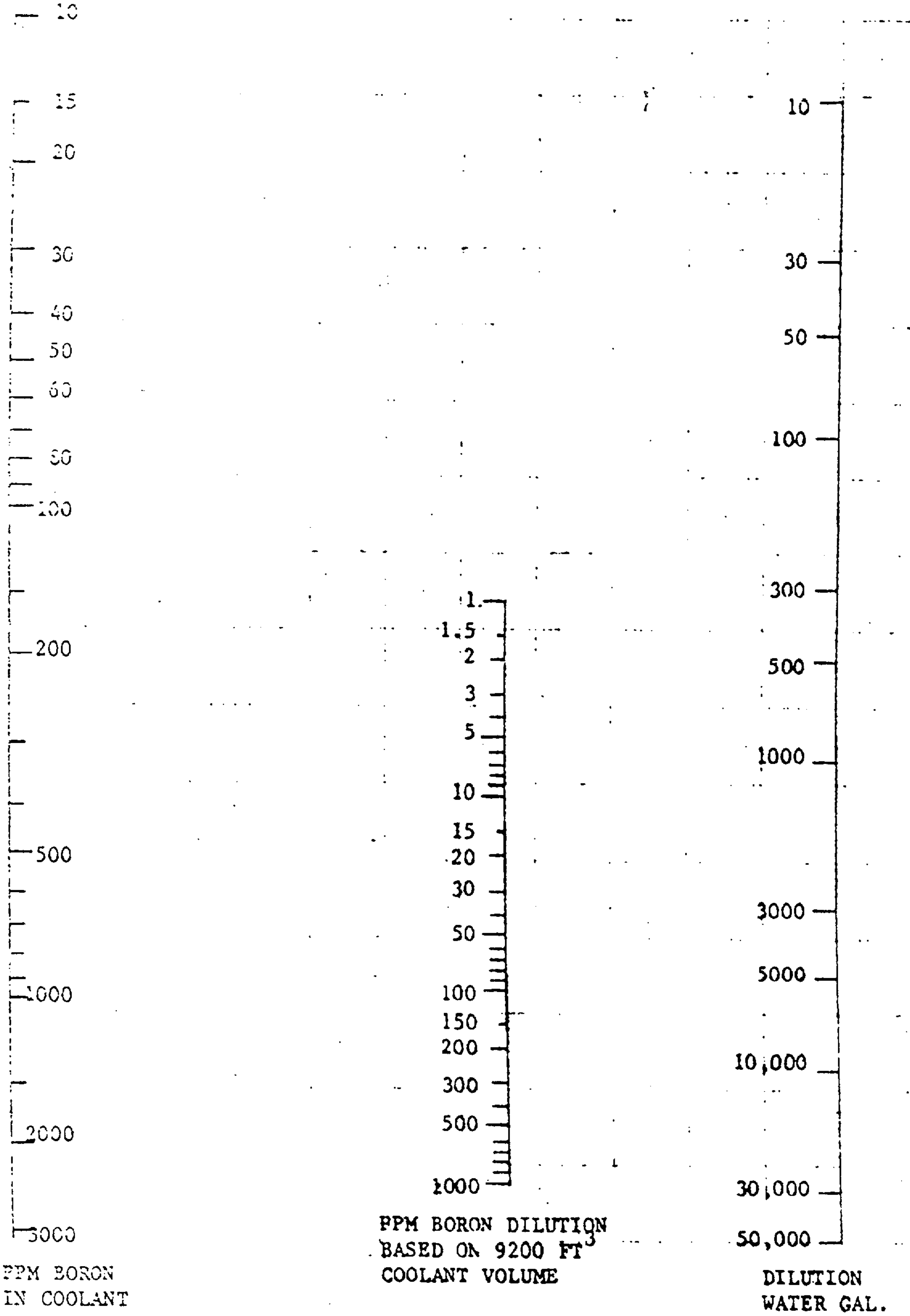


FIGURE S-3.1-7 DILUTION NOMOGRAPH - COOLANT HOT (~580°F)

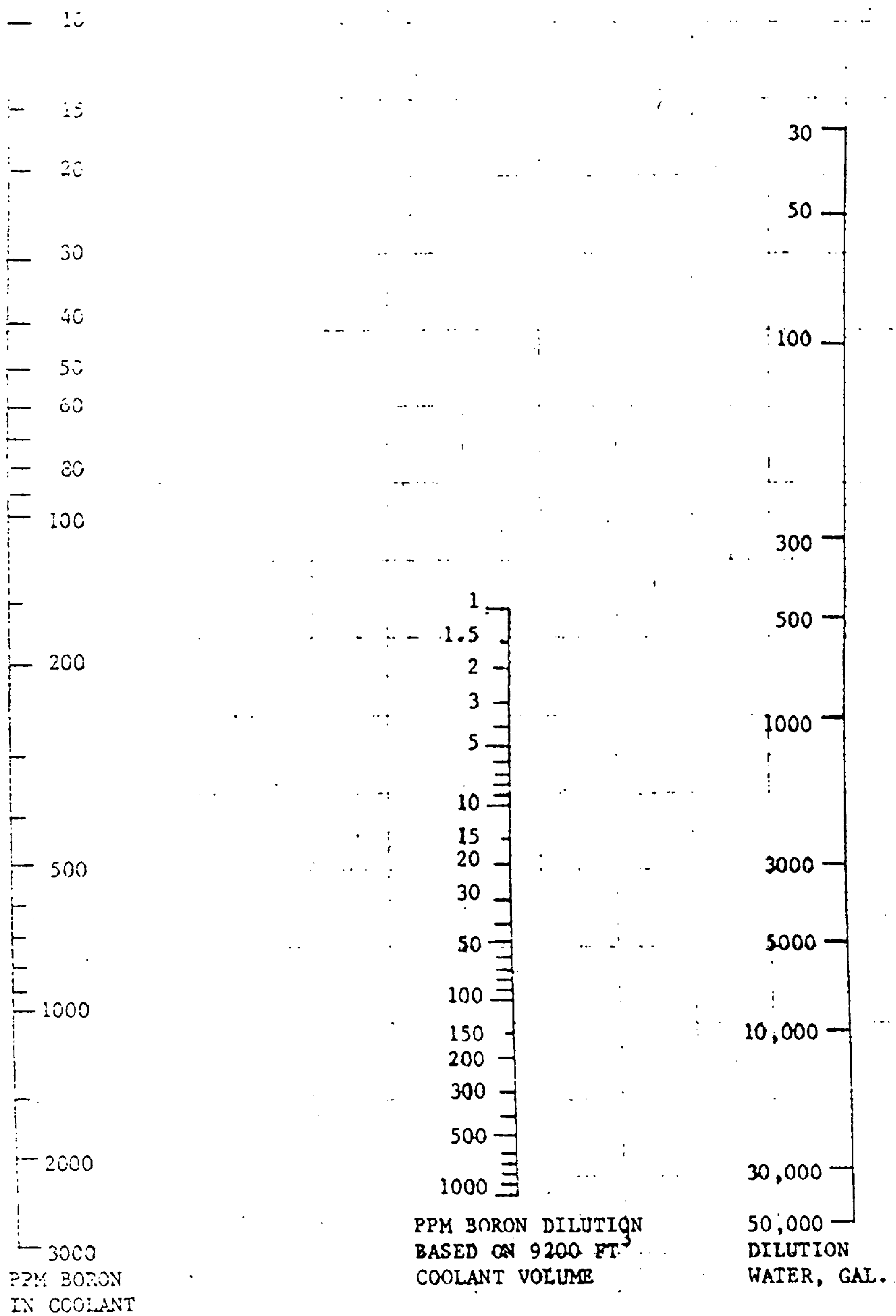


FIGURE S-3.1-8 DILUTION NOMOGRAPH - COOLANT COLD (~100°F)

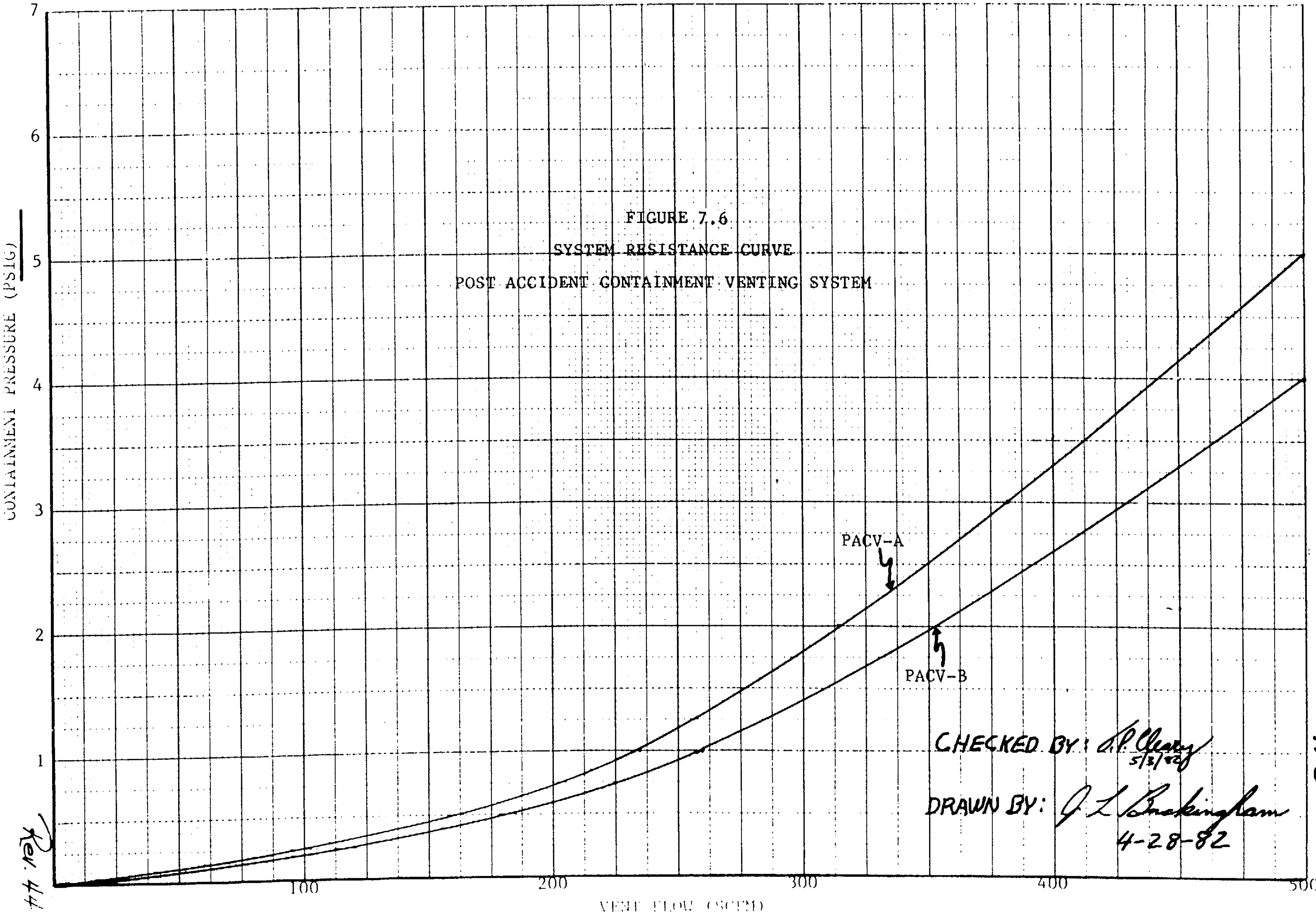


FIGURE 7.6

SYSTEM RESISTANCE CURVE

POST ACCIDENT CONTAINMENT VENTING SYSTEM

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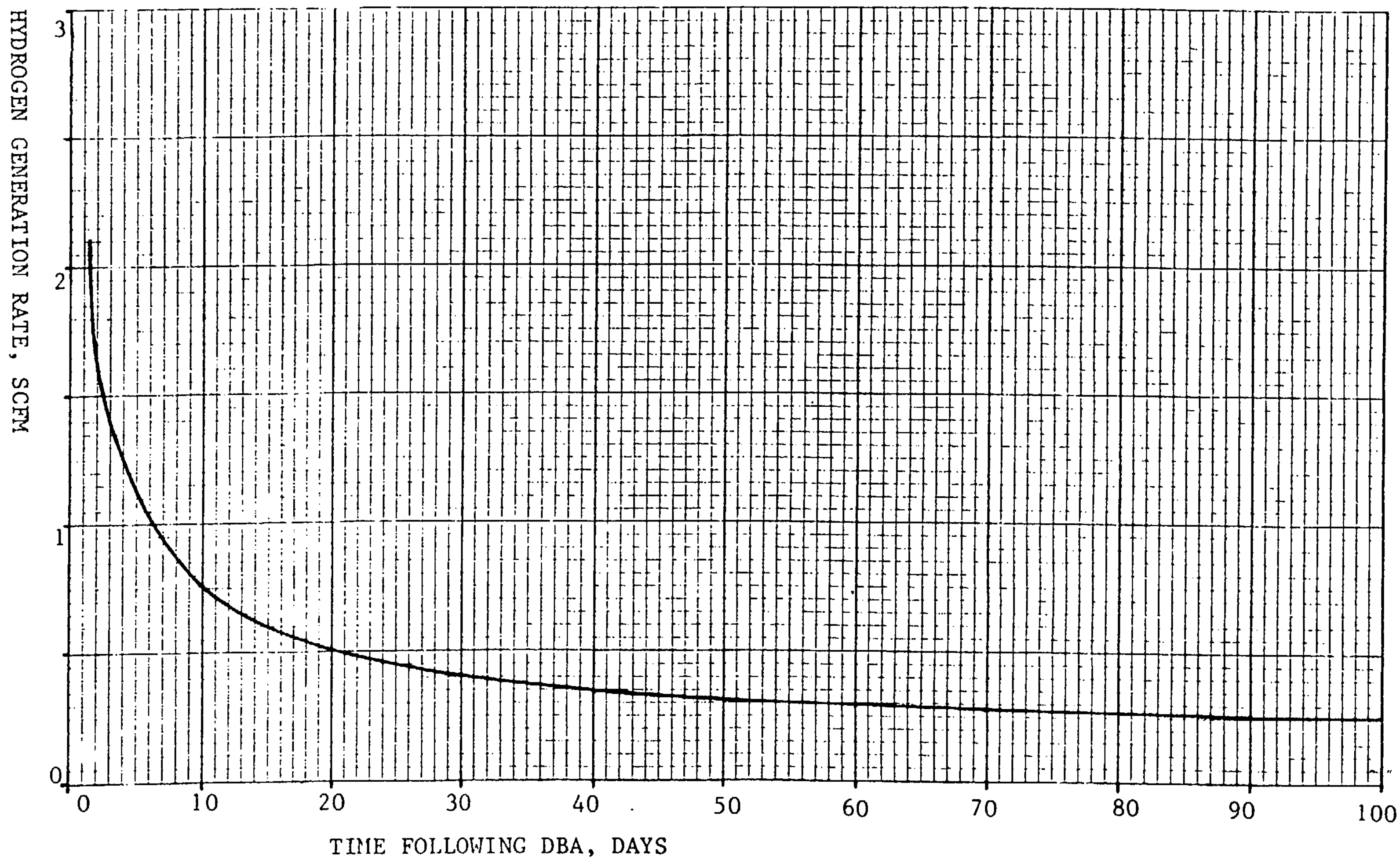


CURVE 7.16 TOTAL HYDROGEN GENERATION RATE FROM ALL SOURCES.

Drawn By: James M. [signature] 10-19-84

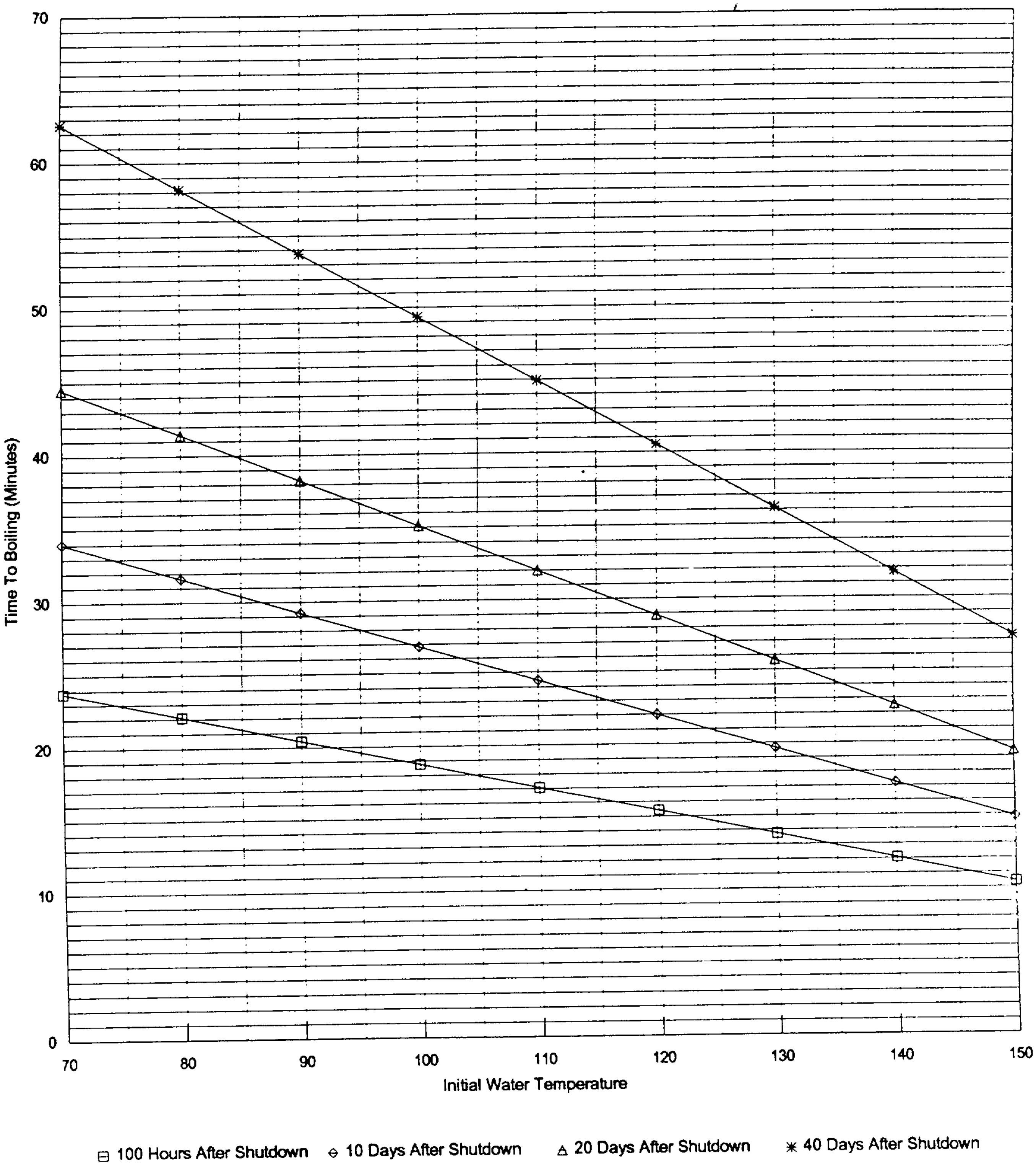
Checked By: [signature] 10/19/84

830 Day Full Power TID Core
DBA Conditions
Hydrogen Sources:

- Zirconium-Water Reaction
- Aluminum Corrosion
- Core Solution Radiolysis
- Sump Solution Radiolysis

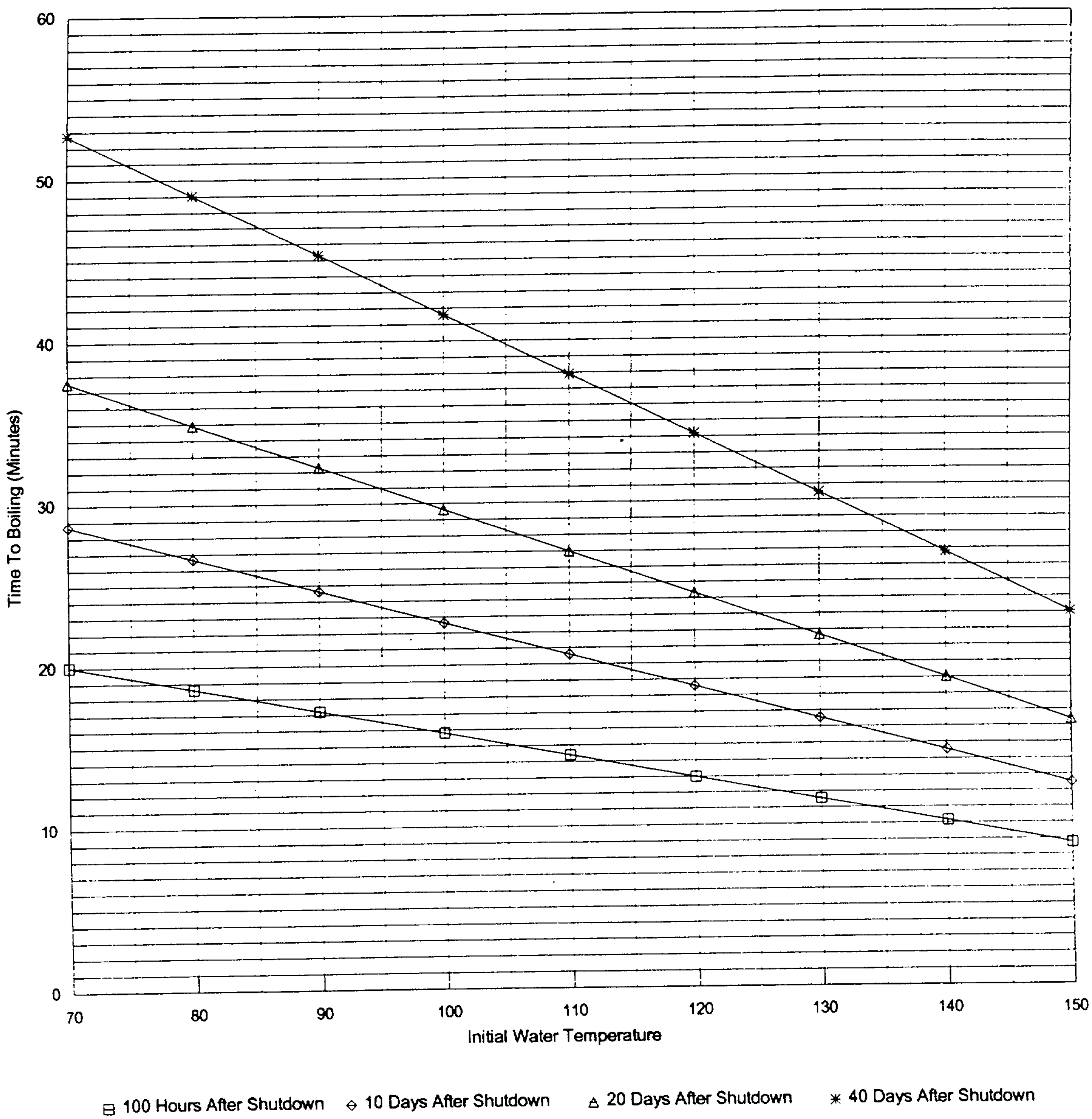
Curve 7.16

Curve 7.19 - Loss of Residual Heat Removal Cooling
Water Level Between 0" to -10" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

Curve 7.20 - Loss of Residual Heat Removal Cooling
Water Level Between -10" to -36" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

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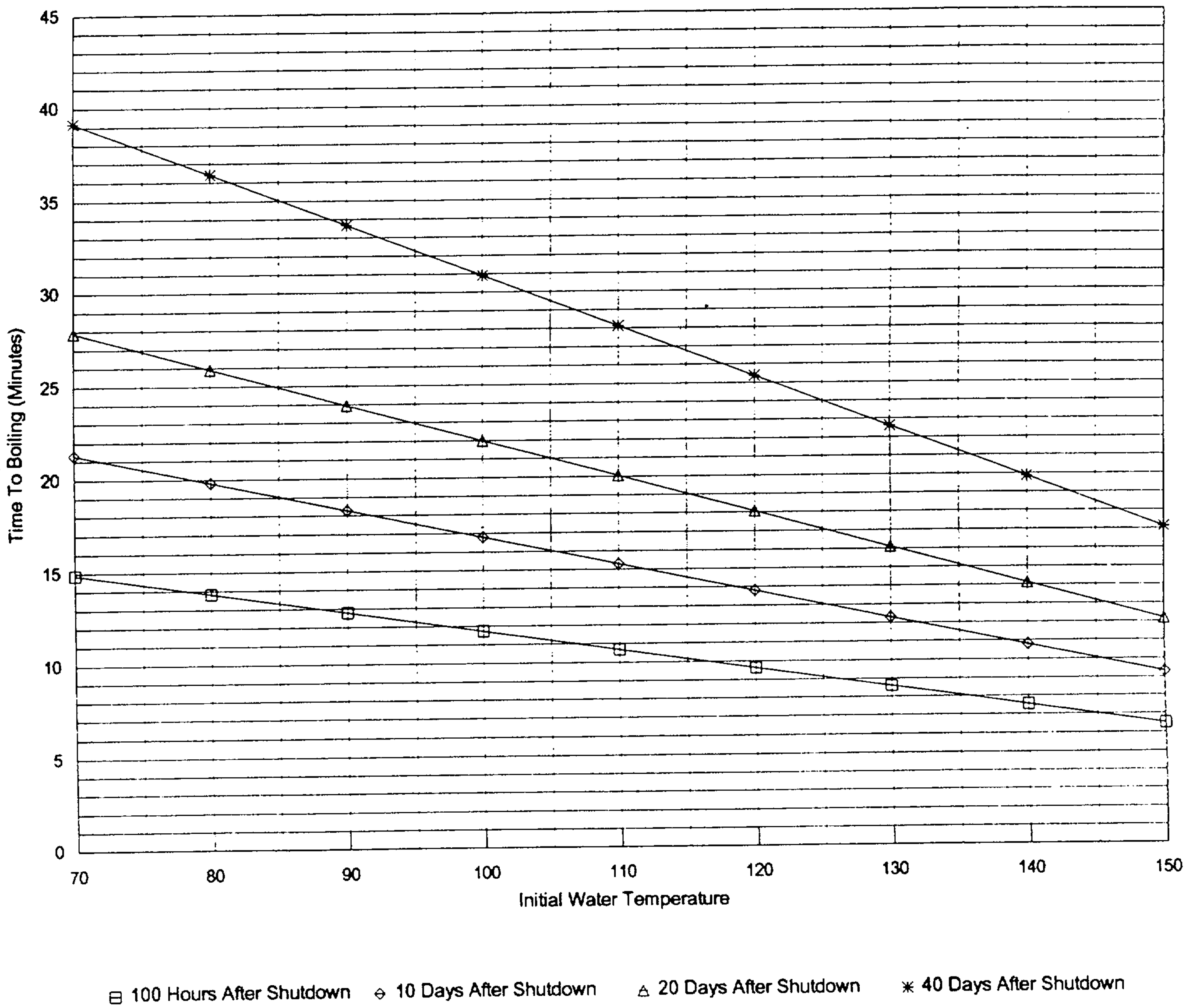
Curve 7.21 - Loss of Residual Heat Removal Cooling Water Level Between -36" to -72" Below Vessel Flange



Based on calculation RNP-M/MECH-1590

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	------------Note------------ LCO 3.0.4 is not applicable. ----------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with $T_{avg} < 500°F$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	-------------------NOTE------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	------------------NOTE------------------ Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

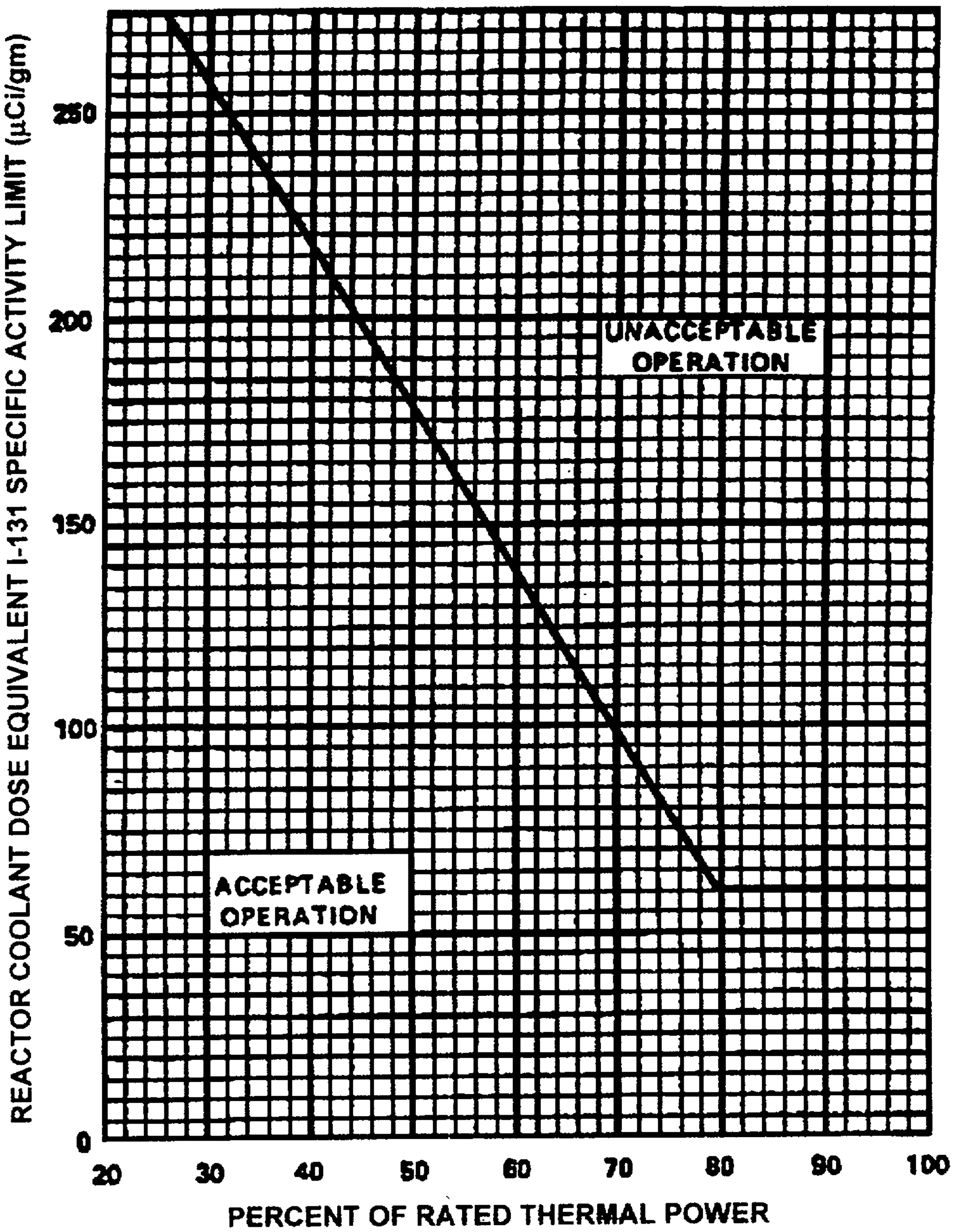


Figure 3.4.16-1
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

3.7 PLANT SYSTEMS

3.7.4 Auxiliary Feedwater (AFW) System

LCO 3.7.4 Four AFW flow paths and three AFW pumps shall be OPERABLE.

-------------------------NOTE-------------------------
Only one AFW flow path with one motor driven pump is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is being used for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AFW pump inoperable in MODE 1, 2, or 3. <u>OR</u> One or two AFW flow paths inoperable in MODE 1, 2, or 3.	A.1 Restore AFW pump or flow path(s) to OPERABLE status.	7 days <u>AND</u> 8 days from discovery of failure to meet the LCO
B. Two motor driven AFW pumps inoperable in MODE 1, 2, or 3. <u>OR</u> Three motor driven AFW flow paths inoperable in MODE 1, 2, or 3.	B.1 Restore one motor driven AFW pump or one flow path to OPERABLE status.	24 hours <u>AND</u> 8 days from discovery of failure to meet the LCO

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	18 hours
D. Steam driven AFW pump or flow path inoperable in MODE 1, 2, or 3. AND One motor driven AFW pump or flow path inoperable in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	18 hours
E. Four AFW flow paths inoperable in MODE 1, 2, or 3. OR Three AFW pumps inoperable in MODE 1, 2, or 3.	E.1	---------NOTE------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW pump and flow path are restored to OPERABLE status. -------------------- Initiate action to restore one AFW pump and flow path to OPERABLE status.	Immediately
F. Required AFW pump and flow path inoperable in MODE 4.	F.1	Initiate action to restore AFW pump and flow path to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in the steam supply flow path to the steam driven AFW pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.4.2	------------------NOTE------------------ Not required to be performed for the steam driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	31 days on a STAGGERED TEST BASIS
SR 3.7.4.3	------------------NOTE------------------ Not applicable in MODE 4 when steam generator is being used for heat removal. -- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.4	------------------NOTES------------------ 1. Not required to be performed for the steam driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is being used for heat removal. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.4.5	------------------NOTE------------------ Not required to be performed for the steam driven AFW pump until prior to entering MODE 1. --- Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days
SR 3.7.4.6	Verify the AFW automatic bus transfer switch associated with discharge valve V2-16A operates automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.6 Component Cooling Water (CCW) System

LCO 3.7.6 Two CCW trains powered from emergency power supplies shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required CCW train inoperable.	A.1	--------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4," for residual heat removal loops made inoperable by CCW. -------------------- Restore required CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	------------------NOTE------------------ Isolation of CCW flow to individual components does not render the CCW System inoperable. -- Verify each required CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.6.2	Verify each required CCW pump starts automatically on an actual or simulated LOP DG Start undervoltage signal.	18 months

INITIAL SUBMITTAL

ROBINSON EXAM 2001-301

MARCH 26 - APRIL 2, 2001

INITIAL SUBMITTAL -
RO/SRO COMMON
WRITTEN EXAMINATION
QUESTIONS

51-95

Question: 51

Given the following conditions:

- A reactor trip and safety injection have occurred.
- Due to multiple failures, an entry has been made to EPP-16, "Uncontrolled Depressurization of All Steam Generators."
- Containment pressure is 8 psig.
- The RCS cooldown rate is 130 °F/hour.
- SG levels are:

SG	**LEVEL**
'A'	1%
'B'	3%
'C'	14%

Which ONE (1) of the following actions should be taken?

a. Secure all AFW to 'A' and 'B' SGs, while feeding 'C' SG at a rate between 80 gpm and 90 gpm

b. Secure all AFW flow to all SGs until 'C' SG is below 8%, then feed **ONLY** 'C' SG at a rate between 80 gpm and 90 gpm

c. Feed 'A' and 'B' SGs at a rate between 80 gpm and 90 gpm, while feeding 'C' SG only as needed to maintain the RCS cooldown rate below 100 °F/hour

d. Feed all SGs at a rate between 80 gpm and 90 gpm

Answer:

d. Feed all SGs at a rate between 80 gpm and 90 gpm

QUESTION NUMBER: 51
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: WE12EK1.2

Knowledge of the operational implications of the following concepts as they apply to the (Uncontrolled Depressurization of all Steam Generators) Normal, abnormal and emergency operating procedures.

K/A IMPORTANCE: ***RO*** 3.5 ***SRO*** 3.8
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: EPP-016-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps as directed in EPP-16.

REFERENCES: EPP-016

SOURCE: ***New*** ☒ ***Significantly Modified*** ☐ ***Direct*** ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible since this is the required rate for 'C' SG and this would limit the cooldown, but all SGs must be fed at a minimum rate of 80 gpm until level is above 18%.

b. Plausible since this would limit the cooldown, but all SGs must be fed at a minimum rate of 80 gpm until level is above 18%.

c. Plausible since this is the required rate for 'A' and 'B' SGs and this would limit the cooldown, but all SGs must be fed at a minimum rate of 80 gpm until level is above 18%, not 8% due to adverse containment conditions.

d. **CORRECT** With an excessive cooldown rate, AFW flow is throttled to between 80 and 90 gpm. All SGs must be fed at a rate of at least 80 gpm due to level being below 18%.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of plant conditions to determine feed rate for all faulted SGs

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	Locally Verify The Following Valves - CLOSED a. BYPASS DRN & WARM-UP LINE TO AFW PUMP: • MS-20 • MS-29 • MS-38 b. STEAM LINE BEFORE SEAT DRAIN ROOT ISOL: • MS-19 • MS-28 • MS-37 c. STEAM LINE AFTER SEAT DRAIN ROOT ISOL: • MS-21 • MS-30 • MS-39	
7.	Check Cooldown Rate In RCS Cold Legs - GREATER THAN 100°F/HR IN LAST 60 MINUTE	Go To Step 11.
8.	Check MDAFW Pump Status - AT LEAST ONE AVAILABLE	Go To Step 10.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
9.	Control Feed Flow To Minimize RCS Cooldown As Follows:	
	a. Throttle feed flow to between 80 gpm and 90 gpm to each S/G using MDAFW FLOW CONTROLLER: • FIC-1424, AFW PUMP A DISCH FLOW OR • FIC-1425, AFW PUMP B DISCH FLOW	a. Establish between 80 gpm and 90 gpm feed flow to each S/G as follows: 1) Open the breakers for MDAFW HEADER DISCHARGE Valves: • V2-16A (MCC-9, COMPT-2ML) • V2-16C (MCC-9, COMPT-3J) • V2-16A (MCC-10, COMPT-4C) • V2-16B (MCC-10, COMPT-4F) 2) Locally throttle AFW HDR DISCH Valves to establish 80 gpm to 90 gpm to each S/G: • AFW-V2-16A - S/G "A" • AFW-V2-16B - S/G "B" • AFW-V2-16C - S/G "C" 3) Go To Step 11.
	b. Go To Step 11	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
10.	Control Feed Flow To Minimize RCS Cooldown As Follows:	
	a. Throttle feed flow to between 80 gpm and 90 gpm to each S/G using FIC-6416, SDAFW FLOW CONTROLLER	a. Establish between 80 gpm and 90 gpm feed flow to each S/G as follows: 1) Open the breakers to SDAFW PUMP TO S/G: • V2-14A (MCC-10, CMPT-3C) • V2-14B (MCC-9, COMPT-1C) • V2-14C (MCC-10, COMPT-4M) 2) Locally throttle SDAFW PUMP FW DISCH TO SG to establish 80 gpm to 90 gpm to each S/G: • AFW-V2-14A - S/G "A" • AFW-V2-14B - S/G "B" • AFW-V2-14C - S/G "C"
11.	Maintain A Minimum Of 80 GPM AFW Flow To Each S/G With Level Less Than 8% [18%]	
12.	Check S/G Levels - ALL LESS THAN 50%	Control feed flow to maintain level less than 50% in all S/Gs.
13.	Request Periodic RCS Boron Samples	
14.	Check RCS Hot Leg Temperatures - STABLE <u>OR</u> DECREASING	Control feed flow <u>OR</u> steam dump to stabilize RCS Hot Leg temperatures.

Question: 52

Given the following conditions:

- The unit is operating at 100% power.
- Testing is being performed on Reactor Trip Breaker 'B' and it is currently open.
- A loss of the 'A' 125 VDC Distribution Panel occurs.
- Reactor Trip Breaker 'A' fails to open.

Which ONE (1) of the following describes the expected response of the plant due to this sequence of events, assuming **NO** operator action?

a. **NO** reactor trip occurs

b. Reactor Trip Bypass Breaker 'B' opens on an Undervoltage trip **ONLY**, resulting in a reactor trip

c. Reactor Trip Bypass Breaker 'B' opens on a Shunt trip **ONLY**, resulting in a reactor trip

d. Reactor Trip Bypass Breaker 'B' opens on **BOTH** an Undervoltage trip and a Shunt trip, resulting in a reactor trip

Answer:

b. Reactor Trip Bypass Breaker 'B' opens on an Undervoltage trip **ONLY**, resulting in a reactor trip

QUESTION NUMBER: 52

TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2

K/A: 012K2.01

Knowledge of bus power supplies to the following: RPS channels, components, and interconnections

K/A IMPORTANCE: ***RO*** 3.3 ***SRO*** 3.7

10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: RPS-06

LIST power supplies for the major RPS System components as listed in the EDPs.

REFERENCES: SD-011
EDP-004

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number RPS-09 006

JUSTIFICATION:

a. Plausible since no trip would occur if 'B' bus were lost or if testing were being performed on 'A' trip breaker.

b. **CORRECT** A loss of 125 VDC bus will cause an undervoltage trip of the related trip breaker and the opposite train bypass breaker, but will not cause a shunt trip since power is required to cause a shunt trip.

c. Plausible since the automatic shunt trip relay loses power, but power is required to actually cause a shunt trip and the bypass breakers only trip on a shunt trip if the local trip mechanism is actuated.

d. Plausible since an undervoltage trip will occur, but no shunt trip will occur.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of the conditions expected during trip breaker testing and the effect of a loss of power

REFERENCES SUPPLIED:

The two Reactor Trip Breakers connect the output of the Rod Drive M-G Sets to the Control Rod Drive Cabinets.

The two Reactor Trip Bypass Breakers also connect the output of the M-G Sets to the Control Rod Drive Cabinets. The Bypass Breakers are used, one at a time, to allow for on-the-line testing and repair of the Reactor Trip Breakers and the logic trains. The Bypass Breakers are not to be in continuous service for > 12 hours.

The Reactor Trip Breakers are opened both mechanically (UV Coils) and electrically (automatic Shunt Trip Relays) by the RPS and/or either of the Reactor Trip Pushbuttons.

The Reactor Trip Bypass Breakers are opened mechanically (UV Coils) by the RPS and/or either of the Reactor Trip Pushbuttons, or electrically (Shunt Trip Relays) by the pushbutton located on the front of the breaker panel.

The UV Coils and Automatic Shunt Trip (AST) Relays for Reactor Trip Breaker "A" and Reactor Trip Bypass Breaker "B" (UV Coil only) are powered from "A" 125VDC Distribution Panel (same Power Supply Breaker as Logic Channel 1). The Trip Coils for Reactor Trip Breaker "A" and Reactor Trip Bypass Breaker "B" are also powered from "A" 125VDC Distribution Panel (different Power Supply Breaker than Channel 1).

The UV Coils and Automatic Shunt Trip (AST) Relays for Reactor Trip Breaker "B" and Reactor Trip Bypass Breaker "A" (UV Coil only) are powered from "B" 125VDC Distribution Panel (same Power Supply Breaker as Logic Channel 2). The Trip Coils for Reactor Trip Breaker "B" and Reactor Trip Bypass Breaker "A" are powered from "B" 125VDC Distribution Panel (different Power Supply Breaker than Logic Channel 2).

3.4 Reactor Trip Pushbuttons

The two Reactor Trip Pushbuttons are wired in series. This series wiring arrangement will cause the Reactor Trip and Reactor Trip Bypass Breakers to be opened by depressing either of the Reactor Trip Pushbuttons.

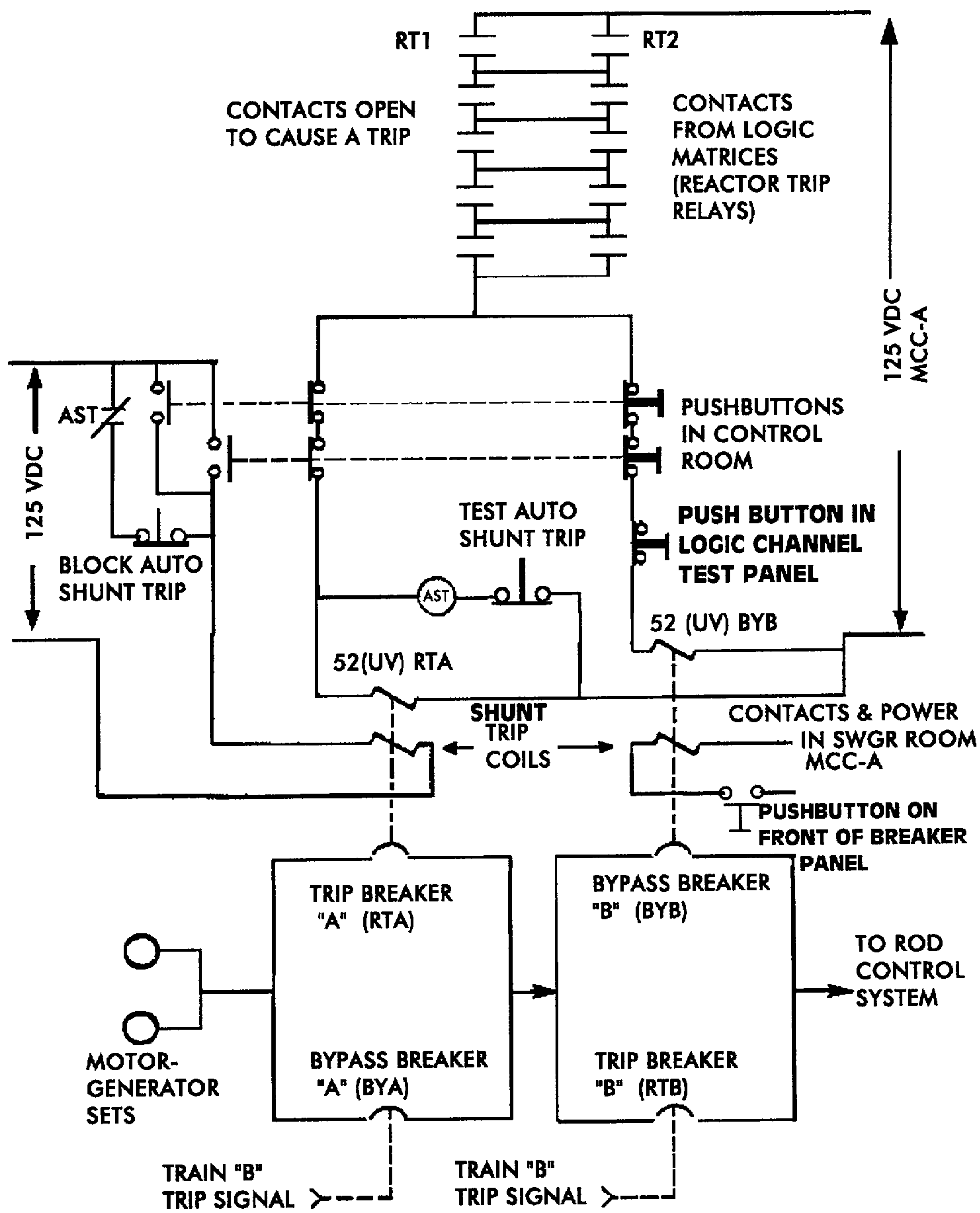
3.5 Trip Reset Pushbutton

After all the Reactor Trip signals have been removed, depressing the Trip Reset Pushbutton will cause the Reactor Trip Breakers to close.

3.6 Bistable Status Panels

SIMPLIFIED TRAIN "A" REACTOR TRIP BREAKER DIAGRAM

RPS-FIGURE-5 (Rev . 1)



2.0 DISTRIBUTION PANEL "A"

DISTRIBUTION PANEL "A"		
Power Supply: 125V DC MCC "A" Drawings:B-190627, SH 40A	Location: On 125V DC MCC "A"	
CKT#	LOAD	CWD
1	480V Switchgear No. E-1	955
2	4160V Switchgear Busses 1 & 2	955
3	Hydrogen Control Panel	875
4	480V Switchgear Busses 1 & 2A	955
5	Lighting Panel LP-33 (Alt Pwr via Auto Transfer Switch)	
6	125V DC Power Panel "A-1"	
7	Startup Transformer Motor Operated Disconnects	925
8	Diesel Generator "A" Exciter	880
9	Inverter "C"	
10	Reactor Trip Breaker "A" and Reactor trip Bypass Breaker "B"	45
11	Inverter "A"	
12	Rod Drive M-G Set "A"	71
13	Main Generator Exciter Field Breaker	862
14	Gas Stripper Control Cabinet "A"	173
`15	Generator Lockout Relay 86P	912
16	Aux. Panel "D-C" Fuse Panel	955
17	Main and Auxiliary Transformer Annunciators	940, 942
18	Reactor Protection Train "A"	955
19	Spare	
20	Safeguards Train "A"	955
21	Spare	
22	Turbine Auto Trip	710
23	Startup Transformer Annunciator	942
24	Diesel Generator "A" Control Power	945

5.0 DISTRIBUTION PANEL "B"

DISTRIBUTION PANEL "B"		
Power Supply: 125V DC MCC "B"		Location: On 125V DC MCC "B"
Drawings: B-190627 Sh 40B		
CKT#	LOAD	CWD
1	480V Switchgear No. E-2	956
2	4160V Switchgear Busses 3 & 4	956, 1341
3	4160V Breaker Test Panel	(Dwg 5379-1702)
4	480V Switchgear Busses 2B & 3	956, 1341
5	125V DC MCC "B-A"	
6	Sample Panel	88
7	Spare	
8	Diesel Generator "B" Exciter	881
9	Reactor Trip Breaker "B" & Reactor Trip Bypass Breaker "A"	46
10	Annunciator Panel (RTGB)	956
11	Waste Disposal Panel	351
12	Diesel Generator "B" Control Panel	950
13	Turbine Emergency Trip	711
14	Gas Stripper Panel "B"	174
15	Gas Analyzer Panel	319
16	Aux . Panel "G-C" Fuse Panel	956
17	Generator Lockout Relay 86 BU	913
18	Reactor Protection Train "B"	956
19	Reverse Current Valves	740
20	Safeguards Train "B"	956
21	Drumming Room Control Panel	378
22	Distribution Panel "B-1"	
23	Steam Driven AFW Pump Control Power	630
24	Rod Drive M-G Set "B"	73

RPS-09 006

Given the following plant conditions:

- The unit is at 100% power
- All equipment is operational
- Maintenance requests to de-energize 125 VDC panels one at a time for testing

Which ONE (1) of the following describes the control power supply for Reactor Trip Breaker "B" and Reactor Trip Bypass Breaker "B"?

	Reactor Trip "B"	Reactor Trip Bypass "B"
A.	"A" 125 VDC Dist. Panel	"A" 125 VDC Dist. Panel
B.	"A" 125 VDC Dist. Panel	"B" 125 VDC Dist. Panel
C.	"B" 125 VDC Dist. Panel	"B" 125 VDC Dist. Panel
✓D.	"B" 125 VDC Dist. Panel	"A" 125 VDC Dist. Panel

Question: 53

Given the following conditions:

- The unit is in Hot Standby.
- A change in boron concentration from 500 ppm to 470 ppm is required.

Given the supplied references, which ONE (1) of the following identifies approximately how many gallons of primary water must be added to make this change?

a. 70 gallons

b. 90 gallons

c. 3000 gallons

d. 4500 gallons

Answer:

c. 3000 gallons

QUESTION NUMBER: 53

TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1

K/A: 004A4.04

Ability to manually operate and/or monitor in the control room: Calculation of boron concentration changes

K/A IMPORTANCE: ***RO*** 3.2 ***SRO*** 3.6

10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: CVCS-10

EXPLAIN the operation of the CVCS.

REFERENCES: OP-301
Plant Curve 5.7

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number CVCS-10 005

JUSTIFICATION:

a. Plausible since this value would be obtained if Curve 5.3 (boron addition - hot) were used, but Curve 5.7 is to be used.

b. Plausible since this value would be obtained if Curve 5.4 (boron addition - hot) were used, but Curve 5.7 is to be used.

c. **CORRECT** Using Curve 5.7 (dilution - hot), a line drawn through 500 ppm coolant boron and 30 ppm dilution will intersect 3000 gallons dilution required.

d. Plausible since this value would be obtained if Curve 5.8 (dilution - cold) were used, but Curve 5.7 is to be used.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Application of given data to plant curves to determine required dilution

REFERENCES SUPPLIED: Plant Curves 5.3, 5.4, 5.7, 5.8

8.2.2.2 (continued)

CAUTION
When the Reactor is subcritical, positive reactivity changes shall **NOT** be made by more than one method at a time.

- Determine the rate and magnitude of the RCS Boron concentration change required to accomplish the desired reactivity change.
- **IF** dilution is for conditions other than normal plant operations, such as fuel depletion or small RCS Tavg adjustments, **THEN** estimate the total volume of dilution water required from the proper dilution nomograph **OR** POWERTRAX.
- For large additions estimate expected PWST level decrease for target dilution.

b. Place the RCS MAKEUP MODE selector switch in DILUTE.

c. **IF** desired, **THEN** place controller FCV-114A, PRIMARY WTR FLOW DILUTE MODE, in MAN **AND** adjust the Controller as follows:

1) Using the UP/DOWN arrow pushbuttons, adjust FCV-114A Controller output to 30-50%.

d. Set the PRIMARY WTR TOTALIZER, YIC-114, to the desired quantity as follows:

1) Depress BUTTON "A".

2) Depress "CLR" BUTTON.

3) Key in the desired quantity **AND** depress the "ENT" BUTTON.

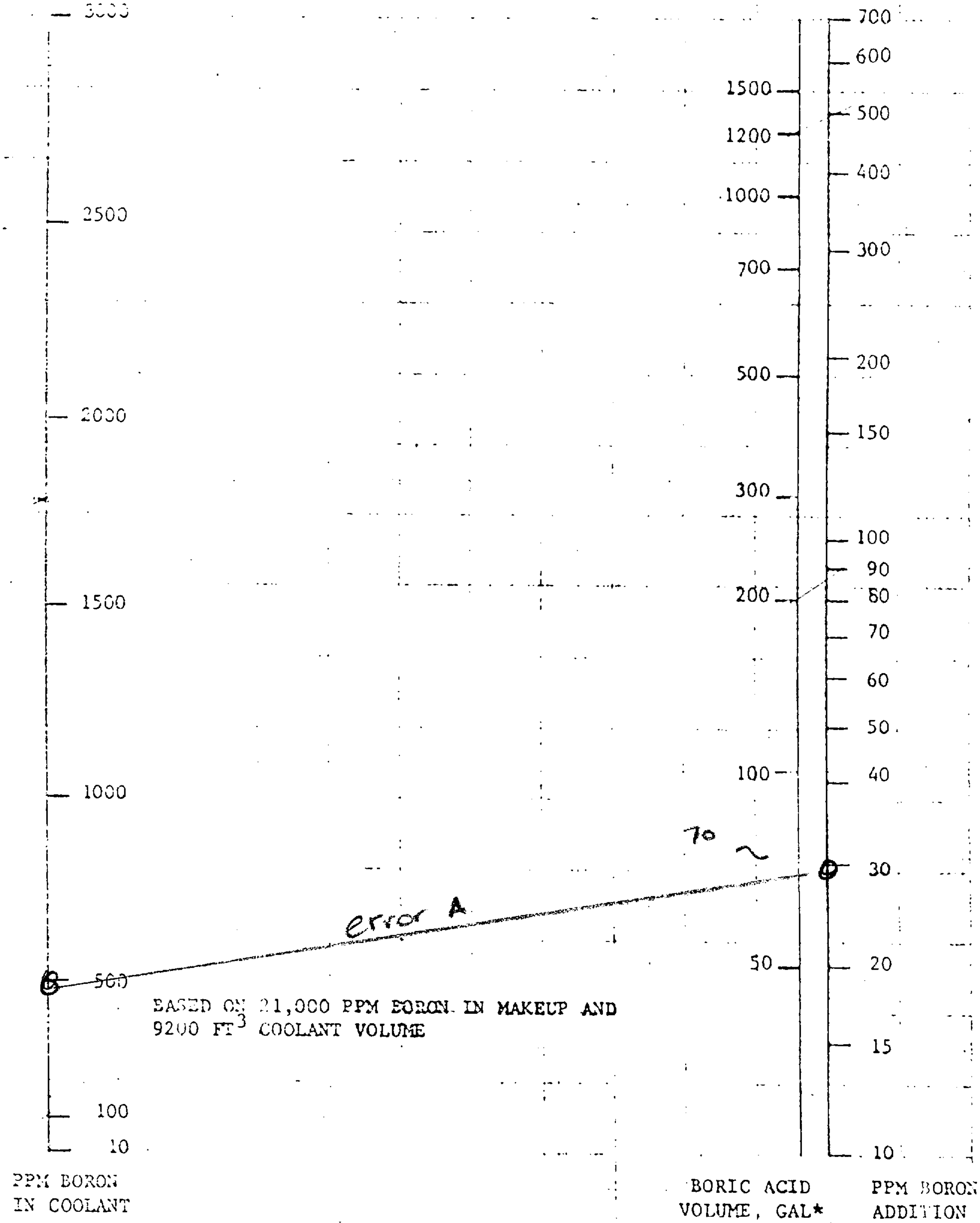


FIGURE S-3.1-3 BORON ADDITION - COOLANT HOT (~580°F)

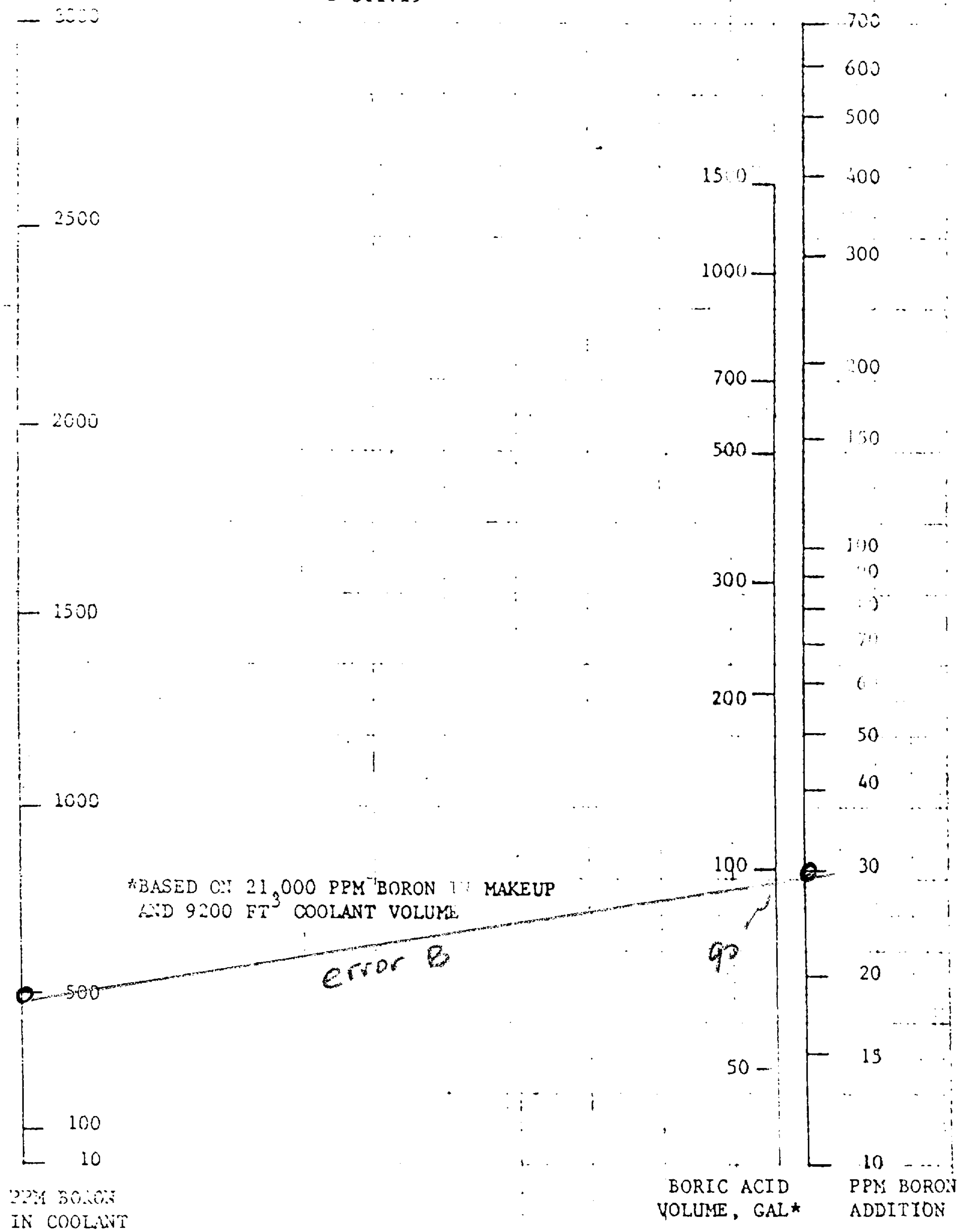


FIGURE S-3.1-4 BORON ADDITION - COOLANT COLD (~100°F)

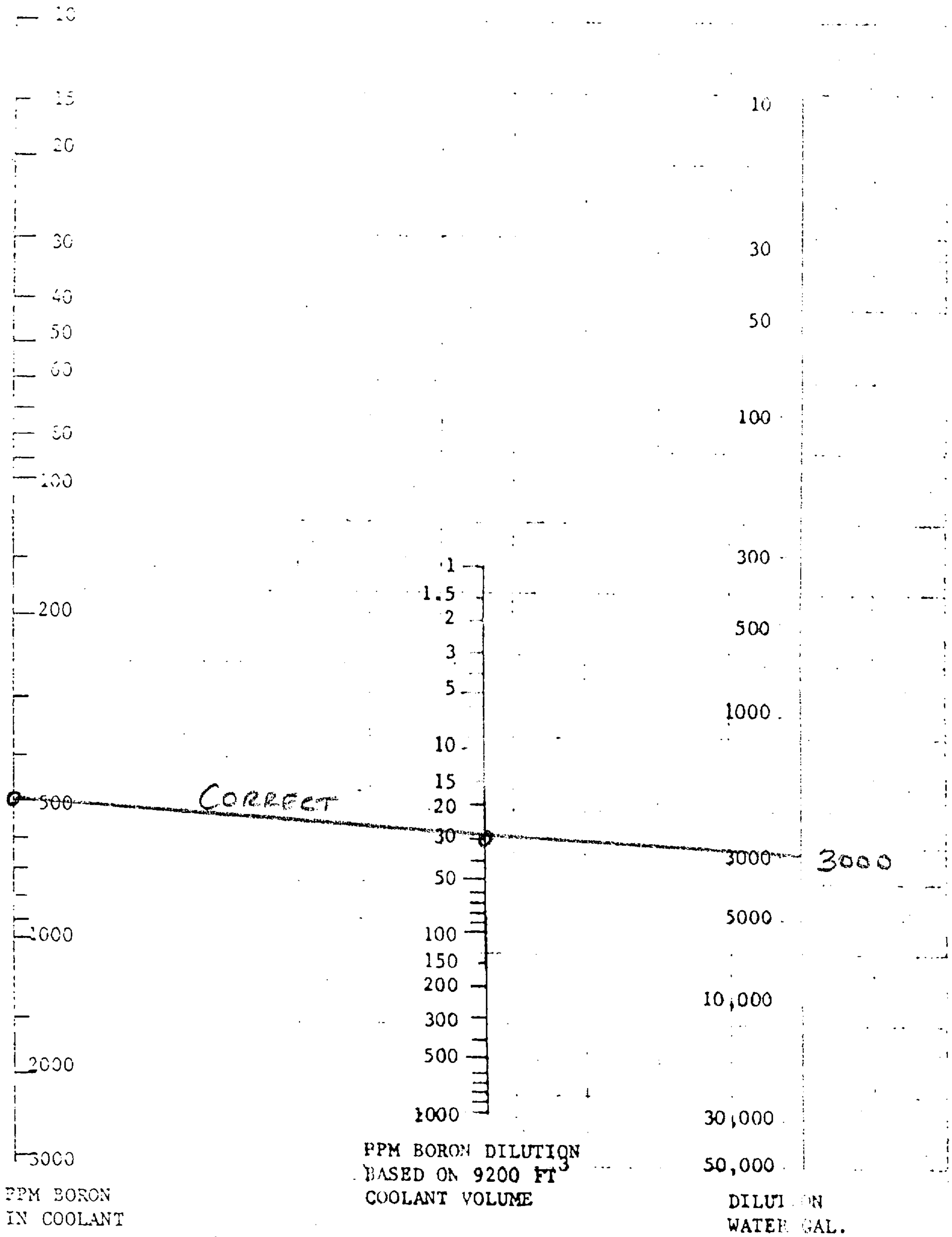


FIGURE S-3.1-7 DILUTION NOMOGRAPH - COOLANT HOT (-580°F)

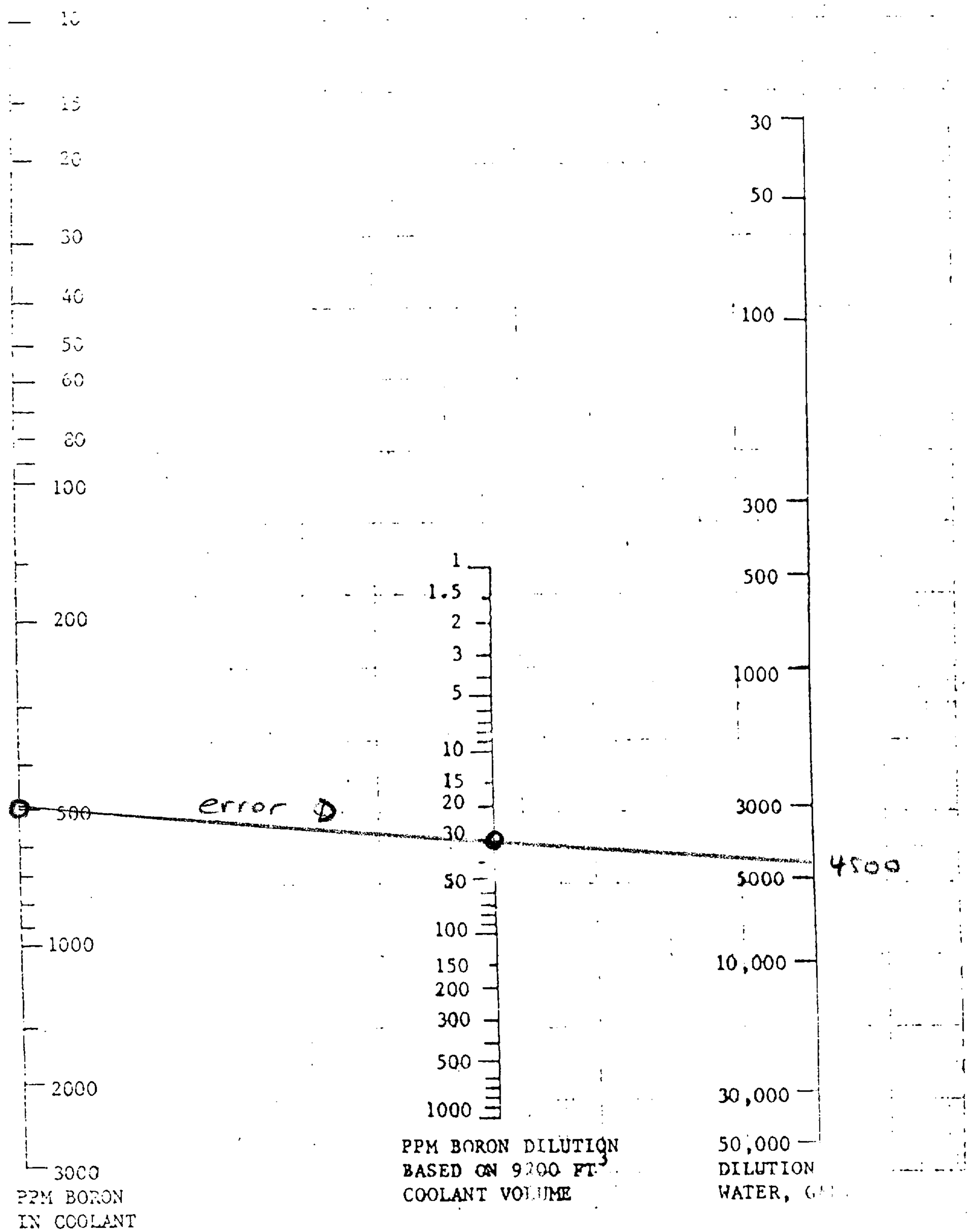


FIGURE S-3.1-8 DILUTION NOMOGRAPH - COOLANT COLD (~100°F)

CVCS-10 005

A power change is going to be made that requires a change in boron concentration from 900 ppm to 810 ppm. Using the attached Nomographs, which ONE (1) of the following is the amount of primary water that will be added to make this change?

A. 3500 gallons

✓B. 5500 gallons

C. 7500 gallons

D. 9500 gallons

Question: 54

Given the following conditions:

- Unit 2 is being ramped to 100% following a refueling outage.
- The following Plant Parameters are noted:

PARAMETER	VALUE
Loop 'A' Tavg	576°F
Loop 'B' Tavg	575°F
Loop 'C' Tavg	576°F
NI-41	100.0%
NI-42	99.0%
NI-43	99.0%
NI-44	100.0%
Loop 'A' ΔT	58.2°F
Loop 'B' ΔT	57.8°F
Loop 'C' ΔT	58.2°F
Loop 'A' Steam Flow	3.40×10^6 lbm/hr
Loop 'B' Steam Flow	3.40×10^6 lbm/hr
Loop 'C' Steam Flow	3.45×10^6 lbm/hr
Loop 'A' Feed Flow	3.40×10^6 lbm/hr
Loop 'B' Feed Flow	3.40×10^6 lbm/hr
Loop 'C' Feed Flow	3.50×10^6 lbm/hr
1st Stage Press (446)	545 psig
1st Stage Press (447)	546 psig
Generator Output	730 Mwe

Given the supplied references, reactor power is ...

a. 99.5%. The power ramp may continue until the plant is at 100%.

b. 99.5%. Power should be held constant to perform a calorimetric.

c. greater than 100%. Power should be held constant to perform a calorimetric.

d. greater than 100%. Power should be immediately lowered.

Answer:

d. greater than 100%. Power should be immediately lowered.

QUESTION NUMBER: 54

TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2

K/A: 002K5.10

Knowledge of the operational implications of the Relationship between reactor power and RCS differential temperature.

K/A IMPORTANCE: ***RO*** 3.6 ***SRO*** 4.1

10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: GP-005-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in GP-005 by explaining the basis of each.

REFERENCES: GP-005

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number GP-005-03 017

JUSTIFICATION:

a. Plausible since NIS average is 99.5%, but other indications indicate power is above 100%.

b. Plausible since NIS average is 99.5%, but other indications indicate power is above 100%.

c. Plausible since indications other than NIS indicate plant is above 100%, but power must be reduced to highest value at or below 100% before calorimetric is performed

d. **CORRECT** .All indications other than NIS indicate plant is above 100%, which requires immediate reduction to maintain at or below 100%.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of conflicting power indications to determine actual power and required actions

REFERENCES SUPPLIED: GP-005, Attachment 10.1

ATTACHMENT 10.1
Page 1 of 1
REACTOR POWER ASCENSION INDICATOR LOG

AVG PWR % (1)	NI-35 amps	NI-36 amps	NI-41A %	NI-42A %	NI-43A %	NI-44A %	LOOP ΔT °F (1)	LOOP 1 ΔT °F	LOOP 2 ΔT °F	LOOP 3 ΔT °F	1st STAGE PRESS psig (1)	PI-446 **OR** 447 psig (2)	NET MWe MAX (1)	NET MWe	CCP % PWR (3)	NR-45 (4)	SSO (1)
15-20							9-11.5				68-90		73				
25-30							14.5-17				113-135		153				
35-40							20-23				158-180		235				
45-50							26-28.5				207-230		316				
55-60							32-34.5				261-285		398				
65-70							37-40				320-345		480				
75-80							43-46				384-410		562				
85-90							49-51.5				449-475		643				
95-100							55-57.5				513-540		725				

(1) Listed ranges and Net MWe maximums are predicted based on past plant performance. The maximum value of each indication is the maximum target value for each power increase. The SSO shall initial if plant management has determined that indications are acceptable to continue with the power escalation.

(2) Use indicator that corresponds to the channel selected on the 1st STAGE PRESSURE selector switch.

(3) Record Continuous Calorimetric Program % Power.

(4) Verify NR-45 is selected to the highest reading channel.

INIT

8.5.29 **WHEN** the highest indicator of Reactor Power listed on Attachment 10.1 indicates 90% power, **OR** as directed by the Reactor Engineer, **THEN** depress the HOLD pushbutton **AND** maintain indicated power. ____

8.5.30 **IF** this power escalation is the initial power escalation to 90% following a core alteration, **THEN** perform the following: (SOER 90-003)

1. Stabilize reactor power between 87% and 90%, **OR** as directed by the Reactor Engineer. ____
2. Perform physics tests as directed by the Reactor Engineer. ____
3. Verify NIS Power Range High Level Trip is set per the Reactor Engineer. ____

8.5.31 Perform OST-010, Power Range Calorimetric. ____

8.5.32 **IF** all indications of Reactor Power agree within 5% of each other, **OR** management approval has been obtained, **THEN** perform the following:

1. Adjust the SETTER indication using the REF ▿ and/or REF ▵ pushbuttons to indicate no greater than 100.0 load. ____
2. Depress the GO and/or HOLD pushbuttons **AND** the REF ▿ and/or REF ▵ as necessary to continue the load increase to less than or equal to 100% Reactor Power, **OR** as directed by the Reactor Engineer **OR** SSO. ____

8.5.33 **WHEN** the highest indicator of Reactor Power listed on Attachment 10.1 indicates 100% power, **OR** the maximum power as directed by the Reactor Engineer **OR** SSO, **THEN** depress the HOLD pushbutton **AND** maintain indicated power. ____

Question: 55

Given the following conditions:

- A Temporary Change (TC) to Revision 44 of OP-305, Boron Recycle Process, was issued on March 1, 2001.
- Revision 45 of OP-305 was issued on March 6, 2001.
- The Temporary Change was **NOT** incorporated into Revision 45, but was cancelled and subsequently reissued (using a new TC number) with the issue of Revision 45.

The Temporary Change now expires on ...

a. March 15, 2001.

b. March 20, 2001.

c. March 22, 2001.

d. March 27, 2001.

Answer:

c. March 22, 2001.

QUESTION NUMBER: 55
TIER/GROUP: ***RO*** 3 ***SRO*** 3
K/A: 2.2.11

Knowledge of the process for controlling temporary changes.

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 3.4
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: AP-022-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in AP-022 by explaining their basis.

REFERENCES: AP-022

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. Plausible if misconception is that expiration is 14-day instead of 21-day since this date would be determined based on original issue date of TC.

b. Plausible if misconception is that expiration is 14-day instead of 21-day and expiration clock is reset, but date is based on original issue date of TC.

c. **CORRECT** Reissue of the same TC, even under a different number, requires that the 21-day clock for expiration of the TC be based on the original issue date of the TC.

d. Plausible since expiration is 21-day, but date is based on original issue date of TC and not reset to reissue date.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Calculation of temporary change expiration based on knowledge of administrative requirements

REFERENCES SUPPLIED:

8.3.3.2(Continued)

d. When revising a document to delete requirements, a review of all references, as well as the document's historical file should be performed to allow appropriate consideration of past revisions which may have incorporated significant commitments or resolutions to problems or requirements. Regulatory requirements and commitments must be addressed elsewhere prior to deleting requirements/commitments or voiding a procedure. The location of the regulatory requirements/commitments are to be noted in the Summary of Changes of the voided procedure.

e. If a procedure is being written or revised to add comments or requirements, note in the Reason for Change.

f. Changes to the document may be highlighted, red-lined, or typed using a different font to facilitate reviews. These must be removed by the **Writer** prior to submitting the document for distribution.

g. If a procedure number needs to be changed, the Sponsor/Writer must first delete the existing procedure.

h. If not previously filled out, enter information into the "Revision is a result of an ESR/DCF/CR/Other" field as applicable.

EXAMPLE: If the revision is a result of an ESR, list the ESR Number and Revision in this field.

i. Incorporate any Temporary Changes intended to be made permanent. If the Temporary Change cannot be included in this permanent revision, cancel the present Temporary Change and reissue it against the new approved permanent revision or cancel the TC. The Temporary Change will have a new TC Number and the time clock will have **only the remainder** of the original 21 days if reissued.

Question: 61

Given the following conditions:

- A licensed operator who has an inactive license has been performing administrative duties in the Training Section for twelve (12) months.
- He is returning to Operations and is to be placed back on shift.
- All licensed operator continuing training and fire brigade qualifications are current.

Which ONE (1) of the following are the additional **MINIMUM** requirements for returning his license to an active status?

a. Complete **FOUR** normal shifts, including shift turnovers IAW plant procedures, before and after each watch, and review all the procedure changes for the past three (3) months

b. Complete **FOUR** normal shifts, including shift turnovers IAW plant procedures, before and after each watch, and conduct a complete plant tour

c. Complete **FIVE** normal shifts, including shift turnovers IAW plant procedures, before and after each watch, and review all the procedure changes for the past three (3) months

d. Complete **FIVE** normal shifts, including shift turnovers IAW plant procedures, before and after each watch, and conduct a complete plant tour

Answer:

b. Complete **FOUR** normal shifts, including shift turnovers IAW plant procedures, before and after each watch, and conduct a complete plant tour

QUESTION NUMBER: 61
TIER/GROUP: ***RO*** 3 ***SRO*** 3
K/A: 2.1.3

Knowledge of shift turnover practices.

K/A IMPORTANCE: ***RO*** 3.0 ***SRO*** 3.4
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: OMM-001-05-03

DISCUSS each section of OMM-001-05, when possible, using the information given in each section of the procedure.

REFERENCES: OMM-001-05

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number 10CFR-55.13-22 001

JUSTIFICATION:

a. Plausible since watchstanding requirements are correct, but must also perform a complete tour of the plant.

b. **CORRECT** Four complete 12-hour watches, plus shift turnovers, and a complete tour of the plant must be completed.

c. Plausible since this would satisfy watchstanding requirements, but must also perform a complete tour of the plant.

d. Plausible since this would satisfy all requirements, but is not the minimum requirement.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of administrative requirements for activating an inactive license

REFERENCES SUPPLIED:

8.2 Inactive Status

8.2.1 If a licensee has **NOT** been actively performing the functions of an operator or senior operator for the periods defined in Section 8.1, then the individual's license is declared inactive and the licensee may **NOT** resume activities authorized by a license issued under 10CFR55.

8.2.2 Inactive licensees may still fulfill the functions of Fire Brigade Team Leader and Fire Brigade Member if all fire brigade training is current, act in the capacity of WCC SCO or WCC CO and perform WCC functions, or perform valve manipulations and independent verifications with approved procedures.

8.2.3 If a non-licensed watchstander has **NOT** been actively performing the functions for which he/she is qualified for the periods defined in Section 8.1, the watchstander may **NOT** resume activities authorized by their qualifications.

8.2.4 Inactive non-licensed watchstanders may still fulfill the functions of Fire Brigade Member (if all fire brigade training is current) or perform valve manipulations and independent verifications with approved procedures.

8.3 Reactivation of a Licensed Watchstander

8.3.1 **IF** an individual's license becomes inactive status, **THEN** before resumption of activities authorized by a license issued under 10CFR55, the Manager - Operations shall certify, using Attachment 10.2, that qualifications and status of the licensee are current and valid as discussed in Section 8.7, **AND** that the licensee has completed a minimum of 48 hours of shift functions in the position in which the individual will be qualified.

8.3.2 **IF** an individual has an active license and it is desired to reactivate his qualification at another watchstation, **THEN** the individual must complete a minimum of 48 hours of shift functions in that watchstation. Attachment 10.2 Steps 1 and 4 are used to document the watches stood and Steps 2 and 3 of the attachment should be marked N/A.

8.3.3 The following guidelines apply regarding the required number of watches stood:

1. The 48 hours shall be performed as four complete shift watches including shift turnovers IAW plant procedures before and after each watch under the direction of an individual qualified to stand that watchstation.

2. The 48 hours shall include a complete tour of the plant as defined in Section 8.5.

3. The 48 hours of reactivation time should take place over a maximum period of four weeks **AND** shall occur in the same calendar quarter.

8.3.4 An inactive SRO may reactivate as an SRO limited to fuel handling duties by standing 8 hours under instruction from an active licensed SRO performing fuel handling duties. The 8 hours are not required to be consecutive.

8.3.5 Inactive licensees may stand watch on non-licensed watchstations provided ONE of the following conditions are met:

- The individual has reactivated their license IAW Section 8.3

- The individual stands one 12-hour watch, including shift turnovers and a complete set of logs, under the supervision of an active watchstander for the particular non-licensed watchstation(s) **AND** completes Attachment 10.3

- Manager - Operations waives requirement to reactivate for the particular non-licensed watchstation(s) **AND** documents on Attachment 10.3

8.3.6 Completed Attachments 10.2 and 10.3 shall be routed to the Operations Scheduler who will revise Attachment 10.4 to reflect the reactivation of qualification.

8.4 Reactivation of a Non-Licensed Watchstander

8.4.1 **IF** active status, as defined herein, is not met, **THEN** before resumption of watchstanding activities, the Manager - Operations shall certify using Attachment 10.3 that <u>ONE</u> of the following conditions are met:

- The individual stands one 12-hour watch, including shift turnovers and a complete set of logs (N/A for STA), under the supervision of an active watchstander for the particular non-licensed watchstation(s) **AND** completes Attachment 10.3

- Manager - Operations waives requirement to reactivate for the particular non-licensed watchstation(s) **AND** documents on Attachment 10.3

8.4.2 The completed Attachment 10.3 shall be routed to the Operations Scheduler who will revise Attachment 10.4 to reflect the reactivation of qualification.

8.5 Plant Tours (For Reactivation Purposes Only)

8.5.1 The plant tour required for reactivation of NRC issued licenses shall be conducted under the supervision of an appropriate active license holder **AND** shall consist of the entire plant inside the protected and vital areas EXCEPT the following areas:

- Containment Vessel

- RHR Pit

- Office buildings that are not part of the watchstanders normal tour

- Areas specifically excluded in writing by the SSO in the Comments section of Attachment 10.2.

8.5.2 The supervising license holder is not required to accompany the reactivating individual. However, the reactivation tour shall be conducted with the cognizance of, and under the direction of, the supervising license holder as appropriate.

8.5.3 Plant tours shall include entry and visual surveillance of each room in the Turbine Building and RCA that is not specifically excluded by this section or the SSO. Room entry is not necessary for rooms that can be visually checked through cage doors with confidence that OMM-001-11 requirements are satisfied.

8.5.4 Plant tours for reactivation of non-licensed watchstations shall consist of a complete set of rounds and logs for the position being reactivated unless waived by the Manager - Operations IAW Section 8.4.

8.6 Crew Rotations and Reassignments (CR 96-02954)

8.6.1 Reactivation of an individual's qualifications or the addition of newly licensed individuals may require crew rotations and reassignments. Prior to making a crew rotation or adding personnel to a shift, the Manager - Operations shall consider the following:

- The experience level of each crew member
- The composite experience levels to achieve a balanced crew
- Personality conflicts
- Maturity level of shift members
- Leadership ability of individuals

The above considerations ensures a nucleus of experienced personnel are maintained.

8.6.2 Crew rotations are typically made following refueling outages or soon after a new group of candidates receive their licenses. For newly licensed individuals, they normally remain as an extra person on the assigned shift. This allows the opportunity for the individual to attend Licensed Operator Continuing Training with the crew and integrate with the crew prior to assuming a licensed position. Other rotations are normally made between training cycles. This allows the crews to change and then be trained as a team at the earliest available opportunity.

8.7 Qualification Documentation

8.7.1 Automatic notifications made in PQD ensure that personnel are aware of the status of qualification expiration dates. PQD information for the following qualifications is maintained on the LAN. (CR 96-01883)

- NRC License and Medicals
- Fire Brigade Qualifications and Medicals (CR 96-00729)
- Respirator Qualifications and Medicals

8.7.2 Watchstander qualifications should be considered active if the following criteria are met:

- License holders and STAs satisfactorily participate in the Licensed Operator Continuing Training program.
- Auxiliary Operators satisfactorily participate in the AO Continuing Training Program.
- The required number of watch hours are satisfied IAW Section 8.10 as documented on Attachment 10.1, Proficiency Watch Tracking Sheets.
- Watchstanders are respirator qualified in at least one type of respiratory protection.
- Fire Brigade training and qualifications are current for Fire Brigade members and Team Leaders.

8.7.3 Licensed watchstanders should be qualified as a member of the Fire Brigade to reactivate. **IF** the individual was previously qualified as a Team Leader, **THEN** it is desired (but not required) that the individual be qualified as Team Leader prior to reactivation. The total number of qualified Team Leaders should not exceed 35 at any one time.

8.7.4 Reactivation of watchstander qualifications are verified using Attachment 10.2 or 10.3 as appropriate **AND** the reports stated above.

8.9.4 The Supervisor **SHALL** notify the Manager - Operations of the situation.

8.9.5 Upon being contacted by the licensed operator, the Examining Physician will determine the need for a clinical assessment of the potential medical restriction. If a clinical assessment is required, a determination of the licensee's medical condition will be made based on the available medical records, the results of the clinical assessment, or any special testing performed. Any documentation or notifications will be IAW SAF-NGGC-2171.

> **NOTE:** SEC-NGGC-2130 states: "Any worker with unescorted access shall notify his or her supervisor or designee of (1) any arrest or (2) any incident that may impact the worker's trustworthiness or reliability, in accordance with SEC-NGGS-2101 Nuclear Worker Screening Program for Unescorted Access."
>
> SEC-CPL-025 has additional directions and forms concerning reporting arrests.

8.9.6 Upon being notified of a licensee felony conviction **OR** upon being notified by the Examining Physician that a licensee has a medical condition exists that invalidates or restricts the individual's license, the Manager - Operations **SHALL** notify Training and Regulatory Affairs.

8.9.7 Control Room personnel who wear corrective lenses as a condition of their license **SHALL** maintain a pair of SCBA glasses in the Control Room. Each shift has a drawer containing trays in which to store the glasses.

8.9.8 Control Room personnel whose respiratory protection qualifications permit the wearing of contact lenses, **MAY** wear soft, gas-permeable contact lenses with their respiratory protection face pieces.

8.10 Tracking Watchstander Hours

8.10.1 At the **END** of each shift, the SSO **OR** CRSS should update attachment 10.1 by initialing and dating the appropriate watch block for the Licensed and Non-Licensed watchstander that stood a watch.

- **DO NOT** make any entry **UNLESS** the watchstander stood the entire 12 hour watch. A watch must be a minimum of 12 hours in length in order to count as a watch used to maintain active watchstation status.

- **AFTER** the minimum number of watches has been completed for the quarter, no further entries for that individual are required **UNTIL** the next calendar quarter.

LICENSED WATCHSTANDER REACTIVATION

NAME: ____ SSN: ____________________

1. Watchstanding Log:

DATE	POSITION	HOURS	TURNOVER YES/NO	ACTIVE WATCHSTANDER SIGNATURE

2. The following are current:
   - Licensed Operator Continuing Training requirements.
   - Fire Brigade qualification (Yes/No)
     circle one
   - Team Leader qualification (Yes/No)
     circle one

______________________________ __________
Superintendent - Operator Training Date

3. Respirator qualified
   (expiration date: __________)

______________________________ __________
E&RC Date

4. Plant tour completed.

______________________________ __________
Superintendent Shift Operations Date

Comments: ____

____

I certify that the qualifications, status, training, and watchstanding requirements of the above named licensed individual are current and valid, allowing return to active license status for the position assigned.

______________________________ __________
Manager - Operations Date

Route completed form to:
- Operations Scheduler
- Operations Tech Aide

Question: 62

Given the following conditions:

- The unit is operating at 100% power.
- RCS Tavg is 575.4°F.
- PZR level is 53%
- VCT level is 23" and stable.
- Letdown flow is 45 gpm (FI-150).
- RCP seal injection flows are:

RCP	SEAL INJ
'A'	8.3 gpm
'B'	7.9 gpm
'C'	7.8 gpm

Which ONE (1) of the following would be the expected flow indication on FI-122A, Charging Header Flow, assuming **NO** RCS leakage?

a. 21 gpm

b. 30 gpm

c. 36 gpm

d. 54 gpm

Answer:

b. 30 gpm

QUESTION NUMBER: 62

TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1

K/A: 004A1.11

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: Letdown and charging flows

K/A IMPORTANCE: ***RO*** 3.0 ***SRO*** 3.0

10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: CVCS-05

DESCRIBE the performance and design attributes of the major CVCS components.

REFERENCES: AOP-016
SD-021

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number CVCS-03 010

JUSTIFICATION:

a. Plausible if misconception is that seal leakoff flow is ignored, but leakoff flow is not required to be made up. 45 - 24 = 21.

b. **CORRECT** Charging flow should equal letdown flow (105 gpm) less seal injection flow (24 gpm) plus seal return flow (9 gpm). 45 - 24 + 9 = 30.

c. Plausible if misconception that seal injection flow is measured as part of charging flow and seal leakoff must be subtracted, but seal injection is required to be included. 45 - 9 = 36.

d. Plausible if misconception that seal injection flow is measured as part of charging flow, but seal injection is required to be included. 45 + 9 = 54.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Calculation of expected charging flow indication based on given CVCS parameters

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

OST-051, Reactor Coolant System Leakage Evaluation, is the preferred method of leak rate determination if plant conditions permit.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
26.	Initiate Leak Rate Determination Using One Or More Of The Following Methods: • OST-051, Reactor Coolant System Leakage Evaluation OR • OST-901, HVH Condensate Measuring System OR • Charging versus letdown flow balance	
27.	Check R-17, COMPONENT COOLING WATER RADIOACTIVE LIQUID - INCREASING OR IN ALARM	Go To Step 29.
28.	Go To AOP-014, Component Cooling Water System Malfunction	
29.	Check Refueling Cavity Status - FLOODED	Go To Step 41.
30.	Check Status Of Fuel Handling Activities Or Reactor Vessel Internals Movement - IN PROGRESS	Go To Step 32.

seal water flow.

Prior to or during the heatup process the CVCS is employed to obtain the correct chemical properties in the RCS. Reactor coolant makeup control is operated on a continuous basis to replace system leakage. Chemicals are added via the chemical mixing tank as required to control reactor coolant chemistry such as pH and dissolved oxygen content.

During a plant startup and power ascension, boron concentration is usually reduced due to the increased negative reactivity addition from power defect and the increase in Xenon and Samarium. For planned plant power changes, Reactor Engineering provides the Control Room with the necessary blended additions to the VCT to accomplish the power change. For instances when Reactor Engineering has not provided the calculated blends, the Reactor Operator must calculate the actual amount of water used to dilute the RCS and account for this. Nomographs and figures such as Figures 18 to 21 of this System Description are used to perform these calculations.

6.2 Normal Operation

Normal Operation includes operation at power and hot zero power.

6.2.1 Letdown and Charging

During normal operation at a constant power level, the letdown flow is equal to the sum of the charging flow (passing through the tubes of the regenerative heat exchanger) and the flow through the thermal barrier of the reactor coolant pumps. The letdown flow is controlled by means of the letdown orifices in the letdown line (under normal condition the 45 gpm orifice is employed), and cooled by the charging line flow in the regenerative heat exchanger.

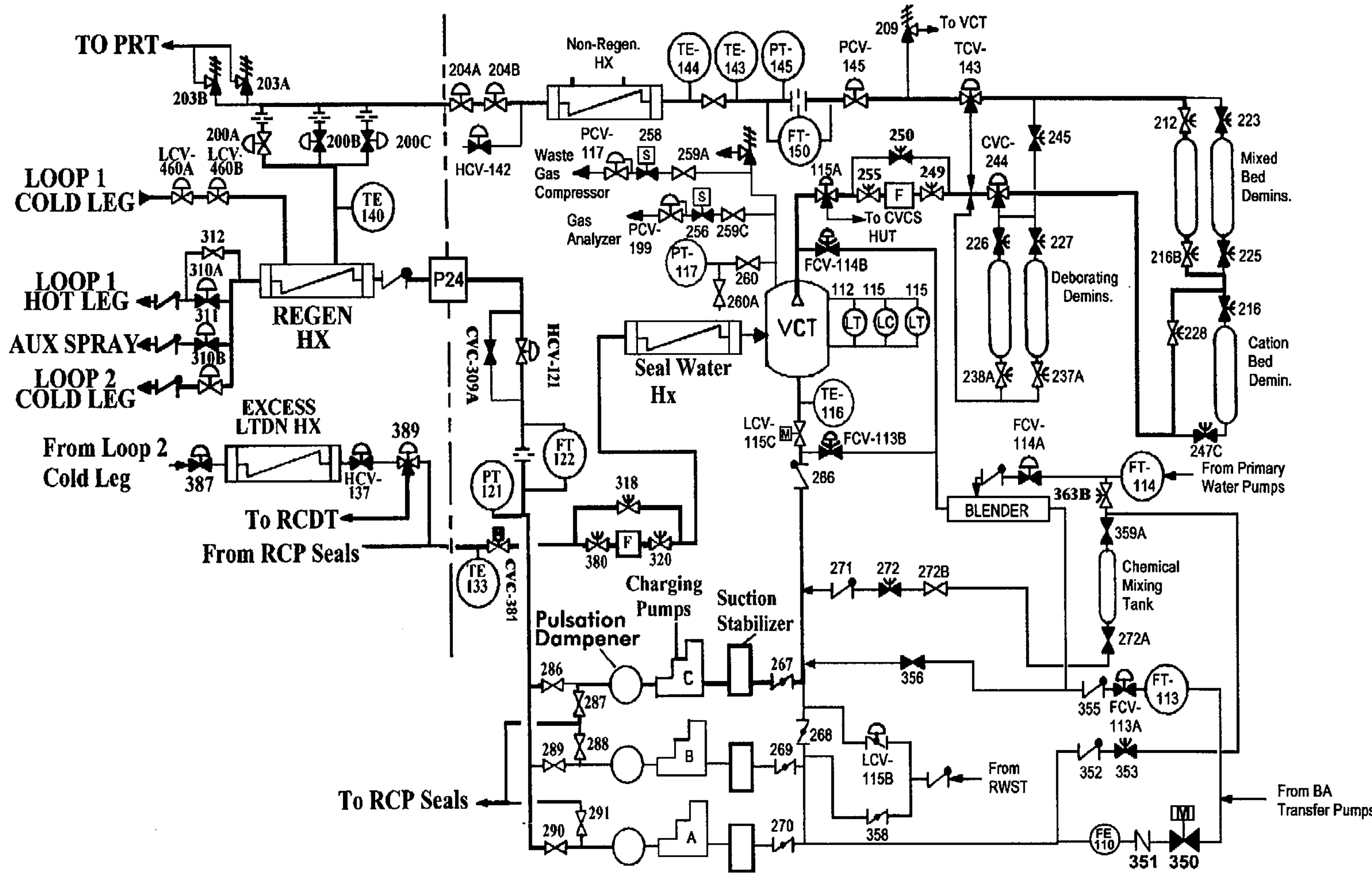
It then flows to the nonregenerative heat exchanger where it is cooled to approximately 105°F by component cooling water. On leaving the nonregenerative heat exchanger the letdown normally flows through one of the two mixed bed demineralizers, through the reactor coolant filter, and then via a spray nozzle into the VCT. Hydrogen pressure in the volume control bank insures that coolant returning to the RCS has the appropriate hydrogen concentration for oxygen control.

From the volume control tank the charging pump delivers the reactor coolant at about 105°F to the tube side of the regenerative heat exchanger. The coolant is heated to approximately 493°F at the exit end. The coolant is then injected into the RCS cold leg.

A portion of the flow from the charging pump passes through the reactor coolant pump seals. The charging pump speed (flowrate) is controlled automatically to maintain the pressurizer water level at the setpoint programmed for the RCS average temperature at

CVCS FLOW DIAGRAM

CVCS-FIGURE-1 (Rev . 1)



INFORMATION USE ONLY

CVCS-03 010

Given the following plant conditions:

- Mode 1 at 100% RTP
- RCS Tavg = 575.4
- PZR level = 53%
- VCT level = 23 " and steady
- Letdown flow is 60 gpm (FI-150)
- RCP seal injection flows: 8.3 gpm (A) 7.9 gpm (B) 7.8 gpm (C)

Which ONE (1) of the following would be the expected flow indication on FI-122A, Charging Header Flow?

A. 24 gpm

✓B. 45 gpm

C. 60 gpm

D. 69 gpm

Question: 63

The following personnel are entering the RCA to perform plant related activities:

1. Two operators doing a valve lineup in the RCA expect to receive a dose of about 125 mrem each.
2. Operators doing routine radwaste processing.
3. Electrical maintenance workers cleaning and inspecting an MCC breaker in the RCA.

Which ONE (1) of the following identifies ALL of the above activities which can be performed using a General RWP in accordance with HPP-006, "Radiation Work Permits"?

a. 1 and 2 **ONLY**

b. 1 and 3 **ONLY**

c. 2 and 3 **ONLY**

d. 1, 2, and 3

Answer:

c. 2 and 3 **ONLY**

QUESTION NUMBER: 63
TIER/GROUP: ***RO*** 3 ***SRO*** 3
K/A: 2.3.2

Knowledge of facility ALARA program.

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 2.9
10CFR55 CONTENT: ***55.41(b) RO*** 12 ***55.43(b) SRO***

OBJECTIVE: 10CFR20-04

Recognize how the practical aspects of the radiation protection program will be effected.
a. Surveys
b. Postings
c. Records

REFERENCES: HPP-006

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number HPP-006 001

JUSTIFICATION:

a. Plausible since routine radwaste processing is permissible, but exposure of > 100 mRem requires a Special RWP.

b. Plausible since maintenance activities which are expected to involve minimal radiological consequence are permissible, but exposure of > 100 mRem requires a Special RWP.

c. **CORRECT** Routine radwaste processing and maintenance activities which are expected to involve minimal radiological consequence are permissible. Any task where an individual is expected to receive > 100 mRem require a Special RWP.

d. Plausible since routine radwaste processing and maintenance activities which are expected to involve minimal radiological consequence are permissible, but exposure of > 100 mRem requires a Special RWP.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of administrative requirements for RWP usage

REFERENCES SUPPLIED:

8.10 ALARA Planning

8.10.1 Work activities and associated dose are tracked by work order task and generic tasks.

8.10.2 The RC Planner ensures work order task (s) are transferred from the WORK MANAGEMENT SYSTEM to RIMS and assigned to the correct RWP number. The following criteria is used to evaluate which type of RWP will be assigned to the task or work activity:

1. General RWP(s) are used by individuals entering RCA(s) whose work activities do not require stringent radiation protection controls. General RWP(s) may be used for the following activities unless deemed otherwise by RC Supervision: (SER 88-023).

   a. Radiation Control Surveillance
   Radiation Control surveys, inspections, and activities can be conducted on a general RWP.

   b. Operations Rounds
   Operations activities to support and maintain desired plant conditions.

   c. Radiochemistry Sample Collection and Analysis
   Radioactive sample collection and work within the radiochemistry laboratory by chemistry and support personnel.

   d. Routine Radwaste Processing
   Routine radwaste processing operations.

   e. General Entry into RCA(s)
   Entries made into made into RCA(s) to work on non-contaminated systems.

   f. Planning, Scheduling, Audits, Security, and Inspections
   Entries made by personnel, to include Security, into RCA(s) provided no physical work is accomplished. Personnel supporting activities controlled by a special RWP should utilize the special RWP rather than the general RWP.

g. Maintenance
Entries made into RCA(s) to perform maintenance activities that are expected to involve minimal radiological consequence as determined by RC Supervision or designee.

2. Special RWP(s) are required for specific plant locations/radiological conditions and tasks which require stringent radiation protection controls and are not otherwise covered by the general RWP criteria. Special RWP(s) are not limited to, but will be used for the following activities unless otherwise authorized by RC Supervision or designee (SER 88-023):

   a. Entries into HPA(s) with the exception of Radiation Control, Chemistry and Operations.

   b. Any breaching of a contaminated system other than instrument calibrations/repairs, sampling, and maintenance activities involving minimal radiological consequences.

   c. Abrasive work (e.g. grinding, cutting, machining, or welding) on contaminated surfaces.

   d. Any task where an individual will receive greater than 100 mrem.

   e. Entries into VHRA(s).

   f. With the exception of Radiation Control, Chemistry, Operations, and inspections, all work in:

      1) Areas with contamination levels in excess of 100,000 dpm/100 cm^2 and/or 30 cm radiation levels of 100 mRem/hr or greater

      2) Locked High Radiation Area Entries

      3) With the exception of Radiation Control , Chemistry, and Operations activities, all work and inspections in Airborne Radioactivity Areas.

8.10.3 The RC Planner evaluates the task to determine any radiological protection measures or hold-points that may need to be added to the Special Instructions of the work order task .

Question: 64

Given the following conditions:

- The unit was operating at 100% power.
- All IRPI indication fails to zero with **NO** rod bottom bistable lights.
- A Turbine Runback to 70% has occurred.
- APP-005-A3, PR DROP ROD ROD STOP, is illuminated.

Which ONE (1) of the following procedures should be used to mitigate this plant transient?

a. AOP-001, Malfunction of Reactor Control System

b. AOP-015, Secondary Load Rejection or Turbine Runback

c. AOP-024, Loss of Instrument Buses

d. AOP-025, RTGB Instrument Failures

Answer:

a. AOP-001, Malfunction of Reactor Control System

QUESTION NUMBER: 64
TIER/GROUP: *RO* 1/2 *SRO* 1/1
K/A: 003 2.4.4

Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (Dropped Rod).

K/A IMPORTANCE: *RO* 4.0 *SRO* 4.3
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: AOP-001-02

RECOGNIZE the selected entry level conditions of AOP-001.

REFERENCES:
AOP-001
AOP-015
AOP-024
AOP-025
APP-005

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number AOP-001-02 006

JUSTIFICATION:

a. **CORRECT** Any indication of a malfunction involving rod position indication is addressed by AOP-001.

b. Plausible since a runback has occurred, but entry into AOP-015 would be caused by an NIS failure not an IRPI failure.

c. Plausible since a loss of power to the rod position indication has occurred, but entry into AOP-024 is excluded for a loss of the instrument bus for rod position indication.

d. Plausible since rod position indiction is located on the RTGB, but entry should be made into AOP-001.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 2

Knowledge of entry requirements / purpose of AOPs

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides the instructions necessary for the Operator to recover a dropped rod, realign a misaligned rod, stop abnormal continuous rod motion and operate with an IRPI failure.

This procedure is applicable in Modes 1, 2, and 3.

2. ENTRY CONDITIONS

Any indication of a dropped rod, misaligned rod, unwarranted rod motion, inability to move rod(s) or suspected IRPI malfunction.

- END -

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

The purpose of this procedure is to provide instructions to stabilize plant conditions following a secondary load rejection or Turbine runback.

> **NOTE**
>
> Entry to AOP-015 is NOT required if a PR NIS failure occurs AND a runback fails to actuate.

2. ENTRY CONDITIONS

a. This procedure is entered upon a secondary load rejection or Turbine runback caused by a failure of a PR NIS.

b. When directed from AOP-026, Low Frequency Operation.

- END -

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   The purpose of this procedure is to provide instructions to be followed in the event of a loss of power to any Instrument Bus (excluding the Instrument Bus for RPI).

   This procedure is applicable under Modes 1, 2, and 3.

2. ENTRY CONDITIONS

   This procedure is entered on any indication of a Loss of an Instrument Bus (excluding Instrument Bus for RPI).

- END -

Purpose & Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides instructions for failure of process variable transmitters which provide input to RTGB controllers.

IF an applicable transmitter fails while the controller is operating in manual OR is being fed from an alternate channel, THEN entry to this procedure is NOT required.

This procedure is applicable in Modes 1, 2, 3, and 4.

2. ENTRY CONDITIONS

Failure of any process variable transmitter which affects automatic operation of RTGB controllers with the following exceptions:

- FT-605, RHR Flow
- LT-115, VCT Level
- LT-112, VCT Level
- PR NIS (NI-41, 42, 43, & 44)

- END -

ALARM

PR DROP ROD ROD STOP

AUTOMATIC ACTIONS

1. Turbine Runback (Load Reference and Load Limit)

CAUSE

1. Dropped Rod
2. Failure of a Power Range Channel

OBSERVATIONS

1. Power Range Indication
2. Generator Output
3. Rod Bottom Lights
4. RPIs

ACTIONS

1. **IF** a dropped rod has occurred, **THEN** refer To AOP-001.
2. **IF** a Power Range channel has failed with the Unit on the line, **THEN** refer to AOP-015.
3. **IF** a Power Range channel has failed with the Unit off the line, **THEN** remove the affected channel from service using OWP-011.

DEVICE/SETPOINTS

1. N-41, N-42, N-43, or N-44 / 5% Power change in 5 sec.

POSSIBLE PLANT EFFECTS

1. Radial Flux Tilt

REFERENCES

1.
2. AOP-001, Malfunction of Reactor Control System
3. AOP-015, Secondary Load Rejection Or Turbine Runback
4. OWP-011, Nuclear Instrumentation (NI)
5. CWD B-190628, Sh 440, Cable BH

Question: 65

Given the following conditions:

- A line break caused the Fire Header pressure to drop.
- Fire Header pressure eventually stabilized at 83 psig.

Which ONE (1) of the following expected fire system responses would have resulted in this condition?

a. The Electric Fire Pump automatically started, then the Diesel Fire Pump automatically started.

b. The Electric Fire Pump automatically started and the Diesel Fire Pump remained in standby.

c. The Diesel Fire Pump automatically started, then the Electric Fire Pump automatically started.

d. The Diesel Fire Pump automatically started and the Electric Fire Pump remained in standby.

Answer:

a. The Electric Fire Pump automatically started, then the Diesel Fire Pump automatically started.

QUESTION NUMBER: 65
TIER/GROUP: *RO* 2/2 *SRO* 2/2
K/A: 086A3.01

Ability to monitor automatic operation of the Fire Protection System including: Starting mechanisms of fire water pumps

K/A IMPORTANCE: *RO* 2.9 *SRO* 3.3
10CFR55 CONTENT: *55.41(b) RO* 4 *55.43(b) SRO*

OBJECTIVE: FPW-09

EXPLAIN the normal operation of the Fire Water control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-041

SOURCE: *New* ☐ *Significantly Modified* ☒ *Direct* ☐

Bank Number FP-05 003

JUSTIFICATION:

a. **CORRECT** The electric fire pump starts at 100 psig and the diesel fire pump starts at 90 psig. Pressure would stablize at some value below the starting setpoint for both pumps based on demand.

b. Plausible since the electric pump would start, but pressure is below diesel pump start setpoint so it would also be operating.

c. Plausible since both pumps would be running, but start order of pumps is backwards.

d. Plausible if misconception is that diesel pump starts first and electric pump setpoint is below 83 psig, but both pumps would be running.

DIFFICULTY:
Comprehensive/Analysis ☐ *Knowledge/Recall* ☒ *Rating* 3

Knowledge of the automatic response of the fire system to decreasing pressure

REFERENCES SUPPLIED:

Fuel oil sufficient for at least eight (8) hours of operation is supplied by a 450 gallon fuel tank located outside of the intake structure. Normal usage is approximately 10 gal. per hour when the pump is running.

Upon a reduction of pressure in the fire main loop, pressure switches initiate a sequential starting of the fire pumps. The Motor Driven Fire Pump starts at 100 psig (95 psig - 105 psig). Should the fire main pressure drop to 90 psig (85 psig - 95 psig), the Engine Driven Fire Pump will automatically start. Each pump discharges through a swing check valve and gate valve to the fire water header. The swing check valves prevent reverse flow through the non-running fire pumps.

Pressure relief valves set at 135 psig provide protection for each fire pump by discharging excess water back into Lake Robinson. Air release valves adjacent to the relief valves are connected to high points and vent air from the discharge piping in an effort to reduce water hammer. Hose manifolds are provided as a means of testing fire pump capacity and can also be used as fire hydrants.

The three (3) fire pumps can be manually operated at their respective local control panels. Remote operation and indications for the Motor Driven Fire Pump (MDFP) are provided in the Control Room on the Containment Fire Protection Panel (CFPP). The Fire Alarm Console (FAC) in the Control Room provides alarms only. Operations, indications and alarms for the Engine Driven Fire Pump (EDFP) and its control system occur locally on its controller. The Fire Alarm Console (FAC) in the Control Room provides alarms only. The booster pump is only operated at its local control panel. There are no local or remote indication or alarms other than pressure gauge indications.

Post indicator gate valves (P.I.V.'s) are strategically located within the fire main loop. The normally open valves permit isolation of a section of the fire main loop without loss of fire service to other than the isolated section. A section of the fire main loop may be defined by its boundary valves.

Attachment 3 lists the systems, hydrants and hose stations which would be rendered inoperable by the isolation of various sections of the fire main loop. Attachment 3 also lists back-up sources of fire water for the affected systems, hydrants and hose stations.

FP-05 003

Which ONE (1) of the following describes the AUTOMATIC operation of the Fire Water Protection System?

✓A. The diesel fire pump will start if system pressure drops to 85 psig.

B. The electric fire pump will start if system pressure drops to 115 psig.

C. The electric fire pump will stop if system pressure is restored to 125 psig.

D. The diesel fire pump will start on a loss of power to the jockey pump.

Question: 66

Given the following conditions:

- Emergency Diesel Generator 'A' is in the process of being started on Unit 2 to parallel it to the E-1 Bus.
- A "Remote Manual Slow Speed Start" is being performed in accordance with OP-604, "Diesel Generators A and B."

Which ONE (1) of the following describes the operation of the diesel generator voltage control during this evolution?

a. The Voltage Regulator will automatically control voltage between 470 VAC and 490 VAC during the entire start after the field is automatically flashed at 200 RPM.

b. The Voltage Regulator must be manually shutdown after the field is automatically flashed at 200 RPM, and will be automatically reinstated when engine speed is above 900 RPM to control voltage between 470 VAC and 490 VAC.

c. The Voltage Regulator will be automatically shutdown 5 seconds after the field is flashed at 200 RPM if engine speed does **NOT** reach 900 RPM, and must be manually reinstated when engine speed is above 900 RPM to control voltage between 470 VAC and 490 VAC.

d. The Voltage Regulator must be manually shutdown after the field is automatically flashed at 200 RPM, and must be manually reinstated when engine speed is above 900 RPM to control voltage between 470 VAC and 490 VAC.

Answer:

d. The Voltage Regulator must be manually shutdown after the field is automatically flashed at 200 RPM, and must be manually reinstated when engine speed is above 900 RPM to control voltage between 470 VAC and 490 VAC.

QUESTION NUMBER: 66
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 064A4.02

Ability to manually operate and/or monitor in the control room: Adjustment of exciter voltage (using voltage control switch)

K/A IMPORTANCE: ***RO*** 3.3 ***SRO*** 3.4
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: EDG-08

EXPLAIN the component operation associated with each switch position for the Emergency Diesel Generator System switches and controls.

REFERENCES: OP-604

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number EDG-08 001

JUSTIFICATION:

a. Plausible since the regulator is designed to control voltage in this range and the field automatically flashes above 200 rpm, but it must be manually shutdown to prevent damage after 5 seconds.

b. Plausible since the field automatically flashes above 200 rpm and must be manually shutdown, but it must be manually reinstated above 900 rpm.

c. Plausible since the voltage regulator is to be shutdown within 5 seconds after reaching 200 rpm, but this is a manual operation not automatic.

d. **CORRECT** The field automatically flashes when speed increases above 200 rpm. The voltage regulator must be manually shutdown within 5 seconds if speed will be maintained below 900 rpm and then manually reinstated when speed is increased above 900 rpm.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of procedural requirements for starting an EDG

REFERENCES SUPPLIED:

4.6 When the Fuel Oil Filter pressure differential exceeds 10 psid, cartridge replacement is required.

4.7 Diesel Generator loads shall **NOT** exceed ratings of 2,500 KW for continuous operation **AND**:

- 2750 KW shall **NOT** be exceeded.
- Operation at 2750 KW for more than 2 hours within a 24 hour period shall **NOT** occur.
- 4,000 amps on the Generator shall **NOT** be exceeded.

4.8 The Diesel Generators should not be set for automatic start after draining the EDG Fuel Oil System. The fuel oil system should be hand primed after the system is restored prior to EDG startup. To ensure the EDG Fuel Oil System is not airbound after refilling, the EDGs should be run prior to being set for automatic. A manual start is preferred so as not to challenge the overcranking feature associated with an automatic start.

4.9 When the Diesel is running and the Trips Defeat Key Switch is in the TRIPS DEFEATED position; the Diesel should be manually tripped if a condition exists that would automatically trip the Diesel. These conditions are: (Rail 92R0044)

- Coolant Temperature High 205°F
- Crankcase Pressure + 0.5 inches H_2O
- Lube Oil Low Pressure 18 psig
- Coolant Low Pressure 12 psig

4.10 Synchroscopes will be left OFF unless in use for synchronizing to prevent damaging by inadvertent energizing of two synchroscopes.

4.11 The Diesel Generator shall not be operated at speeds below 900 rpm with the Field Excitation in service. To take Field Excitation out-of-service, the Diesel Generator shall be above a speed of 200 rpm and below a speed of 900 rpm and the Field flashed. With the Field flashed, depressing the VOLTAGE SHUTDOWN pushbutton on Generator Control Panel, will remove Field Excitation from service. To reinstate Field Excitation, bring Diesel Generator speed to between 890 and 910 rpm and depress the RESET pushbutton on the Generator Control Panel. If the Field Excitation was taken out-of-service and Diesel Generator speed was dropped below 200 rpm or Diesel Generator was stopped, Field Excitation will reset automatically and is required to be taken out-of-service if Diesel Generator speeds stays below 900 rpm.

6.3.2 (Continued) INIT

CAUTION

If EDG Field Excitation is **NOT** removed within 5 seconds after the EDG is started, Regulator damage can result. It should be removed when "Generator AC Volts" and "Generator Hertz" meters first show indication. This occurs approximately 2-3 seconds after the Air Start Solenoids open.

8. Station an Operator at EDG "A" Generator Control Panel to depress the VOLTAGE SHUTDOWN pushbutton. ______

9. Start EDG "A" from the RTGB.
TIME STARTED ________ ______

10. Depress the VOLTAGE SHUTDOWN pushbutton at EDG "A" Generator Control Panel within 5 seconds of EDG "A" start. ______

11. Verify OPEN TCV-1660, DIESEL "A" TEMP CONTROL VALVE. ______

CAUTION

Lube Oil Pressure shall not exceed 55 psig with the Lube Oil Temperature at or below 130°F.

12. Raise EDG "A" speed to maintain Lube Oil Pressure greater than 18 psig. ______

NOTE: Approximately 2 minutes will be required to raise from 400 rpm to 900 rpm.

13. With the Speed Control Lever on EDG "A" Generator Control Panel, raise engine speed to 900 rpm as indicated on the local RPM indicator beside EDG "A" Generator Control Panel **OR** as indicated by portable RPM Indicator **AND** record which RPM indicator was used.
Local / Portable ______
(Circle one)

6.3.2 (Continued) INIT

> **NOTE:** The RESET pushbutton located just left of the VOLTAGE SHUTDOWN pushbutton activates the "Generator Hertz" meter.

14. Perform the following:

    a. Depress the RESET pushbutton located just to the left of the VOLTAGE SHUTDOWN pushbutton. ______

    b. Record Generator voltage. Voltage ______ ______

    c. **IF** EDG "A" voltage is less than 470V **OR** greater than 490V, **THEN** notify the Control Room to perform the following:

        1) Record voltage from ERFIS point DGV3026A, "A" D/G Voltage. Voltage ______ ______

        2) **IF** voltage is less than 467V **OR** greater than 493V, **THEN** request Engineering personnel to review EE107-CS-65 and EE107-CS-68 **AND** perform an Operability Determination. ______

> **NOTE:** Coolant discharge pressure fluctuations on the jacket water system shall be observed. Fluctuations of greater than 3 psig indicate a possible water leak between the jacket water system and the cylinder liner. A water leak of this type could lead to erosion of both the cylinder liner and pistons.

15. Operate EDG "A" for 3 to 5 minutes after starting, to warm EDG "A". ______

EDG-08 001

Given the following plant conditions:

- Emergency Diesel Generator "A" is in the process of being started on Unit 2 to parallel it to the E-1 Bus.

Which ONE (1) of the following describes the operation of the diesel generator voltage control switch during this evolution?

A. Lowering the voltage control knob has no effect on the generator voltage if selected to the AUTO mode of operation

✓B. Raising the voltage control knob to a higher value, will cause the generator to pick up a larger share of the reactive load after breaker closure

C. Raising the voltage control knob will correct a synchroscope which is traveling slowly in the SLOW direction

D. Lowering the voltage control knob raises the speed of the generator when operating in the MANUAL mode prior to paralleling with offsite source

Question: 67

Given the following conditions:

- The unit is in Hot Standby.
- All systems are operating normally.
- SG "A" PORV is closed.
- SG "A" PORV automatic potentiometer is adjusted from "3.10" to "1.50".

Which ONE (1) of the following describes the effect adjusting the potentiometer will have on the PORV?

	SETPOINT	PORV
a.	Increases	Opens
b.	Decreases	Open
c.	Increases	Remains Closed
d.	Decreases	Remains Closed

Answer:

c.	Increases	Remains Closed

QUESTION NUMBER: 67
TIER/GROUP: ***RO*** 2/3 ***SRO*** 2/3
K/A: 041K6.03

Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 2.9
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: SD-09

EXPLAIN the normal operation of the Steam Dump control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-031

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number MSS-12 002

JUSTIFICATION:

a. Plausible since the setpoint is raised, but the PORV would remain closed.

b. Plausible since the setpoint would be decreased on most potentiometer adjusted controllers, but PORV stations are reversed so setpoint actually increases.

c. **CORRECT** Setting of 3.10 is 1035 psig. The range for the ten turn pot is 0-1500 psig. Changing the setpoint to 1.50 would raise the setpoint to 1351.5 psig. Since this is higher than even the safety setpoins, the PORV will remain closed.

d. Plausible since the PORV will remain closed and the setpoint would be decreased on most potentiometer adjusted controllers, but PORV stations are reversed so setpoint actually increases.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the magnitude of the effect of operator actions on the SG PORV

REFERENCES SUPPLIED:

4.1.4 Low T_{avg} Interlock

The steam dump valves will lose their arming signal if 2/3 Low T_{avg} (543°F) signals are received. This signal comes from T_{avg} protection channels and will lock out the steam dumps to prevent inadvertent cooldown due to steam dump valves.

With T_{avg} less than 543°F, three of the dumps (Bank 1) can be made operable by operator action to bypass the T_{avg} interlock. This limits the cooldown rate available from the steam dump system.

4.2 Power Operated Relief Valve Controls

The controls for the S/G PORVs are located in the Secondary Control Panel on the mezzanine level of the turbine building, with the exception of the automatic setpoint adjustment potentiometer which is located on the RTGB. Their normal setpoint at power is 1035 psig, which is 30 psi above the pressure corresponding to the no load Tavg of 547°F. The setpoint can be can be changed by adjusting the potentiometer on the RTGB. This 10 turn potentiometer controls over a 0 - 1500 psig range, with a setting of 10.0 corresponding to 0 psig. This controller is reverse acting. Instead of the potentiometer increasing setpoint with increased value, raising the setting decreases the setpoint at which the pressure will be controlled. When actual pressure increases to the setpoint, the PORV throttles open to relieve pressure.

The controllers for each S/G PORV (PIC-477, PC-487 and PC-497), are adjusted at the secondary control panel. These controllers are pneumatic (with no electronics) and sense S/G pressure directly off the main steam lines upstream of the MSIVs. The directions for adjusting these controllers, which requires coordination between the Outside Auxiliary Operator and the Control Room, are contained in GP-001.

The PORVs can only be controlled by the steam dump controller if the system is selected to Tave mode ,and then, only if a turbine trip has not occured.

4.2.1 Switches

There are three DEFEAT switches located at the Secondary Control Panel to allow manual control of the S/G PORVs from the Secondary Control Panel. After placing each switch in the DEFEAT position, the S/G PORV is controlled by selecting MANUAL on the transfer switch located inside the controller box and using the manual thumbwheel on the pressure controller. When in the DEFEAT position, automatic control from the RTGB is removed as is the ability to place the S/G PORV under steam dump control in the event of a 70% load rejection without a turbine trip. Remote indication of this action

MSS-12 002

Given the following plant conditions:

- Mode 1 at 100% RTP
- All systems are operating normally
- S/G "A" PORV automatic potentiometer is adjusted clockwise 1.7 turns

Which ONE (1) of the following describes the effect adjusting the pot will have on the PORV and Plant conditions?

A. Increases the setpoint; the PORV will open, increasing steam demand above 100%.

✓B. Decreases the setpoint; the PORV will open, increasing steam demand above 100%.

C. Increases the setpoint; the PORV will not open due to current system pressures.

D. Decreases the setpoint; the PORV will not open due to current system pressures.

Question: 68

Given the following conditions:

- A small break LOCA has occurred.
- Entry has been made into FRP-C.1, "Response to Inadequate Core Cooling."
- CETs are all indicating between 740 °F and 760 °F and rising slowly.
- RCS pressure has stabilized at 1605 psig.
- PZR level is off-scale low.
- RVLIS Full Range is indicating 39% and lowering slowly.
- Charging flow is **NOT** available.
- SG pressures are all between 360 psig and 400 psig.

Which ONE (1) of the following actions should be taken?

a. Dump steam to cooldown and depressurize the RCS to provide Safety Injection flow

b. Open the RCS Vent System valves to depressurize the RCS to provide Safety Injection flow

c. Start an RCP immediately to provide forced cooling flow

d. Open the PZR PORVs to depressurize the RCS to provide Safety Injection flow

Answer:

a. Dump steam to cooldown and depressurize the RCS to provide Safety Injection flow

QUESTION NUMBER: 68

TIER/GROUP: *RO* 1/1 *SRO* 1/1

K/A: WE06EK2.2

Knowledge of the interrelations between the (Degraded Core Cooling) and the facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems.

K/A IMPORTANCE: *RO* 3.8 *SRO* 4.1

10CFR55 CONTENT: ***55.41(b) RO*** 5 ***55.43(b) SRO***

OBJECTIVE: FRP-C.1-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of steps related to inadequate core cooling as directed in FRP-C.1.

REFERENCES: FRP-C.1

SOURCE: ***New*** ☒ ***Significantly Modified*** ☐ ***Direct*** ☐

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** SGs should be depressurized in 2 steps (140 psig and atmospheric pressure) in an attempt to cooldown and depressurize the RCS to provide injection flow.

b. Plausible since this is an alternate bleed flowpath if entry had been made to FRP-H.1, but valves are only verified closed in FRP-C.1 to ensure that these are not the cause of the LOCA.

c. Plausible since RCPs will be started if CETs exceed 1200 °F and attempts to cooldown and depressurize using other means are not successful, but start requirements are not yet met.

d. Plausible since this is the normal bleed flowpath if entry had been made to FRP-H.1, but valves are only verified closed in FRP-C.1 to ensure that these are not the cause of the LOCA.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of plant conditions to determine appropriate actions in response to inadequate core cooling

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Determine SI Accumulator Isolation Valve Status As Follows:	
	a. Check SI ACCUM DISCHs - POWER AVAILABLE	a. Locally close the breakers for the following valves: • SI-865C, ACCUMULATOR C DISCHARGE (MCC-5, CMPT 9F) • SI-865A, ACCUMULATOR A DISCHARGE (MCC-5, CMPT 14F) • SI-865B, ACCUMULATOR B DISCHARGE (MCC-6, CMPT 10J)
	b. Check ACCUM DISCHs - OPEN • SI-865A • SI-865B • SI-865C	b. Open the ACCUM DISCH Valves unless closed after Accumulators discharged.
8.	Check Core Exit T/Cs - LESS THAN 1200° F	Go To Step 17.
9.	Check RCP Status - ANY RUNNING	Go To Step 11.
10.	Reset SPDS <u>AND</u> Return To Procedure And Step In Effect	
11.	Check RVLIS Full Range Indication - GREATER THAN 41%	Go To Step 13.
12.	Reset SPDS <u>AND</u> Return To Procedure And Step In Effect	
13.	Check RVLIS Trend - STABLE <u>OR</u> DECREASING	Observe <u>CAUTION</u> prior to Step 1 and Go To Step 1.
14.	Check Core Exit T/Cs - LESS THAN 700° F	Go To Step 16.
15.	Reset SPDS <u>AND</u> Return To Procedure And Step In Effect	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
16.	Check Core Exit T/C Trend - STABLE <u>OR</u> INCREASING	Observe <u>CAUTION</u> prior to Step 1 and Go To Step 1.
*17.	Determine Containment Hydrogen Concentration From Either Of The Following: • PI-8101-1, CHANNEL I H_2 ANALYZER <u>OR</u> • PI-8111-2, CHANNEL II H_2 ANALYZER	Perform the following: a. Notify Chemistry personnel to perform the following: • Obtain a sample of Containment atmosphere using the Post Accident Sample System • Analyze sample to determine Containment hydrogen concentration b. <u>WHEN</u> Containment hydrogen concentration sample results are available, <u>THEN</u> perform Step 18. Go To Step 19.
18.	Evaluate Containment Hydrogen Concentration As Follows:	
	a. Check hydrogen concentration - LESS THAN 6.0%	a. Consult Plant Operations Staff for additional recovery actions. Go To Step 19.
	b. Check hydrogen concentration - LESS THAN 0.5% <u>AND</u> STABLE	b. Notify Plant Operations Staff to make arrangements for delivery of the Hydrogen Recombiner.
*19.	Check CST Level - LESS THAN 10%	<u>IF</u> CST level decreases to less than 10%, <u>THEN</u> perform Step 20. Observe <u>NOTE</u> prior to Step 21 and Go To Step 21.
20.	Align SW To The AFW Pump Suction Using OP-402, Auxiliary Feedwater System	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

The preferred order of S/G use in the subsequent step is intact, faulted, then ruptured. The step refers to "intact", however, faulted OR ruptured may be used if that is all that is available.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
21.	Control Intact S/G Levels As Follows:	
	a. Check any S/G - INTACT	a. IF a faulted S/G is available, THEN use a faulted S/G. IF no intact OR faulted S/G is available, THEN use a ruptured S/G.
	b. Check intact S/G levels - LESS THAN 8% [18%]	b. Go To Step 21.e.
	c. Perform either of the following: • Establish FW bypass flow greater than $0.2x10^6$ pph until level in at least one intact S/G is greater than 8% [18%] OR • Establish AFW flow greater than 300 gpm until level in at least one intact S/G is greater than 8% [18%]	
	d. Check feed flow - GREATER THAN 300 GPM OR $0.2x10^6$ PPH	d. Continue attempts to establish feed flow. Go To Step 30.
	e. Control feed flow to maintain intact S/G levels - BETWEEN 8% [18%] AND 50%	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*22.	Check RCS Vent Paths:	
	a. Check power to PZR PORV BLOCK Valves - AVAILABLE	a. Close the breakers for the following PRESSURIZER PORV BLOCK Valves: • RC-535 (MCC 6, CMPT 7J) • RC-536 (MCC 6, CMPT 8J)
	b. Check RCS pressure - LESS THAN 2335 PSIG	b. WHEN RCS pressure decreases to less than 2335 psig, THEN verify CLOSED PZR PORVs OR associated PORV BLOCK Valves. Go To Step 22.e.
	c. Verify PZR PORVs - CLOSED	c. Verify CLOSED the associated PORV BLOCK Valve(s).
	d. Check PORV BLOCKs - AT LEAST ONE OPEN	d. Open one PORV BLOCK Valve unless it was closed to isolate an open PZR PORV.
	e. Verify RCS Vent System Valves - CLOSED OR DEENERGIZED: • RC-567, HEAD VENT • RC-568, HEAD VENT • RC-569, PZR VENT • RC-570, PZR VENT • RC-571, PRT ISO • RC-572, CV ATMOS	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- Partial uncovery of S/G tubes is acceptable in the following steps due to steaming faster than feeding.
- After the Low Steamline Pressure SI Signal is blocked, main steamline isolation will occur if the high steam flow rate setpoint is exceeded.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*23.	Depressurize All Intact S/Gs To 140 PSIG As Follows:	
	a. Check Steam Dump to Condenser - AVAILABLE	a. Dump steam at maximum rate using STEAM LINE PORVs. Go To Step 23.c.
	b. Dump steam to Condenser at maximum rate	
	c. Check RCS Hot Leg Temperatures - LESS THAN 543°F	c. <u>WHEN</u> RCS hot leg temperatures less than 543°F, <u>THEN</u> perform Step 23.d. Go To Step 23.e.
	d. Defeat Low Tavg Safety Injection Signal as follows: 1) Momentarily place SAFETY INJECTION T-AVG Selector Switch to BLOCK position 2) Verify LO TEMP SAFETY INJECTION BLOCKED status light - ILLUMINATED	
	e. Check S/G pressures - LESS THAN 140 PSIG	e. <u>IF</u> S/G pressure is decreasing, <u>THEN</u> observe <u>NOTE</u> prior to Step 21 and Go To Step 21. <u>IF</u> S/G pressure is increasing, <u>THEN</u> Go To Step 30.

(CONTINUED NEXT PAGE)

Question: 69

Given the following conditions:

- The unit is at operating at 35% power in preparation for increasing power to 100%.
- Circulating Water Pump 'A' is under clearance for maintenance.
- A fault occurs on 4KV Bus #4 and all loads are lost.

Which ONE (1) of the following describes the effect on the turbine to the above conditions?

a. The turbine will **NOT** automatically trip, but must be manually tripped when condenser vacuum lowers to 24.5" Hg

b. The turbine will automatically trip due to all 3 Circulating Water Pump breakers being open

c. The turbine will automatically trip when condenser vacuum decreases to 17" Hg unless load is lowered to within the capacity of the one remaining Circulating Water Pump

d. The turbine will **NOT** automatically trip due to load already being within the capacity of the one remaining Circulating Water Pump

Answer:

b. The turbine will automatically trip due to all 3 Circulating Water Pump breakers being open

QUESTION NUMBER: 69
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 075A2.02

Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) use procedures to correct, control, or mitigate the consequences: Loss of circulating water pumps

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 2.7
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: CW-09

EXPLAIN the normal operation of the CW control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: APP-008
OP-603

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number EHC-11 004

JUSTIFICATION:

a. Plausible since a lowering vacuum without any chance of recovery will require a manual trip, but an automatic trip will occur due to the loss of all 3 CW pumps.

b. **CORRECT** The loss of power will result in all 3 CW pump breakers being open. This will generate an automatic turbine trip.

c. Plausible since an automatic trip on low vacuum would occur if one CW pump were not able to remove enough heat to maintain vacuum, but no CW pumps are available.

d. Plausible since a single CW pump might be able to remove adequate heat at this power level, but no CW pumps are available.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of the knowledge of CW pump power supplies and the effect on the plant of the loss of power

REFERENCES SUPPLIED:

ALARM

CW PMP A MOTOR/DISCH VLV TRIP/OVLD

AUTOMATIC ACTIONS

1. Turbine trip on last CW pump trip.

CAUSE

1. Open Supply Breaker **OR** Motor Overload Trip on Discharge Valve Motor
2. Electrical Fault **OR** Overload Trip of CWP Breaker
3. ∅B Overload of CWP but no TRIP

OBSERVATIONS

1. V6-50A, CIRC WATER PUMP "A" DISCH, position/status
2. CWP Motor Breaker Status
3. Condenser Vacuum
4. CWP Discharge Pressure (PI-1600A)

ACTIONS

1. **IF** Turbine is operating, **THEN** refer to AOP-012.
2. **IF** Turbine is **NOT** operating, **THEN** perform the following:
    1) **IF** V6-50A, CIRC WATER PUMP "A" DISCH, breaker is tripped, **THEN** attempt one reset of the breaker located at MCC 7, Compartment 1M.
    2) **IF** Circ Water Pump "A" breaker is tripped, **THEN** perform the following:
        a. Verify CLOSED V6-50A, CIRC WATER PUMP "A" DISCH.
        b. Start an available Circ Water Pump.
        c. **IF** the minimum number of Circ Water Pumps required for liquid waste releases are **NOT** operating, **THEN** verify any Liquid Waste Batch Releases are terminated.

DEVICE/SETPOINTS

1. CWP Breaker 74 Relay / energized
2. CWP Breaker ∅B51 Device / energized
3. Discharge Valve 74 Relay / deenergized

POSSIBLE PLANT EFFECTS

1. Decrease **OR** Loss of Vacuum
2. Plant Shutdown

REFERENCES

1. AOP-012, Partial Loss of Condenser Vacuum or Circulating Water Pump Trip
2. CWD B-190628, Sheet 811, cable G
3. Flow Diagram G-190199, Sheet 1

ELECTRICAL DISTRIBUTION SYSTEM STARTUP LINEUP

4160V BUS 4

DESCRIPTION	BREAKER POSITION	INITIALS
4KV BUS 3-4 TIE BKR 52/19	CLOSED*	
UNIT AUX TO 4KV BUS 4 BKR 52/20	OPEN*	
CONDENSATE PUMP "B" BKR 52/22	RACKED IN	
CIRCULATING WATER PUMP "B" BKR 52/23	RACKED IN	
FEED TO 4KV BUS 5 BKR 52/24	CLOSED	
Local CIRCUIT BREAKER CONTROL Switch for BKR 52/24, FEED TO 4KV BUS 5	PUSHED IN AND VERTICAL	
HEATER DRAIN PUMP "B" BKR 52/25	RACKED IN	
FEEDWATER PUMP "B" BKR 52/26	RACKED IN	
REACTOR COOLANT PUMP "B" BKR 52/27	RACKED IN	
STATION SERVICE TRANSFORMER 2D BKR 52/28	CLOSED	
Local CIRCUIT BREAKER CONTROL Switch for BKR 52/28, STATION SERVICE TRANSFORMER 2D	PUSHED IN AND VERTICAL	

4160V BUS 5

SPARE BKR. 52/31	RACKED OUT	
STATION SERVICE TRANSFORMER 2E BKR 52/32	RACKED IN	
Local BREAKER CONTROL Switch for BKR 52/32, STATION SERVICE TRANSFORMER 2E	PUSHED IN AND VERTICAL	
CIRCULATING WATER PUMP "C" BKR 52/33	RACKED IN	
SPARE BKR 52/34	RACKED OUT	
4160V BUS 5 BREAKERS RELAY TARGETS	RESET	

*Breaker position varies with Plant conditions IAW general procedures.

Exceptions

______________________ ______________________

______________________ ______________________

______________________ ______________________

Question: 70

Given the following conditions:

- The unit is operating at 2% power.
- The following RCP indications are observed:

INDICATION	RCP 'A'	RCP 'B'	RCP 'C'
Motor Bearing Temperatures	210°F and ↑ slowly	180°F and stable	195°F and ↑ slowly
#1 Seal Leakoff Temperatures	150°F and stable	150°F and stable	165°F and ↑ slowly
#1 Seal Leakoff Flow	5.8 gpm and stable	4.2 gpm and stable	3.8 gpm and stable
Thermal Barrier ΔP	10" and stable	10" and stable	8" and stable
Frame Vibration	3.6 mils and ↑ at 0.1 mil per hr	2.8 mils and stable	4 mils and ↑ at 0.05 mil per hr
Shaft Vibration	12 mils and stable	7 mils and stable	9.5 mils and ↑ at 0.6 mils per hour

Which ONE (1) of the following describes the actions required for this condition?

a. Stop 'A' RCP and enter Technical Specification 3.4.4, RCS Loops - Modes 1 & 2

b. Trip the reactor, stop 'A' RCP, and go to PATH-1

c. Stop 'C' RCP and enter Technical Specification 3.4.4, RCS Loops - Modes 1 & 2

d. Trip the reactor, stop 'C' RCP, and go to PATH-1

Answer:

a. Stop 'A' RCP and enter Technical Specification 3.4.4, RCS Loops - Modes 1 & 2

QUESTION NUMBER: 70
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: 015/017AA1.20

Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP bearing temperature indicators

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 2.7
10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: AOP-018-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in AOP-018 by explaining the basis of each.

REFERENCES: AOP-018
AOP-014

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number AOP-014-03 011

JUSTIFICATION:

a. **CORRECT** A' RCP motor bearing temperature has exceeded limits and the pump must be stopped. With the plant in Mode 2, a reactor trip is not required.

b. Plausible since these would be the correct actions if the plant was in Mode 1, but the plant is in Mode 2.

c. Plausible since these are the correct actions, but 'C' RCP has not reached any trip limits while 'A' RCP has.

d. Plausible since these would be the correct actions if the plant was in Mode 1, but 'C' RCP has not reached any trip limits while 'A' RCP has and the plant is in Mode 2.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 4

Analysis to determine which RCP must be stopped and comparison to power level to determine proper action

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 2 of 5)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Check APP-001-B1, RCP BRG COOL WTR LO FLOW - EXTINGUISHED	Verify the following CCW Valves open: • CC-716A, CCW TO RCP ISO • CC-716B, CCW TO RCP ISO • CC-730, BRG OUTLET ISO IF CCW to the RCP(s) can NOT be restored, THEN perform the following: a. Trip the reactor b. Trip the affected RCPs c. Go To Path-1 while continuing with this procedure. d. Go To Step 4.
* 3.	Check ALL RCP Motor Bearing Temperature - LESS THAN 200°F	IF the Plant is in Mode 2 OR less, THEN stop the affected RCP. IF the Plant is in Mode 1, THEN perform the following: a. Trip the reactor. b. Stop the affected RCP(s). c. Go To Path-1 while continuing with this procedure.
4.	Check CCW HX OUTLET Temperature - GREATER THAN 105°F	Go To Step 12.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

HIGH REACTOR COOLANT PUMP VIBRATION

(Page 1 of 4)

> **NOTE**
>
> Vibration rate changes (increase or decrease) for diagnosing a problem are valid only during steady state conditions.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Check The Following Vibration Levels To Determine If RCP Trip(s) Are Required: • Frame - GREATER THAN 5 MILS OR • Frame - GREATER THAN 3 MILS AND INCREASING AT GREATER THAN 0.2 MILS/HOUR OR • Shaft - GREATER THAN 20 MILS OR • Shaft - GREATER THAN 15 MILS AND INCREASING AT GREATER THAN 1 MIL PER HOUR	IF any of the vibration limits are exceeded, THEN Go To Step 2. Go To Step 8.
2.	Check Plant Status - MODE 1	Stop the affected RCP(s). Go To Step 4.
3.	Perform The Following: a. Trip the reactor b. Trip the affected RCP(s) c. Go To Path-1 while continuing with this procedure.	
4.	Check RCP B OR C - RUNNING	Go To Step 10.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 1 of 11)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Check Any RCP #1 Seal Leakoff Flow - GREATER THAN 5.7 GPM	IF seal leakoff exceeds 5.7 gpm, THEN Go To Step 2. Go To Step 8.
2.	Check Either Of The Following Conditions Exist: • RCP #1 Seal Leakoff Flow On Unaffected RCP(s) - DECREASED OR • RCP Thermal Barrier ΔP On Affected RCP(s) - DECREASED	Perform the following: a. Perform cross-check of all RCP parameters to determine cause of indicated high leakoff flow. b. Observe The NOTE Prior To Step 1 and Go To The Main Body, Step 1 Of This Procedure

****

CAUTION

To prevent damage to the RCP Seal Stack, the affected RCP Seal Leakoff Isolation valve must be closed between 3 minutes and 5 minutes of stopping the RCP.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
3.	Check Plant Status - MODE 1	Stop the affected RCP(s) Observe the CAUTION prior to Step 5 and Go To Step 5.
4.	Perform The Following: a. Trip the reactor b. Trip the affected RCP(s) c. Go To Path-1 while continuing with this procedure.	

AOP-014-03 011

Given the following plant conditions:

- Mode 1 at 35% RTP
- Two charging pumps are running
- The following RCP indications are observed:

	RCP "A"	RCP "B"	RCP "C"
o RCP motor bearing temperatures	180°F	180°F	210°F
o #1 seal leakoff temperatures	150°F	150°F	165°F
o Thermal barrier delta P	10"	10"	8"

Which ONE (1) of the following describes the action(s) required for this condition?

A. Stop "C" RCP, shutdown IAW GP-006, Normal Plant Shutdown From Power Operation To Hot Shutdown, and be in Mode 3 within 6 hours.

B. Throttle CVC-297C, "C" RCP Seal Water Flow Control valve, to obtain between 8 and 13 gpm flow to each "C" RCP Seals.

C. Close CVC-303C, "C" RCP Seal Leakoff valve.

✓D. Trip the reactor, stop RCP "C".

Question: 71

Which ONE (1) of the following requires entry into DSP-001, "Alternate Shutdown Diagnostic"?

a. A fire in the Main Turbine that has the potential to destroy the generator when the reactor is above 10% power

b. A fire in the Containment Vessel that has the potential to destroy the pressurizer heater power cables when in hot standby

c. A fire in the Control Room that has the potential to destroy RHR pump control cables when refueling

d. A fire in the Auxiliary Building that has the potential to destroy the running Charging Pump when in cold shutdown

Answer:

b. A fire in the Containment Vessel that has the potential to destroy the pressurizer heater power cables when in hot standby

QUESTION NUMBER: 71
TIER/GROUP: ***RO*** 1/1 ***SRO*** 1/1
K/A: 067AA2.04

Ability to determine and interpret the following as they apply to the Plant Fire on Site: The fire's extent of potential operational damage to plant equipment

K/A IMPORTANCE: ***RO*** 3.1 ***SRO*** 4.3
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: DSP-001-02

RECOGNIZE the selected entry level conditions of DSP-001.

REFERENCES: DSP-001

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number DSP-001-02 005

JUSTIFICATION:

a. Plausible since operating in Mode 1 and would damage equipment vital to generating capacity, but not located in AB, CV, or CR.

b. **CORRECT** Entry conditions are a fire in the AB, CV, or CR that has the potential to damage vital controls/components and/or their power/control cables when in Mode 4 or higher.

c. Plausible since entry would be made into DSP-001 if in a higher Mode, but temperature is below required entry conditions.

d. Plausible since entry would be made into DSP-001 if in a higher Mode, but temperature is below required entry conditions.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of the entry conditions / purpose of AOPs

REFERENCES SUPPLIED:

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure determines whether conditions exist that warrant the use of the DSPs and to provide guidance as to which specific DSP should be implemented.
For entry into the Dedicated Shutdown Procedures, the following assumptions were made:

a. All plant equipment will function at designed capability and may be lost only as a result of fire damage.

b. No accidents or equipment failures other than those caused by the fire are assumed to occur coincident with a complete 72 hour loss of offsite power.

2. ENTRY CONDITIONS

A fire in the AUX BLDG, CV, or Control Room that has the potential to damage vital plant components/controls and/or their power/control cables when Tavg is greater than 200°F.

- END -

Question: 72

CC-707, Component Cooling Water Surge Tank relief valve, is sized to accommodate the ...

a. maximum CCW insurge to the tank resulting from a loss of the Residual Heat Removal system.

b. maximum flowrate associated with a rupture of a Reactor Coolant Pump Thermal Barrier Heat Exchanger.

c. maximum CCW insurge to the tank resulting from a loss of the Service Water system.

d. maximum flowrate associated with a rupture of a Residual Heat Removal pump cooler during the recirculation phase of an accident.

Answer:

b. maximum flowrate associated with a rupture of a Reactor Coolant Pump Thermal Barrier Heat Exchanger.

QUESTION NUMBER: 72
TIER/GROUP: ***RO*** 2/3 ***SRO*** 2/3
K/A: 008K4.02

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the surge tank, including the associated valves and controls

K/A IMPORTANCE: ***RO*** 2.9 ***SRO*** 3.7
10CFR55 CONTENT: ***55.41(b) RO*** 4 ***55.43(b) SRO***

OBJECTIVE: CCW-05

DESCRIBE the performance and design attributes of the major CCW System components.

REFERENCES: SD-013

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number CCW-04 001

JUSTIFICATION:

a. Plausible since CCW and RHR systems interface, but loss of RHR would cause CCW to cooldown not heatup.

b. **CORRECT** Sized to relieve the maximum flowrate of water following the rupture of a RCP thermal barrier cooling coil.

c. Plausible since CCW is cooled by SW, but loss of SW would not cause a heatup of sufficient magnitude to cause an insurge to challenge the capacity of this valve.

d. Plausible since CCW cools RHR pump cooler, but differential pressure would not cause insurge.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of CCW system design attributes

REFERENCES SUPPLIED:

ATTACHMENT 10.4
Page 1 of 2

CCW RELIEF VALVE AND SET POINTS

RELIEF	DESCRIPTION	SETPOINT
CC-707	Component Cooling Surge Tank Relief Valve.	100 psig ± 3 psig
	Discharges to the LWD-Waste Holdup Tank - Sized to relieve the maximum flow rate of water following a rupture of a RCP thermal barrier cooling coil.	
CC-715	Excess Letdown Heat Exchanger CCW Outlet Relief Valve.	125 psig ± 3.75 psig
	Discharges to the Containment Sump.	
CC-722A	RCP A Thermal Barrier Cooler CCW Outlet Relief Valve.	2485 psig ± 74.6 psig
	Discharges to the Containment Sump.	
CC-722B	RCP B Thermal Barrier Cooler CCW Outlet Relief Valve.	2485 psig ± 74.6 psig
	Discharges to the Containment Sump.	
CC-722C	RCP C Thermal Barrier Cooler CCW Outlet Relief Valve.	2485 psig ± 74.6 psig
	Discharges to the Containment Sump.	
CC-729	RCPs A, B, & C Motor Bearing Oil Cooler CCW Outlet Relief Valve.	125 psig ± 3.75 psig
	Discharges to Containment Sump.	
CC-747A	RHR Heat Exchanger A CCW Outlet Relief Valve.	150 psig ± 4.5 psig
	Discharge to the CCW Return Header.	
CC-747B	RHR Heat Exchanger B CCW Outlet Relief Valve.	150 psig ± 4.5 psig
	Discharges to the CCW Return Header.	

Question: 73

Which ONE (1) following procedures is used to provide instructions in the event of a cask drop when loaded with spent fuel in Dry Shielded Canister (DSC)?

a. AOP-005, Radiation Monitoring System

b. AOP-008, Accidental Release of Liquid Waste

c. AOP-013, Fuel Handling Accident

d. AOP-028, ISFSI Abnormal Events

Answer:

d. AOP-028, ISFSI Abnormal Events

QUESTION NUMBER: 73
TIER/GROUP: ***RO*** 1/3 ***SRO*** 1/3
K/A: 036 2.2.28

Knowledge of new and spent fuel movement procedures (Fuel Handling Accident).

K/A IMPORTANCE: ***RO*** 2.6 ***SRO*** 3.5
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: AOP-028-01

STATE the purpose of AOP-028.

REFERENCES: AOP-028

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-028-01 004

JUSTIFICATION:

a. Plausible since this event could result in increased radiation levels, but AOP-028 specifically addresses this condition.

b. Plausible since this event could result in release, but AOP-028 specifically addresses this condition.

c. Plausible since this event could occur while refueling, but AOP-028 specifically addresses this condition.

d. **CORRECT** Entry conditions for AOP-028 include cask drop when loaded with spent fuel in dry shielded canister.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of the entry conditions / purpose of AOPs

REFERENCES SUPPLIED:

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE:

   This procedure provides the instructions necessary for the operator to respond to any abnormal ISFSI condition under all plant conditions.

2. ENTRY CONDITIONS:

   Any indication of an ISFSI abonormal condition as follows:

   - Blockage of the HSM Drains, Air Inlets, or Air Outlets
   - High Radiation at the Surface of the HSM
   - Cask Drop when Loaded with Spent Fuel in Dry Shielded Canister
   - Damage to HSM Air Outlet Shield Block

\- END -

Question: 74

Given the following conditions:

- The unit is in Mode 2.
- PZR level transmitter LT-460 failed low and was removed from service.
- The PZR high-high level and low level bistables associated with LT-460 were placed in the TRIPPED condition.
- PZR level channel selector switch LM-459 was selected to "461 REPL 460".

Which ONE (1) of the following describes the function provided by PZR level transmitter LT-461 under these conditions?

a. Energizes the backup heaters on a high level deviation

b. Decreases charging pump speed on an increasing level

c. Deenergizes the proportional and backup heaters on a low level

d. Trips the reactor on a high-high level

Answer:

c. Deenergizes the proportional and backup heaters on a low level

QUESTION NUMBER: 74
TIER/GROUP: *RO* 2/2 *SRO* 2/2
K/A: 011K6.04

Knowledge of the effect of a loss or malfunction on the Operation of PZR level controllers

K/A IMPORTANCE:	***RO***	3.1	***SRO***	3.1
10CFR55 CONTENT:	***55.41(b) RO***	7	***55.43(b) SRO***	

OBJECTIVE: PZR-08

EXPLAIN the component operation associated with each switch position for the PZR and PRT System switches and controls.

REFERENCES:
AOP-025
SD-059
SD-011

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number PZR-07 003

JUSTIFICATION:

a. Plausible since LT-461 could perform this function if switch in 461 REPL 459 position, but this function is performed by LT-459 under these conditions.

b. Plausible since LT-461 could perform this function if switch in 461 REPL 459 position, but this function is performed by LT-459 under these conditions.

c. **CORRECT** LT-461 performs all functions normally performed by LT-460. This includes isolating letdown and deenergizing all heaters on a low level. Input to RPS is independent of control switch position.

d. Plausible since 2/3 high levels would trip the reactor if above P-7, but plant is below 10% power.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of the Pressurizer Level Control system design attributes

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

Pressurizer Level Transmitter Failure

(Page 1 of 3)

1.	Check CVC-460 A&B, LTDN LINE STOP - CLOSED	Go To Step 3.
2.	Place CVC-460 A & B In The CLOSE Position	
3.	Place Pressurizer Level Controller, LC-459G, In The MAN Position	
4.	Restore PRZR LEVEL To Between 22% TO 53%	
5.	Check Number Of Operable PZR Level Channels - GREATER THAN ONE	Go To Step 12.
6.	Place LM-459, PZR LEVEL, In The Switch Position For The Alternate Channel Below:	

FAILED CHANNEL	SWITCH POSITION
LT-459	461 REPL 459
LT-460	461 REPL 460

7.	Verify Selector Switch LR-459 - SELECTED TO THE CONTROLLING CHANNEL • REC 459 • REC 461	

There is one alarm associated with each channel of LTOPP. It actuates for 3 reasons: (1) RCS temperature is <360°F and LTOPP is not selected on the key switch for OVERPRESSURE PROTECTION, (2) The PORV has received an actuation signal based upon current pressure and temperature or (3) the associated Block valve is shut.

5.1.5 PZR Level Control (PZR-Figure 12)

PZR level is controlled by controlling charging pump speed. The level is programmed to ramp up as Tavg increases by LC-459G. This maintains approximately constant mass in the RCS as Tavg is increased and the coolant in the RCS expands. Level program is 22.2% at Tavg of 547°F and 53.3% at Tavg of 575.4°F.

There are 3 PZR level channels LT-459, LT-460 and LT-461. LC-459G the PZR level controller is normally fed by level channel LT-459 but can be replaced by LT-461 with a selector switch on the RTGB. The output of LC-459G is then fed to the charging pump speed controllers to control speed of the charging pump if their controllers are selected to Auto.

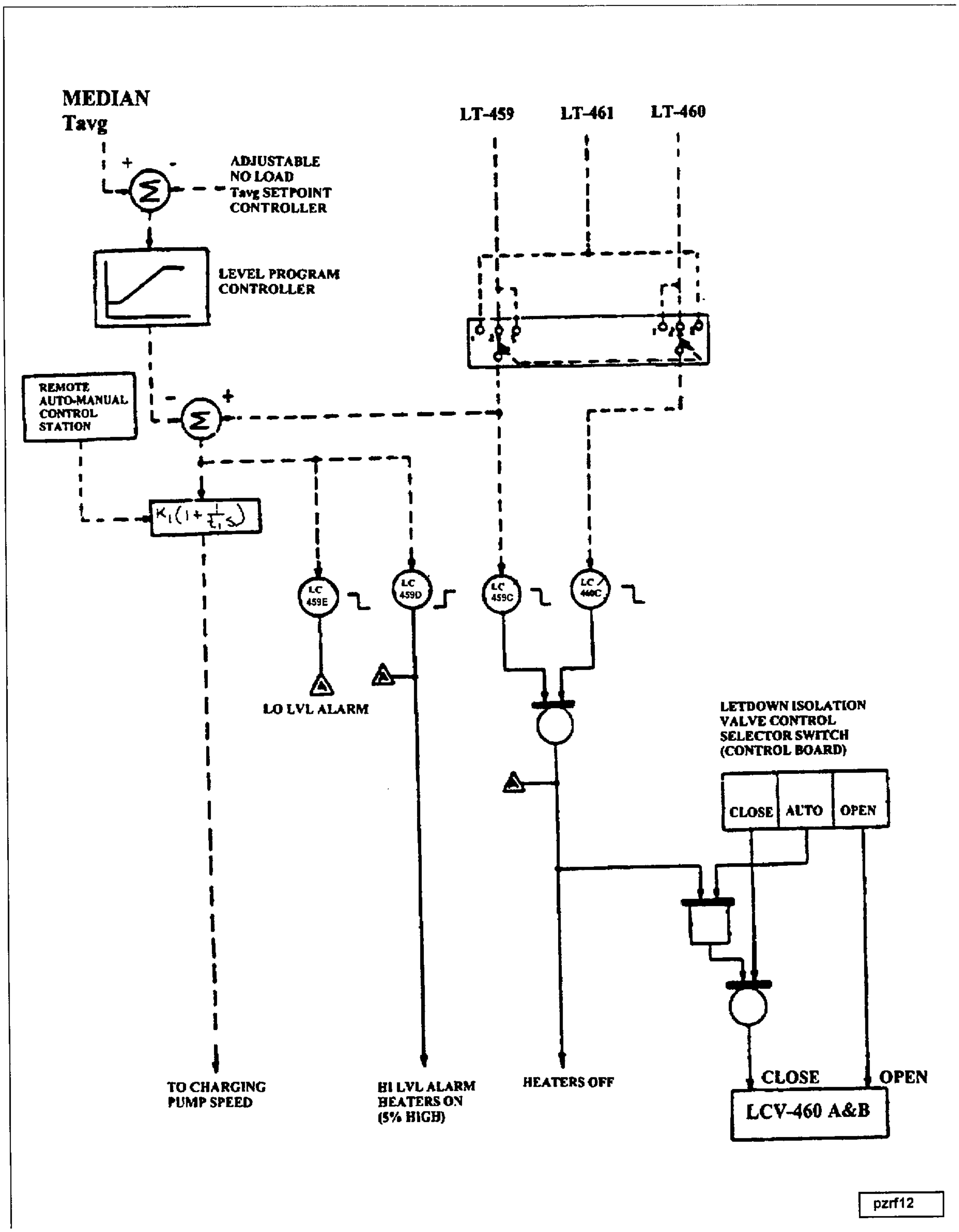
If PZR level increases 5% above program LC-459D will turn on the backup heaters and sound an annunciator for High Level Heaters on.

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would deenergize and any backup heater in manual would remain energized.

LEVEL CONTROLLER

PZR-FIGURE-12 (Rev . 0)



INFORMATION USE ONLY

PZR-07 003

Given the following plant conditions:

- Mode 1 at 100% RTP
- PZR level transmitter LT-459 fails low and is being removed from service
- PZR level channel selector switch LM-459 is selected to "461 REPL 459"

Which ONE (1) of the following describes the function provided by PZR level transmitter LT-461 with the level channel selector switch LM-459 is selected to "461 REPL 459"?

A. Changes the PZR high level reactor trip to 1/3 logic.

B. Can deenergize the backup heaters in AUTO or MANUAL.

✓C. Provides input to PZR low level letdown isolation.

D. Can energize proportional heaters upon 5% level deviation.

Question: 75

Given the following conditions:

- Reactor power was initially 100%.
- All CCW flow has been lost to the RCPs and a reactor trip has been initiated.

Which ONE (1) of the following nuclear instrument indications would warrant entry into FRP-S.1, "Response To Nuclear Power Generation/ATWS"?

a. **BOTH** source range channels are energized and intermediate range startup rate is +0.1 dpm

b. Power range indicates 3%

c. Source range startup rate is +0.3 dpm

d. **NEITHER** source range channel is energized and intermediate startup rate is -0.1 dpm

Answer:

a. **BOTH** source range channels are energized and intermediate range startup rate is +0.1 dpm

QUESTION NUMBER: 75
TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/1
K/A: 029EA2.01

Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation

K/A IMPORTANCE: ***RO*** 4.4 ***SRO*** 4.7
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: FRP-S.1-02

RECOGNIZE the selected entry level conditions of FRP-S.1.

REFERENCES: CSFST

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number FRP-S.1-02 005

JUSTIFICATION:

a. **CORRECT** Either the PR instruments indicating > 5% (RED) or the IR instruments indicating a SUR > 0.0 dpm (ORANGE) would require entry into FRP-S.1.

b. Plausible since excessive power range level indicates that the reactor is not tripped, but power range is below the 5% level which warrants enty into FRP-S.1.

c. Plausible since source range startup rate is greater than 0.0 and CSF-1 is not satisfied, but entry into FRP-S.2 vice S.1 is warranted.

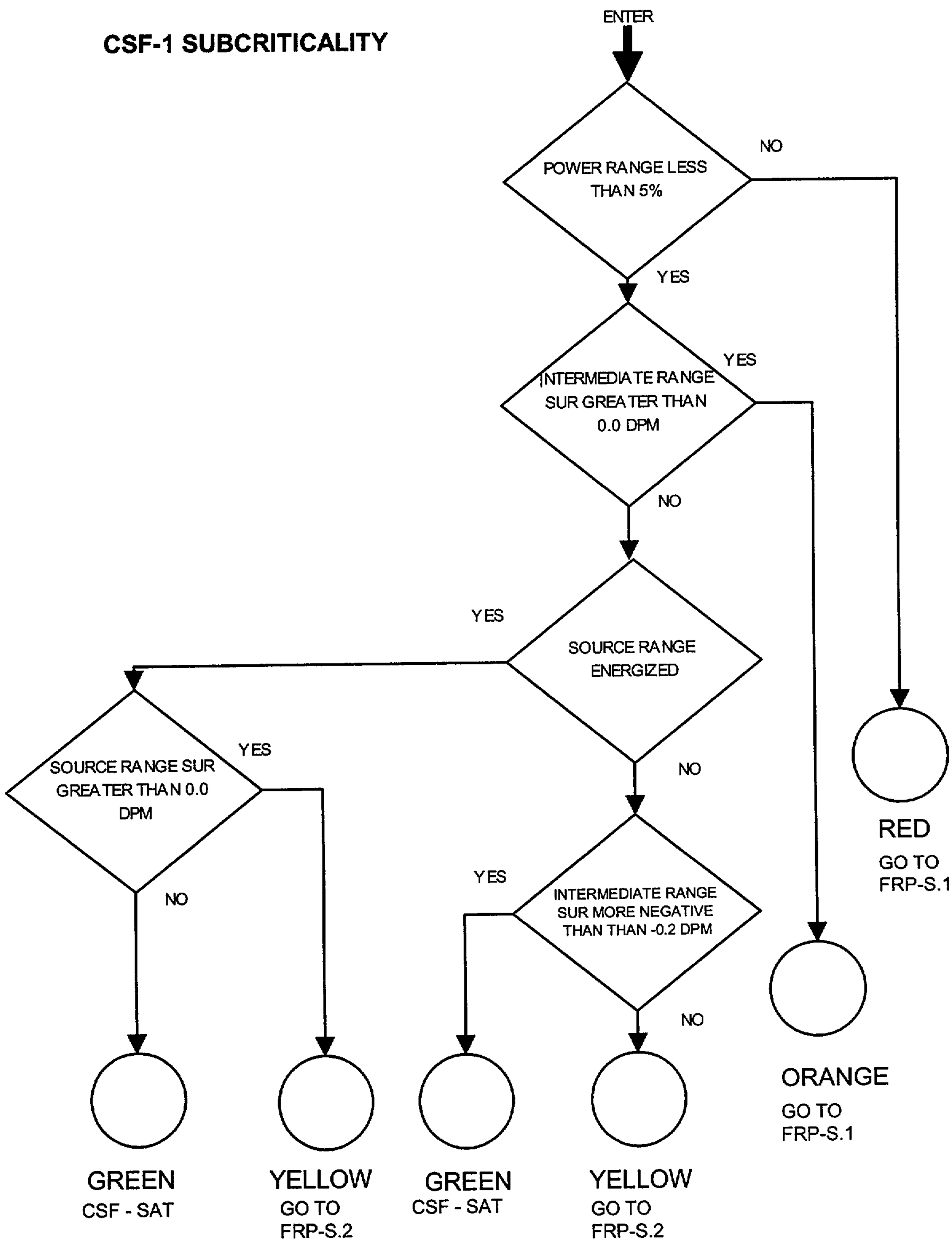
d. Plausible since with the source range not energized and intermediate range startup rate less negative than -0.2 dpm and CSF-1 is not satisfied, but entry into FRP-S.2 vice S.1 is warranted.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of the entry conditions for FRPs

REFERENCES SUPPLIED:

CSF-1 SUBCRITICALITY



Question: 81

Given the following conditions:

- The unit is operating at 100% power.
- Channel III PZR Pressure PT-457 is failed, with all bistables in the TRIPPED condition.
- An electrical fault occurs which results in a loss of Instrument Bus 2.

Which ONE (1) of the following describes the impact that the loss of Instrument Bus 2 has on the plant?

a. A reactor trip and SI occur and **BOTH** trains of Engineered Safeguards loads are automatically started by the sequencers

b. A reactor trip and SI occur, but **ONLY** Train 'A' Engineered Safeguards loads are automatically started by the sequencers

c. A reactor trip and SI occur, but **ONLY** Train 'B' Engineered Safeguards loads are automatically started by the sequencers

d. A reactor trip occurs, but **NO** SI occurs.

Answer:

c. A reactor trip and SI occur, but **ONLY** Train 'B' Engineered Safeguards loads are automatically started by the sequencers

QUESTION NUMBER: 81
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 013K2.01

Knowledge of bus power supplies to the ESFAS/safeguards equipment control

K/A IMPORTANCE: ***RO*** 3.6 ***SRO*** 3.8
10CFR55 CONTENT: ***55.41(b) RO*** 8 ***55.43(b) SRO***

OBJECTIVE: ESF-06

LIST power supplies for the major ESFAS components as listed in the EDPs.

REFERENCES: SD-006
AOP-024

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number ESF-09 015

JUSTIFICATION:

a. Plausible since a reactor trip and SI will occur and train 'A' sequencer is powered by IB 7, but IB 7 gets power from IB2 so only train 'B' sequencer has power available.

b. Plausible since a reactor trip and SI will occur, but only train 'B' sequencer has power available.

c. **CORRECT** A loss of Instrument Bus 2 will cause 2/3 low pressure conditions which will generate a SI and reactor trip. Only train 'B' sequencer has power available since it is powered by IB 3 and train 'A' is powered by IB 7, which gets power from IB 2.

d. Plausible since a reactor trip will occur and some ESF functions, such as CV high pressure, require power to actuate, but low pressure goes to its tripped conditions and an SI will also occur.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 4

Analysis of the effect of multiple failures on the RPS and ESFAS

REFERENCES SUPPLIED:

2.2 Design Basis2.2 Design Basis

Combined with the Reactor Trip System, the ESFAS is designed to perform all protective actions associated with the Reactor Safeguards and Protection System (RSPS).

The RSPS receives redundant inputs that include process variables, nuclear measurements, and equipment operational status. These inputs are provided by the Nuclear Instrumentation System, Analog Process Instrumentation and Control System, the Electrical Power Distribution System, and the Turbine Control System. Inputs to the Reactor Trip System are developed by redundant coincidence logic within the Reactor Protection Relay Racks, while inputs to the ESFAS are developed via similar logic within the Safeguards Relay Racks. These inputs enable the Reactor Trip System and the ESFAS to perform their respective protective actions.

2.3 System Description2.3 System Description

The ESFAS consists of two completely independent trains (A and B). The trains receive DC power from "A" and "B" station batteries respectively. AC power is supplied by station battery backed instrument buses. Instrument Bus 7A supplies Train "A" and Instrument Bus 3 supplies Train "B". Both trains have a complete set of matrices and both receive the same actuating signals. All of the circuits are redundant unless otherwise noted.

The bistables generating the input signals, with the exception of the Hi-Hi Containment Pressure bistables, are designed to actuate upon a loss of power.

3.0 COMPONENT DESCRIPTION3.0 COMPONENT DESCRIPTION

3.1 ESFAS Cabinets3.1 ESFAS Cabinets

Two trains of ESFAS cabinets are provided. They operate completely independent from each other. Switches, pushbuttons and status lights are provided for periodic on-line testing of the ESFAS circuits.

The cabinets, located in the E-1 and E-2 room, are supplied power from independent 125 VDC supplies. The DC power for Train "A" is supplied from MCC "A"; Train "B" is supplied from MCC "B". MCC"A" and "B" are located in the A/B Battery Room. This DC power is used to actuate components.

Instrument Bus 7A supplies AC power to Train "A" while Instrument Bus 3 supplies

Discussion (Continued)

Instrument Power:

On a loss of Instrument Power (Secondary busses), all instrument signals in that Channel will be reduced to a zero state. Thus, for example, Steam Generator A pressure and VCT levels will indicate zero. A zero input signal trips low bistables and inhibits high bistables from providing a trip output to a protection matrix. This will not normally result in, or prevent, protective actuations (Since most matrices are 2/3, a loss of a single channel will change them to 2/2 or 1/2.) Analog Control systems, however, will respond to the signal change (such as a zero S/G Level signal causing an increase in FW Flow).

Control Power:

On a loss of Control Power (Primary bus), all Bistables in that Channel will go to their fail-safe condition. (Exceptions to this are CV Hi-Hi pressure, and the P-6 bistables which are energize to actuate) Thus, the 2/3 matrices will become 1/2, etc. This will not normally result in a protective actuation. However, if for example, a Channel II trip already exists (from some other cause), and Channel I experiences a loss of Control Power, two trips will exist for that protective feature and an actuation will occur. Loss of Primary bus will also result in a loss of Secondary bus (Instrument Power), but the Bistables will trip anyway. However, the indications themselves will fail. This will provide conflicting information to the operator and may cause a plant control response.

2. Safeguard Racks - Control Features:

IB 7 (Train A) and IB 3 (Train B) supply power to the interposing relays for the loads started from their sequencer. A loss of either of these busses will prevent that sequencer from starting its loads. (Note: These loads may still be started manually by the operator after the EDG has loaded the bus)

ESF-09 015

Given the following plant conditions:

- Mode 1 at 100% RTP

- An electrical fault occurs which results in a loss of power to Instrument Bus 3

Which ONE (1) of the following describes the impact that the loss of Instrument Bus 3 has on the automatic operation of the Engineered Safeguards Features (ESF) Actuation System?

A. Neither train of the Engineered Safeguards Actuation System is affected

✓B. The sequencers will not be able to automatically start any train "B" Engineered Safeguards Loads

C. The sequencers will not be able to automatically start any train "A" Engineered Safeguards Loads

D. The sequencers will not be able to automatically start any train "A" or "B" Engineered Safeguards Loads

Question: 82

Given the following conditions:

- The plant is in Hot Shutdown.
- A loss of 4KV Bus 2 occurs.

Which ONE (1) of the following identifies plant equipment that is affected by the power loss?

a.
- Reactor Coolant Pump 'B'
- Station Service Transformer 2B

b.
- Reactor Coolant Pump 'C'
- Station Service Transformer 2A and 2F

c.
- Main Feedwater Pump 'B'
- Station Service Tranformer 2D

d.
- Main Feedwater Pump 'B'
- Reactor Coolant Pump 'C'

Answer:

b.
- Reactor Coolant Pump 'C'
- Station Service Transformer 2A and 2F

QUESTION NUMBER: 82
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 062K2.01

Knowledge of bus power supplies to the Major system loads

K/A IMPORTANCE: ***RO*** 3.3 ***SRO*** 3.4
10CFR55 CONTENT: ***55.41(b) RO*** 3 ***55.43(b) SRO***

OBJECTIVE: KVAC-06

LIST power supplies for the major 230/4KV Electrical System components as listed in the EDPs.

REFERENCES: EDP-001

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number KVAC-06 003

JUSTIFICATION:

a. Plausible since these are both 'B' equipment, but the transformer is supplied by 4 KV Bus 1 and the RCP by Bus 4.

b. **CORRECT** Major loads supplied by 4 KV Bus 2 include Station Service Transformers 2A and 2F and RCP 'C'.

c. Plausible since the FWP is identified as 'B', but the FWP and transformer are both supplied by Bus 4.

d. Plausible since RCP 'C' is supplied by this Bus and the FWP is identified as 'B', but is supplied by Bus 4.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of major component power supplies

REFERENCES SUPPLIED:

1.0 4160V AC Buss No. 1

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
Reactor Coolant Pump "A"	1	52/1	109
Circulating Water Pump "A"	2	52/2	811
Feedwater Pump "A"	3	52/3	615
Station Service Transformer 2B	4	52/4	933
Heater Drain Pump "A"	5	52/5	625
Condensate Pump "A'	6	52/6	605
Unit Aux to 4KV Bus 1	7	52/7	926
PTs and Fan Equipment	8	N/A	948
PTs and Fan Equipment and Metering	9	N/A	948
4KV Bus 1 - 2 Tie	10	52/10	928

2.0 **4160V BUSS NO. 2**

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
PTs and Fan Equipment	11	N/A	948
Start-Up to 4KV Bus 2	12	52/12	927
Station Service Transformers 2A and 2F	13	52/13	932
Reactor Coolant Pump "C"	14	52/14	105

3.0 4160V AC BUSS NO. 3

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
Station Service Transformer 2C and 2G	15	52/15	934
PTs and Fan Equipment	16	N/A	949
Start-Up Transformer to 4KV Bus 3	17	52/17	929B
PTs and Fan Equipment	18	N/A	949

4.0 4160V AC BUSS NO. 4

Location: 4160V Switchgear Room

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
4KV Bus 3 - 4 Tie	19	52/19	931
Unit Aux to 4KV Bus 4	20	52/20	930
PTs and Fan Equipment	21	N/A	949
Condensate Pump "B"	22	52/22	606
Circulating Water Pump "B"	23	52/23	813
Feed to 4KV Bus 5	24	52/24	1344
Heater Drain Pump "B"	25	52/25	626
Feedwater Pump "B"	26	52/26	620
Reactor Coolant Pump "B"	27	52/27	101
Station Service Transformer 2D	28	52/28	1041

5.0 4160V AC BUSS NO. 5

Location: Turbine Bldg., 1st Level,
Grid Location 3B

Power Supply: As per RTGB Line Up

Loads:	Cubicle	Breaker	CWD
4KV Bus 4 to 4KV Bus 5	29	N/A	1344
PTs and Control Power Transformer	30	N/A	N/A
SPARE	31	52/31	N/A
Station Service Transformer 2E	32	52/32	1399
Circulating Water Pump "C"	33	52/33	815
SPARE	34	52/34	N/A

KVAC-06 003

Given the following plant conditions:

- The plant is in hot shutdown

- A partial loss of AC power has occurred

- The operating crew has diagnosed a loss of 4KV bus 1

Which ONE (1) of the following describes the plant equipment that is affected by the power loss?

✓A. Circulating Water Pump "A", Station Service Transformer 2B

B. Main Feedwater Pump "A", Circulating Water Pump "B"

C. Station Service Tranformer 2D, Heater Drain Pump "B"

D. Reactor Coolant Pump "A", Main Feedwater Pump "B"

Question: 83

In accordance with AOP-032, "Response To Flooding From The Fire Protection System," the concern for a fire water break in containment is ...

a. the adverse affects on safeguards equipment.

b. the thermal stress effects of water coming in contact with the reactor vessel.

c. the adverse impact on the instrumentation associated with systems in containment.

d. the unanalyzed dilution caused by the water in the event of a LOCA.

Answer:

d. the unanalyzed dilution caused by the water in the event of a LOCA.

QUESTION NUMBER: 83
TIER/GROUP: ***RO*** 1/3 ***SRO*** 1/3
K/A: WE15EK3.1

Knowledge of the reasons for the following responses as they apply to the (Containment Flooding) Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 2.9
10CFR55 CONTENT: ***55.41(b) RO*** 9 ***55.43(b) SRO***

OBJECTIVE: AOP-032-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in AOP-032 by explaining the basis of each.

REFERENCES: AOP-032

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-032-03 002

JUSTIFICATION:

a. Plausible since some safeguards equipment is located inside containment, but concern is dilution of LOCA water.

b. Plausible since fire water is much colder than the vessel, but thermal stresses on the vessel are an internal stress concern.

c. Plausible since some instruments may be affected, but the qualified post-accident instruments are designed to be in an adverse environment.

d. CORRECT Safeguards equipment will not be affected, but the water needs to be removed from the sump since it represents an unanalyzed condition that would dilute LOCA water before a sump recirc condition.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of basis for actions contained in EPPs

REFERENCES SUPPLIED:

BASIS DOCUMENT, RESPONSE TO FLOODING FROM THE FIRE PROTECTION SYSTEM

DISCUSSION:

The purpose of this procedure is to provide instructions to be followed in the event of flooding caused by a break in the Fire Protection System. This procedure is designed to isolate the break and deal with the resulting water prior to operability concerns arising. This procedure is not intended for small system leaks that do not pose a threat to safety related equipment or that can be handled by the floor drain systems.

A break may range is size from several gpm to the design flow of the system (5,000 gpm at 125 psig), to maximum runout flow (≈7,500 gpm at 80 psig). The larger size breaks can do significant damage and can quickly overwhelm the capacity of installed sump pumps and floor drains.

There are many symptoms, but the first, and most likely, will be when the low header pressure auto-starts one or both of the fire pumps with no corresponding alarm from a system actuation.

Since a leak in one area has different required actions than a leak in another, the most important procedural action is to determine where the break is. The location will normally be discovered by verbal reports of geysers/flooding or area sump high level alarms. If these do not exist, then a walkdown inspection must be performed.

This procedure is divided into three main parts; break in the Auxiliary Building, break in Containment and break at the Intake structure.

The most serious location for a break is in the Aux Building. This is due to the fact that when water level reaches a certain height, both trains of Safeguards Equipment can be rendered inoperable. This event is further compounded by the fact that all spilled water may become contaminated and must be treated so until proven otherwise. (Note that other major system breaks in the Aux Building are addressed by their appropriate procedures - AOP-008 for LWS, AOP-014 for CCW, and AOP-022 for SWS).

The break in Containment is a situation where local inspection could be delayed. Safeguards equipment will not be affected, but the water needs to be removed from the Containment Sump since it represents an unanalyzed condition that would dilute LOCA water before a sump recirc condition.

The break at the Intake Structure is easily isolable and results in restoring the Unit 2 FPS from another source using OP-801, Fire Water System. (Note that this AOP does not refer to AOP-22 for Service Water Pit Breaks to avoid needless isolation of Service Water.)

Question: 84

Given the following conditions:

- Inverter 'C', is being shut down in accordance with OP-601, "DC Supply System."
- The N-43 DROPPED ROD MODE switch is placed in the BYPASS position prior to aligning PP-26 to its alternate supply (IB-3).

Which ONE (1) of the following describes the consequences of failing to place the switch in the BYPASS position?

a. A turbine runback may occur due to an Instrument Bus transient

b. A reactor trip and safety injection may occur due to an Instrument Bus transient

c. The inverter power supply breaker may trip open

d. The backup power supply breaker may trip open when attempting to close

Answer:

a. A turbine runback may occur due to an Instrument Bus transient

QUESTION NUMBER: 84
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/1
K/A: 063 2.1.32

Ability to explain and apply all system limits and precautions (DC Electrical).

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.8
10CFR55 CONTENT: ***55.41(b) RO*** 7 ***55.43(b) SRO***

OBJECTIVE: DC-10

EXPLAIN the operation of the DC Electrical System.

REFERENCES: OP-601

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number DC-10 003

JUSTIFICATION:

a. **CORRECT** Due to the power transient a momentary signal may be generated which results in a turbine runback signal.

b. Plausible since the concern is that a momentary signal may be generated due to the power transient, but it would affect the runback circuitry.

c. Plausible since a power transient may occur which could cause a trip of a breaker, but concern is that a momentary runback signal would be generated.

d. Plausible since a power transient may occur which could cause a trip of a breaker, but concern is that a momentary runback signal would be generated.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 2

Knowledge of basis for actions taken in System Operating Procedures

REFERENCES SUPPLIED:

7.8 Shutdown of Inverter "C" INIT VERI

7.8.1 Initial Conditions

1. This revision has been verified to be the latest revision available.

Name (Print)	Initial	Signature	Date

7.8.2 Instructions

Step	Action	INIT	VERI
1.	Place N-43 DROPPED ROD MODE switch to BYPASS to prevent a possible Turbine Runback.	______	______
2.	Verify PP-26 is on Backup power from Instrument Bus 3.	______	
3.	Place N-43 DROPPED ROD MODE switch to NORMAL.	______	______
4.	Open the AC Output Breaker.	______	______
5.	Open the DC Input Breaker.	______	______

	Initials	Name (Print)	Date
Performed by:	______	______________________	______
	______	______________________	______
	______	______________________	______

Approved by: ______________________________ ________
Superintendent Shift Operations Date

Question: 85

Given the following conditions:

- A batch release of Waste Condensate Tank 'E' is scheduled to be performed.
- The Waste Condensate Recirc Pump is out-of-service.

Waste Condensate Tank 'E' ...

a. can be recirculated after transferring to Waste Condensate Tank 'C'.

b. **CANNOT** be recirculated unless transferred to Waste Condensate Tank 'D'.

c. can be recirculated using Waste Condensate Pump 'B'.

d. **CANNOT** be recirculated until the Waste Condensate Recirc Pump is repaired.

Answer:

d. **CANNOT** be recirculated until the Waste Condensate Recirc Pump is repaired.

QUESTION NUMBER: 85
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 068 2.3.11

Ability to control radiation releases (Liquid Radwaste).

K/A IMPORTANCE: ***RO*** 2.7 ***SRO*** 3.2
10CFR55 CONTENT: ***55.41(b) RO*** 13 ***55.43(b) SRO***

OBJECTIVE: WD-03

Describe the major flow path(s) through the Waste Disposal System. Liquid Waste Disposal

REFERENCES: SD-023

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number RNP-RO-2000 85

JUSTIFICATION:

a. Plausible since 2 Waste Condensate Tanks use their individual pumps to recirc the tank. Waste Condensate Tanks with this capability are 'A' and 'B'.

b. Plausible since transferring contents of Waste Condensate Tank 'E' to a different tank would allow use of Waste Condensate Pump 'C' or 'D' for discharge. Waste Condensate Tanks 'C', 'D', and 'E' must use the Waste Condensate Recirc Pump.

c. Plausible since either Waste Condensate Tank 'A' or 'B' can be recirculated with either Waste Condensate Pump 'A' or 'B'. Waste Condensate Tanks 'C', 'D', and 'E' must use the Waste Condensate Recirc Pump.

d. **CORRECT** Waste Condensate Tanks 'C', 'D', and 'E' can only be recirculated using Waste Condensate Recirc Pump.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of system operations to determine acceptable alternative liquid waste flowpaths

REFERENCES SUPPLIED:

Type	Horiz. Cent.
Design flow rate	20 gpm at 3500 RPM
Design head	100 ft
Material of construction, wetted surfaces	Austenitic SS

The WCT pumps "A" and "B" are horizontal centrifugal type. These pumps, located by the corresponding "A" and "B" waste condensate tanks, are used to pump liquids to the WHUT, to condenser circulating water or to recirculate back to the respective WCT.

3.14 WCT Pumps "C" and "D"

Manufacturer	Gould
Type	Horiz. Cent.
Design flow rate	55 gpm at 3500 rpm
Design head	110 ft
Material of construction, wetted surfaces	Austenitic SS

There are two pumps provided to release wastes from "C", "D" and "E" WCTs. These pumps transfer liquids to the WHUT, to the polishing demineralizers for processing or to the condenser circulating water for discharge.

3.15 WCT Recirculating Pump

Manufacturer	Gould
Model	3196 MT
Number	1
Type	Horiz. Cent.
Design flow rate	275 gpm at 1750 rpm
Design head	110 ft
Material of construction, wetted surfaces	Austenitic SS

A pump is provided to recirculate liquid waste from tanks "C", "D" and "E" for sampling prior to discharge. This pump is located in the same building area as "C" and "D" WCT pumps.

3.16 CHT Pump

Manufacturer	Crane Co.
Number	1
Type	Rotary screw
Design flow rate	Variable
Design head	Variable
Material of construction, wetted surfaces	Austenitic SS

The CHT pump is used for discharging stored waste concentrates from the CHT to the drumming room for drumming. This pump is located in the Auxiliary Building in a cubicle

Question: 86

Given the following conditions:

- The plant is being started up with the Feed Water Regulating Valves and Feed Water Regulating Bypass Valves all open.
- A Reactor Trip occurs.
- RCS Tavg stabilizes at no load Tavg.
- The Feed Water Regulating Valves automatically close.

Which ONE (1) of the following identifies the expected position of the Feed Water Regulating Bypass Valves (FRBVs) and the Feed Water Block Valves (FBVs)?

	FRBVs	**FBVs**
a.	Open	Open
b.	Open	Closed
c.	Closed	Open
d.	Closed	Closed

Answer:

a.	Open	Open

QUESTION NUMBER: 86

TIER/GROUP: *RO* 2/1 *SRO* 2/1

K/A: 059K4.19

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Automatic feedwater isolation of MFW

K/A IMPORTANCE: *RO* 3.2 *SRO* 3.4

10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: FW-09

EXPLAIN the normal operation of the Feedwater control systems. Include function, instrumentation, interlocks, annunciators, and setpoints.

REFERENCES: SD-027

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number FW-09 004

JUSTIFICATION:

a. **CORRECT** Only the FRVs receive a close signal on a reactor trip and low Tavg. The FRBVs and FBVs will close on an SI signal.

b. Plausible since the position of the FRBVs is correct, but the FBVs will also be open.

c. Plausible since the position of the FBVs is correct, but the FRBVs will also be open.

d. Plausible since both sets of valves do receive automatic close signals, but not from a reactor trip with low Tavg.

DIFFICULTY:

Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of plant response to trip to determine FW system response

REFERENCES SUPPLIED:

These flow control valves (FCV-478, -488, -498) regulate flow of feedwater to the steam generators to maintain a specified programmed level. The main feedwater regulating valves (FRVs) are used from approximately 15% load to 100% load. The main feedwater regulating valve bypass valves are used during low load conditions for finer feedwater flow control.

The FRVs are 12 inch air operated plug and cage type valves. The cage has offset variable size ports which act as an orifice. The balanced plug attached to the stem moves up and down to uncover/cover these ports to control flow through the valve. This type of valve internals provides a linear flow change throughout the length of valve operator travel (≈3 1/16 inch). Each valve can be controlled automatically via the Hagan control system through the RTGB controller, with input from the steam generator water level control system. Each valve can also be controlled manually from the RTGB Hagan controller using pushbuttons to open or close the FRV. Operators also have the capability to operate the valves locally using an installed reverse acting manual handwheel. When taking local-manual handwheel control of the valve in accordance with OP-403, the handwheel is rotated clockwise to open and counterclockwise to close. The time limits of Technical Specification 3.7.3 are applicable during this evolution. The air operator is constructed such that the diaphragm is mounted to the frame, and as air is supplied to the actuator, the yoke is pulled up against spring tension (the stem is attached to the yoke).

The FRVs auto-close from the following:

- Feedwater Isolation signal (Safety Injection), all FRVs close
- Reactor Trip with low Tavg (554°F), all FRVs close
- High-High steam generator level (2/3 ≥75%), the FRV associated with the high-high level closes

3.6 Main Feedwater Regulating Valves Bypass Valves

These flow control valves (FCV-479, -489, -499) regulate flow of feedwater to the steam generators under manual control from ten-turn potentiometers located on the RTGB. The main feedwater regulating valve bypass valves are used during low load conditions(<≈15%).

The main feedwater regulating valve bypass valves are four inch air operated valves. The function similar to the FRVs, relative to their flow characteristics and local manual valve operation.

The main feedwater regulating valve bypass valves auto-close from the following:

- Feedwater Isolation signal (Safety Injection), all bypass valves close

The tubes in the heaters are horizontal U-tubes with feedwater flowing through the tubes and extraction steam around the tubes. These heaters may be removed from service individually by a 3-way bypass (FW-3A or -B) and manual isolation valves (FW-4A or -B).

Feedwater heater level indication is provided by either a sight glass, or magnetically coupled level indicator. The sight glass provides direct indication by seeing the actual liquid level through the glass. These types of indications are being replaced, as needed, by Penberthy magnetically coupled level indicators. The Penberthy indicators are not susceptible to the clouding and leaks of the glass type. The magnetic type indicator is a sealed tube. It has a float inside the pressure boundary that is magnetically coupled to a follower outside the pressure boundary to provide level indication. Liquid level changes in the feedwater heater cause the float to rise or fall in the sealed chamber, and the follower rises and falls with the float.

The feedwater enters the steam generator through a header pipe in the form of a ring that distributes the incoming water via inverted "J" nozzles located on the ring. The "J" nozzles are arranged on the feedwater ring to distribute approximately 80 percent of the feedwater toward the hot leg side of the steam generator. This feedwater mixes with recirculated water within the steam generator. This mixture flows down between the shell and down comer (tube bundle wrapper) to the bottom where it enters the tube area. More information concerning the steam generators can be found in SD-048, Steam Generator System.

3.3 Feedwater/Condensate Recirculation (BOP Cleanup)

During periods of cold shutdown a feedwater recirculating line can be utilized to reduce corrosion product buildup in the feedwater system and condensate system using the condensate polishers. Opening locked gate valve FW-232, globe valve FW-238 and turning the spectacle blind flange puts the flow path in service. The recirculating piping runs from the outlet side of the high pressure feedwater heaters 6A and B to the main condenser "B" via the downstream side of main steam dump (PRV-1324B-3).

3.4 Feedwater Header Block Valves

The feedwater header block valves (FW-V2-6A, -6B, -6C) isolate the feedwater pump discharge from each main feedwater regulating valve. Each valve is motor operated, and all three must be closed prior to start of the first main feedwater pump. This reduces the load on the feedwater pump motor, and prevents runout. The valves receive an auto-close signal from any Safety Injection signal. These valves are cycled during plant cooldown to prevent thermal binding of the 16 inch solid wedge gate valve.

3.5 Main Feedwater Regulating Valves

Question: 87

Given the following conditions:

- A small break LOCA has occurred.
- Due to problems with the Containment Cooling system, containment pressure increased to 6.1 psig.
- After establishing proper operation of the Containment Cooling system, containment pressure has been lowered to 3.2 psig.
- A step in one of the EPPs states:

"Depressurize RCS To Minimize RCS Leakage:

c. Check EITHER of the following:

PZR LEVEL - GREATER THAN 71% [60%]

OR

RCS SUBCOOLING – LESS THAN 45 °F [65 °F]

d. Stop RCS depressurization"

- As the RCS is being depressurized, PZR level is noted to be 62% and RCS Subcooling is 76 °F.

The RCS depressurization should ...

a. be stopped immediately.

b. continue until PZR level exceeds 71%.

c. continue until RCS subcooling drops below 65 °F.

d. continue until RCS subcooling drops below 45 °F.

Answer:

a. be stopped immediately.

QUESTION NUMBER: 87
TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1
K/A: 022K3.02

Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment instrumentation readings

K/A IMPORTANCE: ***RO*** 3.0 ***SRO*** 3.3
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: OMM-022-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in OMM-022 by explaining the basis of each.

REFERENCES: OMM-022

SOURCE: ***New*** [X] ***Significantly Modified*** [ ] ***Direct*** [ ]

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** Although adverse containment conditions no longer exist due to pressure being below 4 psig, adverse values are used until the EOP network is exited.

b. Plausible since containment pressure is below the adverse containment value, but adverse values are used until the EOP network is exited.

c. Plausible since adverse containment value must be used, but pressurizer level already exceeds the adverse value.

d. Plausible since containment pressure is below the adverse containment value, but adverse values are used until the EOP network is exited.

DIFFICULTY:
Comprehensive/Analysis [X] ***Knowledge/Recall*** [ ] ***Rating*** 3

Application of the uasge requirements for adverse containment conditions

REFERENCES SUPPLIED:

GLOSSARY

1.1 Definitions

1.1.1 **Adverse Containment Conditions** - If the CV pressure is greater than or equal to 4 psig, then adverse containment conditions exist. When adverse setpoints are provided, they will be enclosed by brackets: [ ].

1.1.2 **Core Cooling Mode** - When referenced for the current status of the RHR System, the system is aligned to remove decay heat via the normal pathway from RCS loop "B" hot leg back through RHR to the loop cold legs.

1.1.3 **Diverse** - (In reference to an indication) Having multiple indications of different types for indication of the same parameter. An example of diverse indications for the same parameter would be the use of S/G level increase, as well as AFW Line Flow Indication to verify that AFW Flow exists.

1.1.4 **Go To** - An action verb requiring the operator to leave the procedure or step currently in effect and implement the referenced procedure or step. The operator does not return to the EOP or AOP unless explicitly directed to by the procedure transitioned to.

1.1.5 **Injection Mode** - When referenced for the current status of the RHR System, the system is aligned to take a suction on the RWST and discharge to the loops. (Normal at-power RHR line up)

1.1.6 **Normal** - Describes a condition in which the parameter under consideration is within a range that can be expected during routine plant operation or is being controlled in accordance with approved plant procedures. When making this determination previous trends should be used. (RAIL 94R0296)

1.1.7 **Nuclear Safety Concern** - A condition is said to have a Nuclear Safety Concern when that condition has the possibility of jeopardizing the health and/or safety of the public to the extent that the SSO determines that action is needed to mitigate the condition.

1.1.8 **Perform** - An action verb directing the operator to accomplish certain actions using the referenced procedure and implicitly requiring the operator to remain in the procedure in effect. This action may be reinforced by the statement, "while continuing with this procedure".

8.3.10 Incorrect EOP Transition

1. Should the Operator determine that he is in an incorrect Path or EPP, he has two options:

   - If the incorrect transition is immediately recognizable **AND** no alterations of the WOG mitigative strategy have occurred, he may move back to the point in the Network where the incorrect transition has occurred.

   - If the incorrect transition is not immediately recognizable **OR** alterations in the mitigative strategy have occurred, the Operator should move to Path-1, Entry Point A, and start over.

2. During the rediagnosis described above, complete reactuation of the Engineered Safety Features is allowed, but not required. Reactuation of necessary safety features during rediagnosis is guided by the requirements of the applicable Foldout and Operator judgement based on the symptoms present.

8.3.11 Adverse Containment Conditions Usage

1. When adverse containment conditions develop, the use of adverse containment condition setpoints shall be initiated.

2. The use of adverse containment condition setpoints shall be maintained from that point forward, even when adverse containment conditions no longer exist.

3. An adverse containment condition setpoint may or may not be provided. The operator shall use a setpoint with no brackets if no setpoint within brackets is provided, even if adverse containment conditions exist.

8.3.12 Special EPP Priority

1. Certain contingency EPPs take precedence over FRPs because of their treatment of specific initiating events. In all such cases, this precedence is identified in a CAUTION or NOTE at the beginning of the EPP.

Question: 88

Given the following conditions:

- The unit is in Hot Shutdown.
- The Startup Transformer (SUT) is supplying all 4KV buses.
- A severe short has resulted in a loss of the 'B' DC Bus.

Which ONE (1) of the following describes the response of the emergency diesel generators (EDG's)?

	EDG 'A'	**EDG 'B'**
a.	Starts and loads	Does **NOT** start
b.	Does **NOT** start	Starts, but field fails to flash
c.	Starts and loads	Starts, but does **NOT** load
d.	Starts, but does **NOT** load	Starts and loads

Answer:

b.	Does **NOT** start	Starts, but field fails to flash

QUESTION NUMBER: 88
TIER/GROUP: *RO* 1/2 *SRO* 1/2
K/A: 058AK3.01

Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs

K/A IMPORTANCE: *RO* 3.4 *SRO* 3.7
10CFR55 CONTENT: *55.41(b) RO* 7 *55.43(b) SRO*

OBJECTIVE: EPP-026/27-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in EPP-26 by explaining the basis of each.

REFERENCES: EPP-27

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number EPP-026/27-14 001

JUSTIFICATION:

a. Plausible since the loss of DC control power does affect the 'B' EDG, but it affects it by not flashing the field or allowing the output breaker to close.

b. **CORRECT** The 'B' EDG will start, but field flashing will not be available due to no DC power. The 'A' train is not affected.

c. Plausible since the 'B' EDG will start and will not load due to no field flash or output breaker closure, but the 'A' EDG is not affected.

d. Plausible since the 'B' EDG will start, but the 'B' EDG field will not flash and the output breaker will not close.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Analysis of effect of loss of control power on operation of EDGs

REFERENCES SUPPLIED:

INFORMATION USE

ATTACHMENT 1

MAJOR EFFECTS / LOAD LIST

(Page 1 of 4)

Major Effects:

Reactor	Will trip due to loss of power to 52/RTB undervoltage coil.
Turbine	Will trip via 20/AST from Rx Trip (20/ET has lost power).
Generator	Will receive lockout signal. However, 86P cannot open OCB 52/8 & 52/9 due to the loss of their control power. This causes a Breaker Failure scheme which trips OCB 52/3, 52/6, 52/7, 52/12 and the downstream breakers on the Darlington SCPSA line. The Exciter Field Breaker will open.
4KV Busses 1 & 2	If initially on SUT, nothing will happen. If initially on UAT, the busses will auto-transfer due to the Rx Trip. In either case, 4KV busses 1 and 2 and all downstream busses and equipment will remain energized.
4KV Bus 3	Will remain energized on the SUT. 4KV Bus 3 and 480V Bus 3 will lose DC Control Power (including a loss of protective relaying).
4KV Busses 4 & 5	4KV Bus 4 will try to auto-transfer to Bus 3 but cannot due to the loss of DC Control Power. Thus, 4KV Busses 4 & 5 and all downstream busses and equipment will deenergize. 4KV Bus 4 and 480V Bus 4 will lose DC Control Power (including a loss of protective relaying). Control Power (and protective relaying) will remain for 4KV Bus 5 and 480V Bus 5.
Emergency Bus E-1	Will remain energized. SST 2F will lose cooling fans.
Emergency Bus E-2	Will remain energized on the SUT but will lose DC Control Power (including a loss of protective relaying). SST 2G will lose cooling fans.
DS Bus	Will remain energized with Control Power available.
EDG A	Remains available, if needed.
EDG B	Auto-starts due to loss of power to air start solenoids but will not field flash and output breaker will not close.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
25.	Place Normal RCS Letdown In Service Using OP-301, Chemical And Volume Control System (CVCS)	

****

CAUTION

If Starting Air has been cut in to the EDG for more than 2 minutes, the air distributor may be damaged and the EDG may fail during the next start attempt.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
26.	Perform The Following For EDG B:	
	a. Reset Fuel Racks as follows:	
	1) Slowly move the Reset Lever towards the EDG SW Heat Exchangers	
	2) Release the Reset Lever	
	3) Repeat Steps 26.a.1 and 26.a.2	
	b. Check the FUEL RACK TRIP Light - EXTINGUISHED	b. On the Engine Control Panel, depress the ALARM RE-SET pushbutton.
	c. Restore EDG B Starting Air to normal using the Starting Air Valve Lineup of OP-604, Diesel Generators A and B	
	d. Notify System Engineer that Starting Air Distributor has been in service while the EDG was running	

Question: 89

Given the following conditions:

- The plant is operating at 90% power.
- Control Bank "D" Step Counters indicate 198 steps.
- A check of the Rod Position indications for Control Bank "D" shows the following rod positions:

  D8 at 124"
  M8 at 116"
  H4 at 120"
  H8 at 121"
  H12 at 131"

Which ONE (1) of the following describes the status of the rods in Control Bank 'D'?

a. **BOTH** rods M8 and H12 are misaligned from the bank

b. **ONLY** rod M8 is misaligned from the bank

c. **ONLY** rod H12 is misaligned from the bank

d. All rods are within rod alignment limits

Answer:

c. **ONLY** rod H12 is misaligned from the bank

Replacement.

QUESTION NUMBER: 89

TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/1

K/A: 014A2.04

Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.9

10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: AOP-001-03

DEMONSTRATE an understanding of selected steps, cautions, and notes in AOP-001 by explaining the basis of each.

REFERENCES: AOP-001

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-001-03 08

JUSTIFICATION:

a. Plausible since rod H12 is misaligned and 198 steps corresponds to 123.75" so rod M8 would also be considered misaligned if requirement was to compare to group counter, but average IRPI is used.

b. Plausible since 198 steps corresponds to 123.75" so rod M8 would be considered misaligned if requirement was to compare to group counter, but average IRPI is used.

c. CORRECT With group position less than 200 steps, rod alignment must be within 7.5" of the average IRPI position in the bank. The average IRPI for these rods is 122.4", so only rod H12 is misaligned.

d. Plausible since the rods would be considered aligned if group position was greater than or equal to 200 steps, but rod H12 is considered misaligned since group position is less than 200 steps.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Determination of rod misalignment

REFERENCES SUPPLIED:

Question: 89

Given the following conditions:

- The unit is operating at 80% power.
- A misaligned rod in Group 2 of Control Bank ‘D’ has occurred.
- A recovery of the misaligned rod has begun.
- APP-005-E2, ROD CONT SYSTEM URGENT FAILURE, has just alarmed.

The power cabinet causing the urgent alarm is ...

a. 1AC.

b. 2AC.

c. 1BD.

d. 2BD.

Answer:

c. 1BD.

QUESTION NUMBER: 89

TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/1

K/A: 014A2.04

Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.9

10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: RDCNT-14

EXPLAIN the effect on the Rod Control System due to selected failures.

REFERENCES: AOP-001

SOURCE: ***New*** ☒ ***Significantly Modified*** ☐ ***Direct*** ☐

Bank Number HNP-RO-2000 76

JUSTIFICATION:

a. Plausible since other group of rods, Group 1, causes alarm, but group must be powered from same power cabinet.

b. Plausible since alarm caused by other group, and this is other bank, but group must be powered from same power cabinet.

c. **CORRECT** The other group of rods in the bank do not move when directed due to the lift coil disconnect switches being open and cause the urgent failure.

d. Plausible since this is the group of rods which are being moved and other rods in the group have the disconnect switch open, but caused by other group in same bank.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of rod control system design and operation during misaligned rod recovery

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

IMMOVABLE/MISALIGNED RODS

(Page 17 of 31)

NOTE

- APP-005-E2, ROD CONT SYSTEM URGENT FAILURE, will illuminate when the rod is moved due to all Lift Coil Disconnect Switches being off in the unaffected group.
- APP-005-B5, ROD BANKS A/B/C/D LO LIMIT, and APP-005-C5, ROD BANKS A/B/C/D LO-LO LIMIT, may illuminate when the rod is stepped in due to the P-A Converter input to the Rod Insertion Limit Monitoring System.

37. Align The Affected Rod As Follows:

    a. Depress AND hold the AUTO ROD DEFEAT Pushbutton

    b. Select the affected bank with the ROD BANK SELECTOR Switch

    c. Release the AUTO ROD DEFEAT Pushbutton

    d. Insert the rod at the rate specified in Step 28 to the Group Step Counter position recorded in Step 31

Question: 90

Given the following condiditons:

- Pressurizer pressure transmitter PT-457 has failed low and is being removed from service in accordance with the OWP.
- The OWP requires the low pressure bistables in the Hagan racks be placed in the TRIPPED condition.

Which ONE (1) of the following describes the verification required for this function?

a. Independent verification with the second initials "N/A'd" by the SSO

b. Independent verification with the second initials required

c. Concurrent verification with the second initials required

d. Functional verification with second initials required

Answer:

c. Concurrent verification with the second initials required

QUESTION NUMBER: 90
TIER/GROUP: *RO* 3 *SRO* 3
K/A: 2.1.29

Knowledge of how to conduct and verify valve lineups.

K/A IMPORTANCE:	***RO***	3.4	***SRO***	3.3
10CFR55 CONTENT:	***55.41(b) RO***	10	***55.43(b) SRO***	

OBJECTIVE: PLP-030-04

Given a set of conditions or components needing position checks or positioning actions DETERMINE the applicable functional testing and independent verification requirements.

REFERENCES: OPS-NGGC-1303

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number PLP-030-04 005

JUSTIFICATION:

a. Plausible since independent verification is identified on the OWP, but concurrent verification is what is used.

b. Plausible since independent verification is identified on the OWP, but concurrent verification is what is used.

c. CORRECT Concurrent verification is used where an improper positioning of a component has a high probability of resulting in an immediate plant trip or safety actuation. Selecting the wrong cabinet or bistable could cause a trip in this condition.

d. Plausible since functional verification can be used to verify bistable status change, but the bistable would already be tripped in this condition.

DIFFICULTY:
Comprehensive/Analysis ☐ ***Knowledge/Recall*** ☒ ***Rating*** 3

Knowledge of administrative requirements for independent verification

REFERENCES SUPPLIED:

6.3 Concurrent Verification Guidelines

6.3.1 CONCURRENT VERIFICATION, as defined in Section 3.0, is particularly useful in preventing an unintended plant response while conducting tests. CONCURRENT VERIFICATION positively identifies the correct unit, train, or component and ensures a review of the intended action is performed, eliminating the possibility of an unintended plant response due to a single personnel error. Identification of actions that, if performed improperly, could result in an immediate threat to safe and reliable plant operation, along with use of CONCURRENT VERIFICATION prior to performing such actions, will enhance plant reliability during system testing.

6.3.2 CONCURRENT VERIFICATION satisfies the requirements of INDEPENDENT VERIFICATION under the following circumstances unless specifically required by procedure and should be performed for the following:

1. Specific evolutions or actions where an improper positioning of a component has a high probability of resulting in an immediate plant trip, Safety System actuation <u>OR</u> could result in an immediate threat to safe and reliable plant operation. Examples of such evolutions are:

   - Installing or removing jumpers
   - Lifting or landing leads
   - Fuse removal
   - Operating a valve, switch or breaker
   - Placing bistable switches in the TRIP position

2. Positioning a throttle valve to a specific position when the valve has no accurate and discernable position indicator and is a component or system listed in Attachments 1(BNP), 2(HNP), and 3(RNP).

3. Performing position verification of locked valves which require a second position verification.

4. When INDEPENDENT VERIFICATION would invalidate initial component positioning (throttled valve position).

Question: 91

Given the following conditions:

- The unit has just experienced a reactor trip.
- **NO** SI equipment has actuated.
- 1/2 turbine stop valves are shut.
- 3/4 turbine governor valves are shut.
- RCS pressure is 1860 psig.
- Tavg is 542°F.
- All MSIVs are open.
- SG Pressures and Steam Flows are:

SG	PRESSURE	STEAM FLOW
'A'	925 psig	0.1×10^6 lbm/hr
'B'	935 psig	0.1×10^6 lbm/hr
'C'	845 psig	1.3×10^6 lbm/hr

The reactor is tripped, the turbine is ...

a. tripped, and SI is **NOT** required.

b. tripped, and SI is required.

c. **NOT** tripped, and SI is **NOT** required.

d. **NOT** tripped, and SI is required.

Answer:

c. **NOT** tripped, and SI is **NOT** required.

QUESTION NUMBER: 91
TIER/GROUP: ***RO*** 1/2 ***SRO*** 1/2
K/A: 007EK3.01

Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip

K/A IMPORTANCE: ***RO*** 4.0 ***SRO*** 4.6
10CFR55 CONTENT: ***55.41(b) RO*** 10 ***55.43(b) SRO***

OBJECTIVE: PATH-1-05

DEMONSTRATE an understanding of the steps of PATH-1 which require outside assistance

REFERENCES: SD-006
FRP-S.1

SOURCE: ***New*** ☐ ***Significantly Modified*** ☒ ***Direct*** ☐

Bank Number PATH-1-05 003

JUSTIFICATION:

a. Plausible since the steamflow SI coincidence has not been exceeded, but the turbine is not considered tripped.

b. Plausible since the turbine valves have received a close signal, but the turbine is not considered tripped.

c. CORRECT The turbine is only considered to be tripped if both stop valves or all 4 governor valves are closed, but no SI setpoints have been reached.

d. Plausible since the turbine is not tripped, but no SI setpoint has been exceeded.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comparison of abnormal response to reactor trip to determine equipment status and requirements

REFERENCES SUPPLIED:

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
2.	Check Turbine Trip As Follows: • BOTH Turbine Stop Valves - CLOSED OR • All Governor Valves - CLOSED	Perform the following: a. Manually trip the Turbine by simultaneously depressing the THINK and TURBINE TRIP Pushbuttons. b. IF Turbine will NOT trip, THEN run back Turbine at maximum rate until the Governor Valves are closed. c. IF Turbine can NOT be run back, THEN verify CLOSED the following: • All MSIVs • All MSIV BYPs
3.	Verify All AFW Pumps - RUNNING	

4.1.2 Reactor Coolant Temperature (ESF-Figure-1)4.1.2 Reactor Coolant Temperature

The RCS Low Tavg signal (2 of 3 channels below 543°F) is used to initiate the Safety Injection signal, when coincident with high steam flow; and close the Main Steam Isolation Valves, when coincident with high steam flow (i.e., generate the Steam Line Isolation Signal).

4.1.3 Steam Flow (ESF-Figure-1)4.1.3 Steam Flow

Hi Steam Flow (37.25% flow at no load to 20% load, increases linearly to 109% at full load) detected by at least one sensor on two of three steam lines, coincident with low Tavg (543°F) or low steam line pressure (614 psig), generates a Safety Injection signal and closes all MSIVs. Two flow controllers on each steam line are used to sense high steam line flow. This circuit is designed to detect steam line breaks downstream of the MSIVs.

4.1.4 Steam Line Pressure (ESF-Figure-1 & 3)4.1.4 Steam Line Pressure

Steam Line Pressure measurement is utilized for steam line break protection. Low steam line pressure (614 psig) in two of three main steam lines or Low Tavg (543°F) in two of three loops, coincident with high steam line flow in two-of-three main steam lines, will initiate the Steam Line Isolation and Safety Injection signals. This is to protect against: a steam line break upstream of the main steam check valves, a feed line break, and/or an inadvertent opening of a SG safety.

In addition, each steam line pressure measurement is compared with a main steam header pressure measurement to determine if a high steam line differential pressure exists. A coincidence of two-of-three steam line differential pressures (100 psid) in any one steam line, that is, steam line pressure lower than main steam header pressure, will initiate a Safety Injection signal.

The steam header pressure is electronically limited to a minimum value of 585 psig. Therefore, this SI signal must be blocked before a plant cooldown is started to prevent SI actuation when S/G pressures drop below 485 psig(approximately 467°F). The steam line differential pressure circuit detects faults upstream of the MSIVs. Since the steam line check valves prevent reverse flow to the faulted S/G, excessive steam line differential pressure does not close the MSIVs.

4.1.5 Containment Pressure (ESF-Figure-4 & 5)4.1.5 Containment Pressure

PATH-1-05 003

Given the following plant conditions:

- The Unit has just experienced a reactor trip
- Both turbine stop valves are shut
- Three turbine governor valves are shut
- RCS pressure is 1860 psig
- Tavg is 542°F
- S/G Pressures: A-895, B-915, C-835 psig
- Steam flows: A-0.1, B-0.1, C-1.3x10E6 lbm/hr
- No SI equipment has actuated

Which ONE (1) of the following contains the correct plant status and operator actions?

The reactor is tripped, the turbine is:

✓A. tripped, SI is not initiated or required; verify two charging pumps running.

B. not tripped, SI is not initiated or required; trip the turbine and verify two charging pumps running.

C. not tripped, SI is not initiated but is required; trip the turbine and initiate SI.

D. tripped, SI is not initiated but is required; initiate SI.

Question: 92

Given the following conditions:

- A reactor trip occurred due to a loss of offsite power.
- The plant is being cooled down on RHR per EPP-005, "Natural Circulation Cooldown."
- RVLIS upper range indicates greater than 100%.
- Both CRDM fans have been running during the entire cooldown.
- RCS cold leg temperatures are 190 °F.
- Steam generator pressures are 50 psig.

Steam should be dumped from all SGs to ensure ...

a. boron concentration is equalized throughout the RCS prior to taking a sample to verify cold shutdown boron conditions.

b. all inactive portions of the RCS are below 200 °F prior to complete RCS depressurization.

c. RCS and SG temperatures are equalized prior to any subsequent RCP restart.

d. RCS temperatures do **NOT** increase during the required 29-hour vessel soak period.

Answer:

b. all inactive portions of the RCS are below 200 °F prior to complete RCS depressurization.

QUESTION NUMBER: 92
TIER/GROUP: *RO* 1/1 *SRO* 1/1
K/A: WE09/10EK3.1

Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Operations) Facility characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity

K/A IMPORTANCE: *RO* 3.3 *SRO* 3.6
10CFR55 CONTENT: *55.41(b) RO* 5 *55.43(b) SRO*

OBJECTIVE: EPP-005-03

EXPLAIN the basis for selected steps, precautions, and limitations associated with EPP-5.

REFERENCES: EPP-005

SOURCE: *New* ☐ *Significantly Modified* ☐ *Direct* ☒

Bank Number HNP-RO-1998 53

JUSTIFICATION:

a. Plausible since this action would have been performed in this procedure, but must be completed prior to depressurizing the RCS below 1900 psig.

b. CORRECT SG pressure above 0 psig indicates that the SGs are above 200 °F. Depressurizing the RCS under this condition will result in additional void formation in the SG u-tubes.

c. Plausible since RCP operation throughout NC Cooldown is desirable, but will not be performed at this point in the procedure.

d. Plausible since a soak period is addressed, but only if continued operation of both CRDM fans had not been maintained.

DIFFICULTY:
Comprehensive/Analysis ☒ *Knowledge/Recall* ☐ *Rating* 3

Application of plant conditions, using steam tables as needed, to determine if NC procedural requirements are met

REFERENCES SUPPLIED: Steam Tables

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
40.	Determine If RHR System Can Be Placed In Service:	
	a. Check RCS temperature - LESS THAN 350°F	a. WHEN RCS temperature is less than 350°F, THEN Go To Step 40.b.
	b. Check RCS pressure - LESS THAN 375 PSIG	b. WHEN RCS pressure is less than 375 psig, THEN Go To Step 40.c.
	c. Place RHR system in service using Supplement I	
41.	Continue RCS Cooldown To Cold Shutdown	
42.	Continue Cooldown Of Inactive Portions of RCS As Follows:	
	a. Verify both CRDM Cooling Fans - RUNNING • HVH-5A • HVH-5B	a. Perform the following: 1) Maintain RCS temperature less than 212°F for 29 hours. 2) Go To Step 42.c.
	b. Cool upper head region using Both CRDM Cooling Fans • HVH-5A • HVH-5B	
	c. Cool S/G U-tubes by dumping steam from all S/Gs until the S/Gs have stopped steaming	
43.	Check Cooldown Status - ALL REQUIREMENTS OF STEP 42 SATISFIED	WHEN all requirements met, THEN observe CAUTION prior to Step 44 and Go To Step 44.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Depressurizing the RCS before the entire RCS is less than 200°F may result in void formation.

****

44.	Determine If RCS Depressurization Is Permitted:	
	a. Check entire RCS - LESS THAN 200°F	a. Do NOT depressurize RCS. Go To Step 41.
	b. Return to procedure and step in effect as determined by Plant Operations Staff	

- END -

Question: 53

A reactor trip occurred due to a loss of offsite power. The plant is being cooled down on RHR per EPP-006, Natural Circulation Cooldown with Steam Void in Vessel with RVLIS.

- RCS cold leg temperatures are 190°F.
- Steam generator pressures are 50 psig.
- RVLIS upper range indicates greater than 100%.
- Three CRDM fans have been running during the entire cooldown.

Steam should be dumped from all SGs to ensure ...

A. boron concentration is equalized throughout the RCS prior to taking a sample to verify cold shutdown boron conditions.

B. all inactive portions of the RCS are below 200°F prior to complete RCS depressurization.

C. RCS and SG temperatures are equalized prior to any subsequent RCP restart.

D. RCS temperatures do not increase during the required 29 hour vessel soak period.

Answer:

B all inactive portions of the RCS are below 200°F prior to complete RCS depressurization.

Question: 93

Given the following conditions:

- The unit is operating at 100% power.
- A release is in progress from Waste Gas Decay Tank 'A'.
- A loss of Instrument Bus 3 occurs, requiring termination of the release.

Which ONE (1) of the following describes how the release is terminated as a result of the loss of the Instrument Bus?

a. Automatically due to the loss of R-14, Plant Vent Monitor

b. Manually due to the loss of R-14, Plant Vent Monitor

c. Manually due to the loss of power to the Waste Disposal Boron Recycle Panel

d. Automatically due to the loss of power to the Waste Disposal Boron Recycle Panel

Answer:

a. Automatically due to the loss of R-14, Plant Vent Monitor

QUESTION NUMBER: 93

TIER/GROUP: ***RO*** 2/1 ***SRO*** 2/1

K/A: 071A2.05

Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System ; and (b) use procedures to correct, control, or mitigate the consequences: Power failure to the ARM and PRM systems

K/A IMPORTANCE: ***RO*** 2.5 ***SRO*** 2.6

10CFR55 CONTENT: ***55.41(b) RO*** 11 ***55.43(b) SRO***

OBJECTIVE: AOP-024-08

Given plant conditions EVALUATE the appropriate actions to mitigate consequences of loss of an Instrument Bus as directed in AOP-024.

REFERENCES:
AOP-024
EDP-008
SD-019
AOP-005
RMS Lesson Plan

SOURCE: ***New*** ☐ ***Significantly Modified*** ☐ ***Direct*** ☒

Bank Number AOP-024-08 001

JUSTIFICATION:

a. **CORRECT** Instrument Bus 3 supplies power to R14. Loss of power to R14 will cause RCV-014 to close and terminate the release.

b. Plausible since Instrument Bus 3 supplies power to R14, but the release will terminate automatically.

c. Plausible since a WDBRP Trouble alarm is received, however no significant WDBRP power is lost and the release terminates automatically due to the loss of R14.

d. Plausible since a WDBRP Trouble alarm is received and the release is terminated automatically, however no significant WDBRP power is lost and the release terminates due to the loss of R14.

DIFFICULTY:

Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Comprehension of the effect of a loss of power during a gaseous release

REFERENCES SUPPLIED:

CONTINUOUS USE

ATTACHMENT 4

EXTENDED LOSS OF INSTRUMENT BUS 3 (AND 8)

(Page 1 of 3)

NOTE

The following control functions/indications will be lost until Instrument Bus 3 and 8 are restored:

PT-446, Turbine 1st Stage Pressure
FCV-114A, PW to Blender (locked up full open)
FRV A, B, & C Automatic Control
FRV Bypass Valve A (FCV-479)
PCV-455B, Spray Valve (Indication only)
Safeguards Train B Sequencer
FCV-1425 (AFW PUMP B inoperable)
Charging Pump C Controller, SC-153A (locks up)
ICCM - Channel II
Steam Dump Steam Pressure Control
RMS Racks 2 & 3 and R-32B
PT-138 Excess Letdown Pressure Indication
S/G A PORV Control
TCV-1447 and TCV-1448 Exhaust Hood Spray Valves
Solenoids for R-11/12 Skid (fail closed)

1. Place Turbine First Stage Pressure Selector Switch to PT-447 position.

NOTE

In the event that the Plant experiences a trip due to difficulty in maintaining all S/Gs in manual level control, feed flow to the S/Gs will be accomplished via the AFW Pumps OR FRV Bypass Valves.

2. Continue to operate FRVs A, B, & C in MAN.

3. Contact Operations Staff for availability of a dedicated FRV watch.

INSTRUMENT BUS NO. 8

Location: Safeguards Room, East Wall
Power Supply: Instrument Bus No. 3, Ckt 10 (fused panel)

CKT	FUSE SIZE	FUSE TYPE	LOAD
1			Emergency Response Facility Instrumentation System "MUX Cabinet 2" (CWD 1499)
2			RMS Console No. 1, 2, & 3 (NOTE 1) (CWD 83, 574)
3			Hagan Rack 8 (CWD 417); PI-156A (CWD 475); FR-154A pen 1, FR-154B pen 1 (5379-3473); TI-116, PI-117 (CWD 473); PI-445 (CWD 455A); FI-156B (CWD 478); LR-477 pen 3 (5379-3517)
4			Hagan Rack 14 (CWD 460); LI-461 (CWD 460); TI-432C, TI-432D (CWD 411); TI-432B (CWD 413); TI-432A (CWD 408);
5			Hagan Rack 15 (CWD 457); FI-416 (CWD 463); FI-426 (CWD 464); FI-436 (CWD 465); PI-954 , PI-955 (CWD 496, 5379-3504); PI-457 (CWD 457);
6			Hagan Rack 16 (CWD 418); LI-476 (CWD 418); LI-486 (CWD 419); LI-496 (CWD 420, 5379-3485/3515); FR-478 pen 3 (5379-3513); FR-488 pen 3 (5379-3514/3487); FR-498 pen 3 (5379-3515/3485); PI-446 (CWD 428);
7			Hagan Rack 17 (CWD 421); FI-474 (CWD 424); PI-475 (CWD 429); FI-477 (CWD 421); PI-485 (CWD 430); PI-495 (CWD 431); PI-466 (CWD 427)
8			Hagan Rack 18 (CWD 422); FI-484 (CWD 425); FI-487 (CWD 422); FI-494 (CWD 426); FI-497 (CWD 423);
9			RTGB receptacles, Sections A, C and D (CWD 114, 459, 479, 481, 963 , 964); TR-448 (CWD 114, 964)
10			FT-110E/I Boric Acid Bypass Flow, FI-110 (CWD 474)
11			Pressurizer Spray Valve PCV-455B position lights (CWD 470)
12			TI-580, LI-802, PI-957, PI-8111-2 (CWD 533)
13	N/A	N/A	SPARE
14			FQ-958B CV Spray Flow, FI-958B (CWD 494B)
15	N/A	N/A	SPARE
16			V2 Safeguard relay (Rack 63) (CWD 397)
17	N/A	N/A	SPARE
18			Channel II CET/CCM Signal Processor Cabinet TM-578 (CWD 1700); TI-433 & pen 3 on TR-413 (5379-3502)
19	N/A	N/A	SPARE
20			FI-1425A, FI-1426B (AFW) (CWD 623A, 623B)
21	N/A	N/A	SPARE
22			Excore Neutron Flux Detector System Channel N-52, NI-52A, NI-52B, NR-53 pens 3&4(CWD450C&D)
23	N/A	N/A	SPARE
24	N/A	N/A	SPARE
25			Boric acid heat trace Local Annunciator No 3 (CWD180C);
26			Boric acid heat trace recorder No. 3 (CWD 180B)
27			CV Ave Temp Channel TI-950, TI-950B (CWD 044)
28			Aux. panel "GB" TB 5 and 6 (turbine auto stop trip) (CWD 711)
29			Reactor Protection Rack 55 (CWD 438)
30			Reactor Protection Rack 60 (CWD 438)
31			Turbine oil temperature TT-2097, TI-2097A (CWD 726)
32			PQ 4005 Turbine main oil pump discharge (CWD 726)
33			Hydrogen control cabinet electronics , PI-1900 Gen H2 press, AI-1900 Gen H2 Purity (CWD-876)

NOTE 1: RMS console No. 1: R-32B and ERFIS Multiplexers for RMS
RMS console No. 2: R-11 & 12, R-14A-E digital displays, R-15, R-16, and the RMS recorder
RMS console No. 3: R-17, R-18, R-19A-C digital displays, R-20, R-21, R-30, R-31A-C.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 13

PROCESS MONITOR R-14 - PLANT EFFLUENT

(Page 1 of 2)

NOTE

When PLANT EFFLUENT NOBLE GAS LOW RANGE RI-14C reaches a predetermined high level, the monitor will default to 1M CPM and valid readings will only be displayed on RI-14D and RI-14E.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Waste Gas Decay Tank Release - IN PROGRESS	Go To Step 6.
2.	At The WDBRP Verify RCV-014, WASTE GAS DECAY TANK RELEASE ISOLATION Valve - CLOSED	
3.	Do Not Restart Release Until Cause Of High Radiation Alarm Is Determined And Corrective Actions Are Complete	
4.	Request E&RC To Perform The Following, As Applicable: • Resample Waste Gas Decay Tank aligned for release • Perform background radiation survey for Radiation Monitor R-14	
5.	Go To Step 10	
6.	Start One Of The Following AUX BLDG CHARCOAL EXH FANs: • HVE-5A • HVE-5B	
7.	At The WDBRP, Check All Waste Gas Decay Tank Pressure Indications - ANY UNEXPLAINED OR UNCONTROLLED DECREASE	Go To Step 10.

b. Control Room indications while in the Low Range Flow Path:

- Digital readouts for R-14A, R-14B, and R-14C will be reading actual radiation levels.
- Digital readouts for R-14D and R-14E will be defaulted to a minimum value (approximately 10 CPM).

c. When the noble gas activity for R-14C exceeds a precalculated setpoint (well above the normal R-14C High radiation alarm setpoint) R-14 Skid sample flow will be diverted through channels R-14D and R-14E (High Range Flow Path). The value of this switch over setpoint, always in the upper decade of R-14C range, depends on the overlap of R-14C and R-14D. RCV-014 automatically closes as a result of an alarm on R-14C.

NOTE: Grab samples results must be used from analysis of the Particulate and Iodine pre filter in the High Range Flow Path.

d. Control Room indications while in the High Range Flow Path:

- Digital readouts for R-14A and R-14B will be erroneous since there will be no flow through these channels.
- Digital readout for R-14C will default to a maximum value (approximately 1M).
- Digital readout for R-14D should read 50 CPM or greater.
- Digital readout for R-14E should read its true background or higher (10 CPM to 50 CPM).
- R-14C HIGH Alarm light (auto functions will be activated).

e. When the noble gas activity for R-14D goes below a pre-calculated setpoint the R-14 Skid sample flow will be diverted back through channels R-14A, R-14B and R-14C (Low Range Flow Path). The value of this switch over setpoint, always in the lower decade of R-14D range, depends on the overlap of R-14C and R-14D.

6. Stack Isokinetic Flow Measurement

The purpose of isokinetic sampling is to draw a representative sample of particles in an air or gas stream at the same rate (velocity) at which the air or gas flow through the stack. Isokinetic sampling means that the R-14 skid flow is proportional to the flow rate in the plant stack (1:30,000 ratio).

HBR uses Kurz' unique eight-point, independent, individual velocity sensors mounted on two orthogonal 316 stainless steel bars to measure the Plant Stack flow. Two bars are used because of the geometry of the stack (i.e.elbow and feeder ducts). This design minimizes particle loss and optimizes sample flow

ATTACHMENT 10.2
Page 1 of 2

RMS INSTRUMENT CONTROL FUNCTIONS

MONITOR	MEDIUM MONITORED	FUNCTION
R-1	Control Room Air	Switches Control Room ventilation into the emergency pressurization operating mode.
R-11	CV Air or Stack Particulate	Closes C.V. purge supply and exhaust; pressure and vacuum relief valves.
R-12	CV Air or Stack Gas	Same function as R-11
R-14C	Stack Gas (Low Range)	Closes waste gas decay tank release valve (RCV-014); swaps R-14 Skid over to high range (two different setpoints).
R-14D	Stack Gas (Mid Range)	Swaps R-14 Skid over to low range.
R-18	Liquid Waste Disposal	Closes waste disposal system liquid release valve (RCV-018)
NOTE The blowdown tank release isolation valve (V1-31) will close if all three SG monitors (R-19A, R-19B and R-19C) are in alarm.		
R-19A	SG "A" Blowdown	Closes; blowdown isolation valves FCV-1930A & FCV-1930B, sample isolation valves FCV-1933A & FCV-1933B, rate flow control valve FCV-4204A.

LESSON BODY	KEY AIDS
I. **<u>GENERAL DESCRIPTION</u>**	
A. **SYSTEM PURPOSE**	RM-TP-1 **OBJ. #1**
1. Sections 2.1 of SD-019	
B. **SYSTEM FLOWPATHS AND BASIC OPERATION**	RM-FIGURES-1,2,3, & 4
1. Section 2.2 and 2.3 of SD-019	
II. **<u>COMPONENT DESCRIPTION</u>**	
Components of the Radiation Monitoring System include: - Area Radiation Monitoring System - Process radiation Monitoring System - Accident Radiation Monitors	
A. **AREA RADIATION MONITORING SYSTEM**	**OBJ. #4,5**
1. Section 3.1 of SD-019	**OBJ #6**
2. Power Supply is IB #7	RM-FIGURES 5,6,&7
B. **PROCESS MONITORS**	RM-FIGURES-5, 8 thru 14
1. Section 3.2 of SD-019	
2. Section 3.2.4 of SD-019 for individual channels	RM-FIGURES-15 thru 22
3. Power Supply is IB #8	**OBJ. #6**
C. MISCELLANEOUS EQUIPMENT	RM-FIGURE-21
1. Section 3.3 of SD-019	
D. POWER SUPPLIES TO RMS COMPONENTS	
1. Instrumentation Bus 7A	

Question: 94

Which ONE (1) of the following conditions related to the Pressurizer would require entry into a Technical Specification action or a Technical Requirment Manual compensatory action, as applicable?

a. Pressurizer level is 68% with the plant operating at 8% power

b. Pressurizer pressure is 2184 psig with the plant operating at 2% power

c. SST-2A Disconnect, used to supply emergency power to the pressurizer heaters from EDG 'A', is removed from service for maintenance with the plant operating at 35% power

d. Auxiliary Spray, at 400 °F, is used to depressurize the RCS from 2235 psig, resulting in a cooldown rate of the Pressurizer of 135 °F per hour

Answer:

a. Pressurizer level is 68% with the plant operating at 8% power

QUESTION NUMBER: 94
TIER/GROUP: *RO* 2/2 *SRO* 2/2
K/A: 010 2.1.33

Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications (Pressurizer Pressure).

K/A IMPORTANCE: *RO* 3.4 *SRO* 4.0
10CFR55 CONTENT: *55.41(b) RO* 10 *55.43(b) SRO*

OBJECTIVE: PZR-12

STATE the Technical Specification Limitations and explain the bases for the PZR and PRT.

REFERENCES:

TS 3.4.1
TS 3.4.9
TRMS 3.4
SD-059

SOURCE: *New* [X] *Significantly Modified* [ ] *Direct* [ ]

Bank Number NEW

JUSTIFICATION:

a. **CORRECT** TS limit is 63.3% for Mode 1 operation and 92% for Mode 2 and 3. Since the plant is operating at 8%, the Mode 1 limit applies.

b. Plausible since this would require an entry into TS 3.4.1 if the plant was in Mode 1, but at 2% power the plant is in Mode 2 where the TS does not apply.

c. Plausible since at least 125 KW of heaters capable of being supplied by an emergency source are required, but this condition only renders one set of the heaters inoperable and the other can still provide > 125 KW.

d. Plausible since a limit exists for both the differential temperature between spray and the steam space and a cooldown limit, but both limits (320 °F spray differential and 200 °F per hour cooldown) are met.

DIFFICULTY:

Comprehensive/Analysis [X] *Knowledge/Recall* [ ] *Rating* 3

Analysis of conditions to determine if TS and / or TRM limits for pressurizer are met

REFERENCES SUPPLIED:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2205 psig;

b. RCS average temperature ≤ 579.4°F; and

c. RCS total flow rate ≥ 97.3 x 10^6 lbm/hr.

APPLICABILITY: MODE 1.

--------------------------NOTE--------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 63.3% in MODE 1;

b. Pressurizer water level ≤ 92% in MODES 2 and 3; and

c. Pressurizer heaters OPERABLE with a capacity of ≥ 125 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. Capacity of required pressurizer heaters ≤ 125 kW.	B.1 Restore required pressurizer heaters to OPERABLE status.	72 hours
C. Required pressurizer heaters not capable of being powered from an emergency power supply.	C.1 Restore capability to power the required pressurizer heaters from an emergency power supply.	72 hours.

(continued)

3.4 PRESSURIZER HEATUP AND COOLDOWN LIMITS

TRMS 3.4 (CTS 3.1.2.3)

a. The maximum heatup rate of the pressurizer shall be ≤ 100°F/hr and the maximum cooldown rate of the pressurizer shall be ≤ 200°F/hr.

AND

b. Pressurizer spray shall not be used if the temperature between the pressurizer and the spray fluid is > 320°F.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirements of TRMS not met.	A.1 Initiate action to restore compliance With the TRMS.	Immediately
	AND	
	A.2 Initiate a Condition Report in accordance with the Corrective Action Program.	Immediately

TEST REQUIREMENTS

TEST	FREQUENCY
None.	NA

condition requiring venting of non-condensable gases from the PZR, a connection to the High Point Vent System is provided.

3.0 COMPONENT DESCRIPTION

3.1 PZR (PZR-Figure 2)

Design pressure	2485 psig
Design temperature	680°F
Water Volume, full pwr.	780 ft^3 (60% of net interval Volume)
Steam Volume, full pwr.	520 ft^3
Shell I.D.	84 in.
Minimum Shell thickness	4.1 in.
Minimum Clad thickness	0.188 in.
Steady State heat losses	90 KW

The PZR is a vertical, cylindrical vessel with a surge line in the bottom from RCS loop C hot leg, and a spray line in the top from RCS loops B and C cold legs. Electrical immersion heaters penetrate the lower head. PORVs and safety valves are connected to the upper head.

3.2 PZR Heaters

Capacity total	1300 KW (Minimum KW required = 800)
Control Group	400 KW
Backup Group #1	450 KW
Backup Group #2	450 KW

The electrical heaters installed in the PZR are replaceable, direct immersion, tubular sheath type, that are hermetically sealed. Located in the lower portion of the vessel, they heat and maintain the steam and water contents at equilibrium (saturated) conditions. The heater bundle consists of 78 individual heater elements rated at 16.67 KW each.

The control heaters consist of eight banks with three heaters per bank (400 kW). The purpose of the control heaters is to make up for the ambient heat losses. The power to this group is controlled in inverse proportion to PZR pressure when they are on.

Two groups of backup heaters are provided. Each group consists of nine banks with three heaters per bank (450 kW) and are turned on and off by PZR pressure signals when in the automatic mode of operation. Controls are also provided for manual operation.

The control bank and both backup groups are operated from the RTGB.

Power Supply: Control Bank - 480V BUS 2B

Backup Group A - 480V BUS 1
Backup Group B - 480V BUS 2A

The capability exists to power 150 kW of PZR heaters from Emergency Bus E1 and another 150 kW of heaters from emergency bus E2. This capability would be used during a loss of offsite power event to ensure proper RCS pressure control capability is maintained. The power supply must be manually transferred to the selected emergency bus following the loss of offsite power to ensure that the PZR temperature remains above the RCS temperature. Once the power supply is transferred, the heaters are controlled from the RTGB. If the PZR heaters are being powered from one of the emergency busses, they will automatically trip upon receipt of a Safety Injection Signal, to ensure the Emergency Diesel Generators are not overloaded by these non-safety related loads. This trip feature is enabled by the PZR Heater "Arm" switch in the E1/E2 room. PZR control group heaters can also be energized from the DS bus in the event of a loss of all AC power.

3.3 PZR Spray Lines

Spray nozzle press drop at max. flow	15.0 PSI at 70°F
Continuous spray rate	1 gpm
Pipe Diameter	4 in.
Pipe Schedule	160
Design Pressure	2485 psig
Design Temperature	650°F

The PZR spray system is designed to pass a total flow of 600 gpm, 300 gpm per valve. The driving force of the spray water is a combination of the differential pressure between the hot and cold legs and the velocity head obtained by using a scoop in the reactor coolant piping.

The spray nozzle, which is also protected with a thermal sleeve, is connected to the head of the PZR. It is designed to produce a narrow angle cone spray pattern which prevents cold water impingement on the PZR walls.

The spray water is drawn from cold legs of loops B and C. The two lines tie together downstream of the control valves, form a loop seal, and supply water through a single spray nozzle. The loop seal is provided to prevent the backup of steam into the piping when the spray valves are closed. A small continuous spray flow is provided, by means of the throttle valves (needle valves) which bypass the spray valves, to help ensure that the PZR liquid is in chemical equilibrium with the rest of the reactor coolant system(RCS) and to prevent thermal shock of the spray piping and the auxiliary spray connection.

Question: 95

Given the following conditions:

- The unit is operating at 70%.
- Rod Control is in AUTO.
- Bank 'D' control rods are at 195 steps.
- Tref is 566.9 °F.
- Loop Tavgs are:

LOOP	T-AVG
'A'	569 °F
'B'	567 °F
'C'	566 °F

Which ONE (1) of the following failures will cause control rods to step inward?

a. Loop 1 Thot fails high

b. Loop 1 Tcold fails low

c. Loop 2 Tcold fails high

d. Loop 3 Thot fails low

Answer:

c. Loop 2 Tcold fails high

QUESTION NUMBER: 95
TIER/GROUP: ***RO*** 2/2 ***SRO*** 2/2
K/A: 016K3.01

Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: RCS

K/A IMPORTANCE: ***RO*** 3.4 ***SRO*** 3.6
10CFR55 CONTENT: ***55.41(b) RO*** 6 ***55.43(b) SRO***

OBJECTIVE: AOP-001-02

EXPLAIN the basis of selected steps, cautions, and notes in AOP-001.

REFERENCES: SD-007

SOURCE: ***New*** ☒ ***Significantly Modified*** ☐ ***Direct*** ☐

Bank Number NEW

JUSTIFICATION:

a. Plausible if misconception is that average Tavg is used as average Tavg will increase, but median Tavg is used which will still be loop 2.

b. Plausible since this will cause loop 3 to be the median Tavg, but loop 3 is below Tref so no rod motion will occur.

c. **CORRECT** Rod control uses median Tavg. Currently, loop 2 is the median. When loop 2 Tcold fails high, loop 2 will become the high channel and loop 1 will be the median. Loop 1 is more than 2 degrees above Tref, so inward rod motion will occur.

d. Plausible if misconception is that average Tavg is used as average Tavg will increase, but median Tavg is used which will still be loop 2.

DIFFICULTY:
Comprehensive/Analysis ☒ ***Knowledge/Recall*** ☐ ***Rating*** 3

Analysis of the effect of a temperature failure on the Rod Control system

REFERENCES SUPPLIED:

temperature. There is a direct relationship between RCCA position and power and it is this relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor. There are two alarm setpoints to alert the operator to take corrective action in the event a control group approaches or reaches its lower limit.

Any unexpected change in the position of the control group under automatic control or a change in coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor core depletion.

5.5.2 Shutdown Groups Rod Control

The shutdown groups of RCCAs together with the control groups are capable of shutting the reactor down. They are used in conjunction with the adjustment of chemical shim and the control groups to provide shutdown margin of at least one percent following reactor trip with the most reactive RCCA in the fully withdrawn position for all normal operating conditions.

The shutdown groups are manually controlled during normal operation and are moved at a constant speed. Any reactor trip signal causes them to fall into the core. They are fully withdrawn during power operations and are withdrawn first during startup. Criticality is always approached with the control groups after withdrawal of the shutdown groups.

5.5.3 Manual Control Group Rod Control

Manual rod control is used during plant operation below 15% power and may be used at anytime. With the bank selector switch in the manual position, the operator can move the rods in the out or in direction using the IN-HOLD-OUT switch. The control banks will move sequentially (in overlap) as long as the IN-HOLD-OUT switch is held in any position except HOLD.

5.5.4 Reactor Control System, T_{AVG} Control (Refer to RDCNT-Figure-22 & 23)

The automatic Rod Control System maintains the average reactor coolant temperature (T_{AVG}) by adjusting the RCCA positions. The Reactor Control System, T_{AVG} Control, is provided as part of the automatic Rod Control System to develop the necessary signals to provide automatic control of the RCCAs during power operation of the reactor. The system uses input signals including neutron flux, coolant temperature and pressure, and the plant turbine load. The Chemical and Volume Control System (CVCS) supplements the Reactor Control System by the addition and removal of varying amounts of boric acid. The ultimate goal of the automatic Rod Control System is to manipulate the control rods in order to maintain the T_{AVG} consistent with the reference temperature (T_{REF}). The automatic withdrawal portion of the Rod Control System has been defeated.

The T_{AVG} Control Unit develops rod speed and direction demand signals for the Logic Cabinet (when in AUTOMATIC control) from two error signals, the sum of which is input to the rod speed programmer which produces a speed demand signal.

The two channels used to generate the total error signal are: the deviation of the median of the three reactor coolant system average temperatures (T_{AVG}) from the programmed average temperature (T_{REF}) and the mismatch between turbine load and nuclear power.

The programmed average temperature (T_{REF}) represents the desired reactor coolant system average temperatures (T_{AVG}) based on turbine load. The selected turbine first stage pressure channel, PT-446 or PT-447, provides the turbine load input to the T_{REF} program. The output of the T_{REF} program is a linear function as follows:

- 0% load = 547°F
- 100% load = 575.4°F
- High limit = 575.4°F
- Low limit = 547°F
- 0.284°F/% load

5.5.4.1 Average Temperature Channel

The T_{AVG} channel (T_{AVG} T_{REF}) functions to provide fine control during steady state operations. When power is essentially constant, the power mismatch channel provides no input. Under this condition the summing unit just compares T_{REF} to T_{AVG} and generates a corresponding error signal. If this error signal exceeds the prescribed dead band (+ 0.5, - 2.5°F) rod direction will be determined and motion will be initiated.

5.5.4.2 T_{AVG} Channel

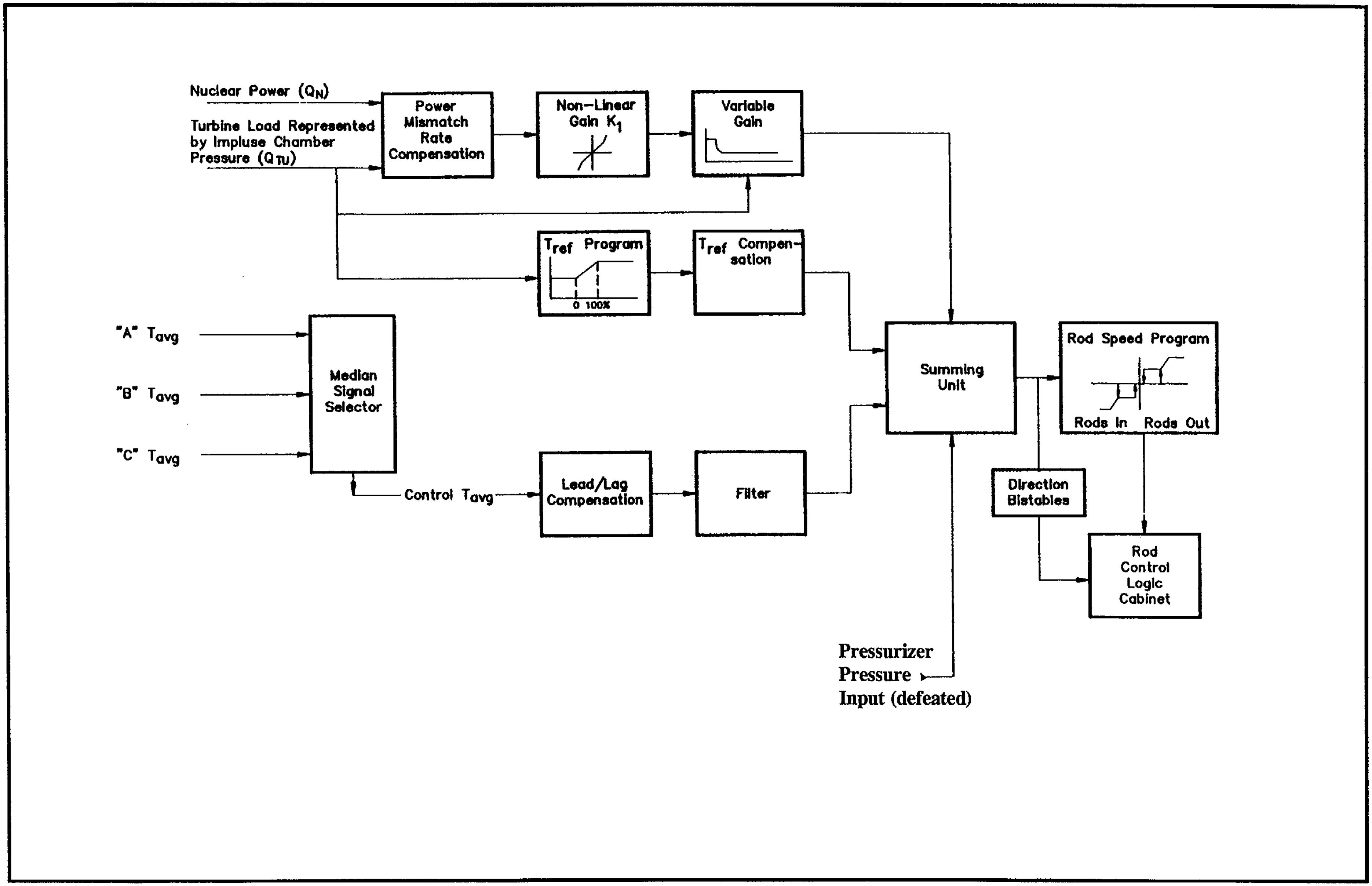
The T_{AVG} Median Signal Selector (MSS) receives an isolated input from each protection grade loop T_{AVG}. The MSS selects the median of the three loop T_{AVG} measurements and supplies this signal to the automatic rod control program as well as to other control systems.

5.5.4.3 Power Mismatch Channel

This channel provides fast response to a change in load (by means of the turbine load feed-forward signal) as well as control stability (by means of the nuclear power feedback signal in cases where the moderator coefficient is zero or slightly negative).

TAVG CONTROL BLOCK DIAGRAM

RDCNT-FIGURE-22 (Rev . 0)



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